TECHNISCHE UNIVERSITÄT DRESDEN

WKET

Faculty of Mechanical Science and Engineering Institute of Power Engineering

TRAINING REACTOR AKR-2

EXPERIMENTS

IN REACTOR PHYSICS AND RADIATION PROTECTION

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TRAINING COURSE AT AKR-2

Experiments in Reactor Physics and Radiation Protection

Content:

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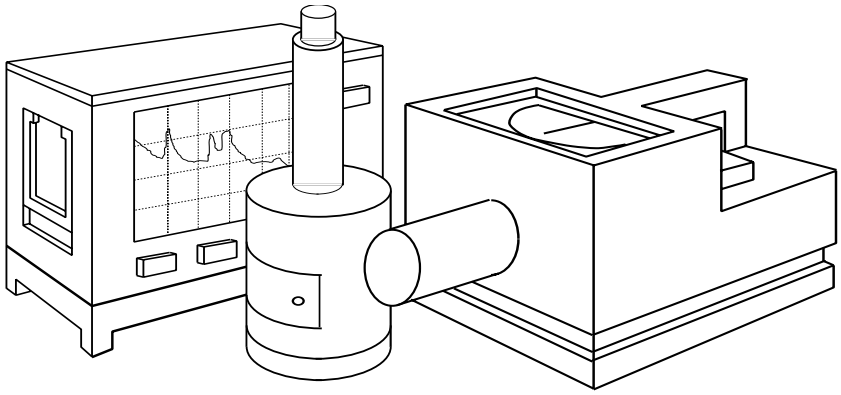
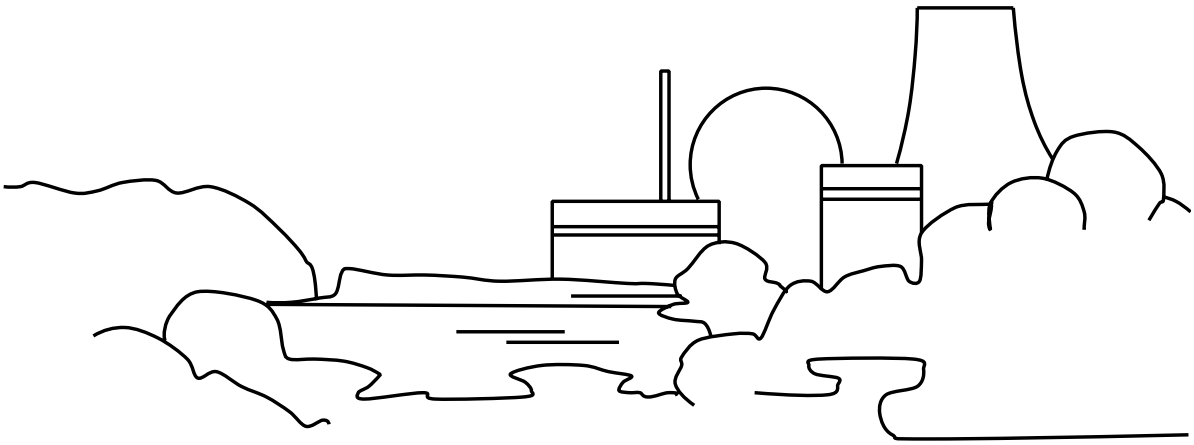


Training Reactor AKR-2

Description of the Reactor Facility
Procedure of Operation



Technical Description and Procedure of Operation for the Reactor Facility AKR-2

Content:

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(issued: March 2015)

1. Description of the Training Reactor AKR-2

1.1. Introduction

The training and research reactor AKR-2 (from the German **A**usbildungs**k**ern**r**eaktor) of the Technical University Dresden is a thermal, homogeneous, solid moderated zero-power reactor with maximum continuous (thermal) power of 2 Watt.
That means:
- nuclear fission is mainly caused by thermal neutrons,
- nuclear fuel and moderator material are distributed homogeneously in the fuel plates,
- the moderator material is solid polyethylene (and not any liquid like water!),
- the extremely low nuclear power of only 2 Watt ("zero power") allows effects of temperature, fuel burnup, formation of nuclear waste, activation of structural materials, Xe-poisoning and others to be neglected.

Advantages of the design of the AKR are:
- application of low enriched uranium (LEU), i.e. U-235 content in the fuel is < 20 %,
- low absolute amount of nuclear fuel (total mass of U-235 in the reactor < 1 kg),
- desired characteristics of the reactor with respect to its inherent safety features (negative temperature coefficient of reactivity; integrity of the core maintained in case of power excursion),
- simple construction resulting in high reliability, low costs for inspections, maintenance and operation,
- strict avoidance of any liquid in the facility (no corrosion, no danger of contamination in case of leakage, no maintenance of water systems, no substitution or cleaning procedures of liquids),
- miscellaneous irradiation capabilities in the experimental channels ($\Phi_{max} \approx 2.5 \cdot 10^7$ n/(cm² · s)),
- free space above the core for possible installation of additional experimental equipment.

Precursor of the AKR-2 was the AKR-1. The AKR-1 training and research reactor of the TUD was put into operation in 1978 based on the concept of the proven training reactor SUR-100 from SIEMENS company. For more than 25 years, the AKR-1was successfully used for training and education of students, for nuclear research projects, and as an information centre for the public.

In 1998, a new licensing procedure began with the goal of completing a comprehensive refurbishment of the reactor in order to update the facility concerning the current level of science and technology.

The refurbishment took place in 2004 and included civil work as well as new electrical and I&C equipment, based on the digital reactor protection instrumentation and control system TELEPERM XS (by AREVA GmbH Erlangen, Germany) and new installation of 3 channels for the neutron flux monitoring: two (logarithmic) wide-range and one (linear) power-range channel.

On the other hand, the nuclear design of the reactor itself remained the unchanged and no internal components or structures (reactor core, control rods, reflector, shielding, neutron detectors etc.) had to be replaced with the exception of a few minor details. Also the same nuclear fuel can be used furthermore for the whole reactor life expectancy.

After having successfully completed all start-up requirements (nuclear commissioning with the critical experiment on March 23, 2005), the regulatory body authorized normal operations of the new AKR-2 reactor in April 2005.

Currently, AKR-2 is the most up-to-date training reactor in Germany.

Main purpose of the AKR-2 is education and training of students in nuclear and reactor physics as well as in nuclear engineering and radiological protection. Fundamental experiments are provided and carried out in practical exercises (see chapters 3 and 4 in more detail).

According to its physical properties, the AKR is also included in specific research projects of the institute or is used as irradiation facility.

Additionally, the reactor facility and its equipment is also used as an general information centre providing practical experience in nuclear and reactor physics, nuclear energetics and radiation protection for
- interested students of all faculties of the university,
- pupils of the grammar schools of Dresden City and its surroundings or
- organised groups and individuals from the public.

1.2. Safety Features

Safe and foolproof operation of the reactor is guaranteed by a combination of
- inherent safety,
- designed safety and
- administrative procedures.

The design and operation conditions of the AKR are based upon the following considerations:
- prompt criticality must not occur at all and
- undue increase of the power must not endanger the operators, the environment or the reactor itself.

The safe nuclear operation of the reactor is secured by the following measures:
- The excess reactivity is restricted to a maximum value of $\Delta k/k \leq 0.3$ %.
- All three control rods are designed as combined control and safety rods. The reactivity value of each of them is sufficiently high to shut-down the reactor. Thus, even in case of failure of single rods the AKR will be shut-down reliably.
- Independently on control rod movement, a separation of the two core sections by 50 mm reduces the reactivity by $\Delta k/k = - 5.8$ %. In case of a scram this negative reactivity becomes effective within about 100 ms. It ensures a fast shut-down of the reactor and a high degree of nuclear safety in the shut-down state.
- The temperature coefficient of reactivity is negative. From measurements a value of (2.90 ± 0.05) ¢/K was found out. Estimations have shown that power excursions would be self-limiting before damages occur at the reactor itself or in its environment. This high inherent safety results from the physical properties of the fuel elements.
- By monitoring the core temperature and including the measured value into the protective logic

circuit it is prevented that additional positive reactivity could be introduced due to decreasing core temperature.
- All drives were dimensioned in such a way that the rates of reactivity changes are below 0.0001 s^{-1}.

Due to these measures, a power excursion with harmful consequences can be virtually excluded.

1.3. Radiation Protection

The reactor is designed to keep the external as well as the inner radiation exposure for the reactor operational staff and the people in the surroundings of the AKR-2 down to be negligible.

The operation of the AKR-2 as a zero-power reactor (or a so-called critical assembly) with extremely low nuclear power causes only a negligible low uranium fuel consumption and hence, a minimum production of radioactive fission products. After 30 years estimated continuous non-stop operation at 2 Watts power the resulting activity in the facility would be less than 100 GBq.

The multiple barrier concept with the capsuled core and the reactor tank as an additional subatmospheric pressure barrier guarantees that no fission products can be released into the surrounding of the reactor.

For supervising the ambient local dose rate around the reactor there are 5 gamma-ray measuring channels installed. If well defined dose rate levels would be exceeded, it would be indicated by optical as well as acoustic alarm signals and would be registered in the logfile.

1.4. Review of Technical Parameters of the AKR-2

Reactor Vessel:	Total Diameter:	250 cm
	Total Height:	350 cm (reactor vessel 280 cm)
Fuel Elements:	Plates:	ϕ 250 mm, height 2 - 23 mm
	Material:	homogeneous mixture from Uranium oxide (UO_2 , enrichment 19.8 %) and PE
	Critical Mass:	794 g U-235
Reflector:	Material:	pure graphite (ρ = 1.75 g/cm^3)
	Thickness:	32 cm radial, 20 cm axial
Neutron Source:	Am-Be:	source strength: $2.2 \cdot 10^6$ n/s
Control Rods:	3 independent combined control and safety rods	
	Material:	Cadmium (0.5 mm thick sheets)
	Total reactivity:	about 37 ¢ per rod

Nuclear Instrumentation:

2 (logarithmic) wide-range channels with fission chamber detectors
1 (linear) power-range channel with γ-compensated ionisation chamber

Biological Shield:

axial:	48 cm heavy concrete
radial:	15 cm paraffin and 58 cm heavy concrete

Experimental Irradiation Channels:

horizontal:	4, $\Phi_{max} \approx 2.5 \cdot 10^7$ n/(cm^2 · s)
vertical:	2

Fast Shut-Down Possibilities: drop down of the lower core section for about 5 cm
drop down of the 3 control rods

Maximum Excess Reactivity: 0.3 %

Maximum Continuous Power: 2 W

1.5. Technical Design of the Facility

The design of the reactor is shown in Figs. 1 and 2 (vertical and horizontal cross section). The cylindrical core has a diameter of 250 mm and a critical height of 275 mm. The disk-shaped fuel elements consist of a homogeneous dispersion of polythene and uranium oxide (19.8 % enriched in Uranium-235, O/U ratio 2.27). The Uranium-235 density in the fuel elements amounts to 0.060 g/cm³.

The core is completely surrounded by a graphite reflector (density 1.75 g/cm³). The axial and radial thicknesses are 20 cm and 32 cm, respectively. Therefore, the critical mass is relatively small (about 790 g U-235). Within certain constraints the AKR is a minimum-critical-mass reactor.

For safety reasons, the core consists of two separable sections. The fuel elements of each section are enclosed in a hermetically sealed aluminium container. A second, larger gas-tight reactor tank encloses both, core sections and parts of the reflector. The pressure inside the reactor tank is lowered by (8…18) kPa compared to the ambient atmospheric pressure. This subatmospheric pressure barrier prevents an uncontrolled leakage of radioactive fission products even in the unlikely case that all the other internal retention barriers would fail. This subatmospheric pressure in the tank is continously maintained by a pre-vacuum pump. Control is automatically performed by means of a pressure controller even when the reactor is shut down.

In the shut-down position the distance between the lower and the upper core section is about 50 mm. The lower section is lifted by means of a core drive mechanism including an electro-magnetic holder of the core section. Through a tube within this mechanism the start-up neutron source (Am-Be, neutron yield $2.2 \cdot 10^6$ s^{-1}) is moved from the source container to the bottom side of the core. The distance between the bottom side of the core and the middle of the source container measures about 800 mm.

The reactor is controlled by three cadmium absorber plates. These plates are moved vertically in a gap within the reflector outside the reactor tank. They are designed as combined control and safety rods.

The lower core section and the control rods are held by electromagnets in their "working" positions. Any scram signal opens the direct-pull holding magnets allowing the rods and the lower core section to fall down in the shut-down position simply by influence of gravity.

Reactor core, reactor tank, graphite reflector and control rods are arranged on a horizontal steel plate (diameter 1000 mm, thickness 30 mm) inside the reactor vessel. This is a cylindrical double-walled vessel containing the biological shield and having an outer diameter of 2500 mm and a height of 2800 mm.

The top cover of the reactor is removable. The free space above the core could be used for e.g. installing a thermal column or a subcritical assembly.

There are four horizontal and two vertical experimental channels with different diameters and shapes. They provide adequate inpile irradiation volume with different neutron spectra.

The permissible power level is limited only by the effectiveness of the biological shield. It consists of paraffin and baryte heavy concrete with an overall thickness of 75 cm inside the reactor vessel. Proceeding from the condition, that the dose rate equivalent just outside the shield should not exceed 10 µSv/h, a continuous 2 Watt operation is possible. At 2 Watt power level the maximum flux density of thermal neutrons in the central experimental channel amounts to about $2.5 \cdot 10^7$ n/($cm^2 \cdot s$).

The reactor vessel is arranged on the ground floor of the building on a separately grounded fundament as additional shock protection e.g. for the unlikely event of an earthquake. The drives for lower core or start-up neutron source movement and the control rods are accessible from the basement.

An overview of the plan of the reactor hall is given in Fig. 3.

The control desk (Fig. 4) and the separate cabinet with the reactor protection system are also located in the reactor hall in about 6 m distance from the reactor. The cabinet hosts the modules of three redundant channels for neutron flux measurement and the computers of the digital reactor protection system (TelepermXS by AREVA GmbH, Erlangen, Germany) whereas the operational units are hosted in the control desk.

The control desk holds all operating elements, signal displays and operational instruments required for the control of the reactor. It has an operation and signal panel with conventional display and operational elements (Fig. 5) and 3 monitor screens for reactor control and supervising. On each of these 3 different screens all operational soft-buttons and parameters (Figs. 6, 7 and 8) can be displayed. Alternatively, I&C system announcements or alerts (example in Fig. 9) or different graphic presentations of reactor parameters can be shown. User access permits are possible by means of key-switches on the conventional operation and signal tableau. The reactor control is done by point-and-click on the corresponding soft-buttons on the screens.In order to avoid unwanted reactivity change (especially positive reactivity rise) all corresponding operations

has to be chosen first and then confirmed by a second mouse-click.

Signals of peripheral technical units (ventilation system, subatmospheric pressure in the reactor tank, local dose rate limits etc.) are displayed on an independent signal panel in the reactor hall next to the control desk (Fig. 10) even in the case of shut-down reactor.

1.6. Instrumentation and Control (I&C) System with TELEPERM XS

1.6.1. Architecture of the I&C System

The I&C system of the reactor comprises
- the nuclear instrumentation and
- the operational instrumentation with the safety and control system (German abreviation: SUS) and the control and display units on the control desk.

Purpose of the measuring channels for neutron flux density (nuclear instrumentation) is to control the reactor in its shut-down subcritical state as well as to supervise the conditions for reactor start-up and operation in the whole power range.
For this purpose, it provides
- a signal, which is proportional to the reactor power and which quickly follows temporary changes,
- a signal, which is proportional to the relative neutron flux exchange speed and thus proportional to the reciprocal reactor period (the so-called RELFAEG signal).

These signals are input parameters for the safety and control system (SUS) and for the control and display units on the control desk. They are used by the safety and control system (SUS) as shut-down criteria or for prevention of reactor start-up in case of reactivity irregularities or malfunction of instrumentation units. Furthermore, these signals are used in the operational instrumentation in order to allow only permitted operations by the operators.

The digital safety system TELEPERM XS (by AREVA GmbH Erlangen, Germany) serves as the basis for the technical solution (reactor protection as well as operational instrumentation).

1.6.2. Nuclear Instrumentation

The nuclear instrumentation consists of three independent redundant channels for neutron flux measurement (digital system TK250, MGP Instruments GmbH Munich, Germany):
- two (logarithmic) wide-range channels (DAK 250-i, pulse measurement with fission chamber detectors),
- one (linear) power-range channel (DAK 250-g, DC measurement with γ-compensated ionisation chamber detector).

The neutron flux channels of the TK250 system were developed for BWR and PWR applications and are designed and certified according to the requirements of the German KTA-3501 safety standards.

1.6.2.1. Wide-Range Channels (Start-up/Middle/Power Range)

The logarithmic wide-range (start-up/middle/power range) monitors with two redundant measuring channels DAK 250-i cover 6 decades (approx. 2 ... 2×10^6 $n\cdot cm^{-2}\cdot s^{-1}$) of neutron flux density at the measuring location outside the graphite reflector. Fission chambers (type 9R100) are used as detectors.

In the safety system, the absolute fission chamber signals and the so-called RELFAEG signals (being reciprocal to the reactor period) are evaluated for the following limits (prevention of the reactor start-up as well as release SCRAM):
- minimum neutron flux required for reactor start-up (1∪2 logic),
- SCRAM signal for exceeding maximum reactor power (1∪3 logic together with the signal of the power-range channel,
- SCRAM signal for too small reactor period (1∪2 logic) or too high relative neutron flux exchange speed (RELFAEG), respectively.

1.6.2.2. Power-Range Channel

The linear power-range channel, DAK 250-g, measures the upper 2 - 3 decades (approximately 5×10^3 - 2×10^6 $n\cdot cm^{-2}\cdot s^{-1}$) of the neutron flux density at a detector position outside the graphite reflector. A γ-compensated ionisation chamber is used as a neutron detector. In the safety system, the chamber signal is evaluated for following limits (release SCRAM):
- SCRAM signal for exceeding maximum reactor power (1∪3 together with the signals of the wide-range channels).

1.6.3. Safety and Control System

The safety and control (German: SUS) system should protect the facility from inadmissible demands and shall minimize the effects on the staff, the surroundings and the reactor in case of accidents. The safety and control system causes automatic reactor shut-down (SCRAM) in the following cases:
- malfunction of drives of control rods, neutron source or core,
- too fast power increase (reactor doubling time < 10 s, warning signal at 20 s),
- too high absolute reactor power (> 2.4 W, warning signal at 2.2 W),
- too low moderator temperature (< 18°C),
- too high pressure in the reactor tank (> -8 kPa),
- in case of malfunctions in the safety and control system,
- non-availability of the I&C computer,
- in case of malfunction in an external experiment (optional),
- in case of malfunctions in the I&C rack,
- an implicit SCRAM is initiated in case of a safety computer failure (fail-safe by closed-circuit connection).

1.6.4. Safety Relevant Protective Logic

1.6.4.1. Protective Logic for Reactor Start-up

The protective logic for reactor start-up (Fig. 11) guarantees that rector operation is only possible in the case that all initial requirements are fulfilled and the correct command sequence is maintained.

Initial conditions are:
- no SCRAM signal, no warning signal, no alarm signal,
- neutron start-up source is in its container AND position signalisation works correctly,
- lower core section is in lower end position AND position signalisation works correctly,
- all control rods are in their end position (shut-down position) AND position signalisation works correctly,
- key-lock for RESA-Reset (SCRAM) was triggered.

If all conditions are fulfilled, the start-up neutron source drive will be released.

The lower core section drive mechanism is released for lifting, if
- start-up neutron source is in its upper end position AND required minimum count rate at the neutron instrumentation was measured (about 2-3 cps).

The drive mechanism for control rod withdrawal is released, if
- lower core section is in its upper end position (upper and lower core sections close together) AND minimum required neutron count rate at the neutron instrumentation was measured (about 2-3 cps) AND key-lock for reactor start-up is in its ON-position.

An interlock circuit guarantees that not more than only one control rod can be withdrawn from the core at the same time.

At a reactor power >0.25 W the start-up neutron source can be driven back down into the container.

1.6.4.2. Protective Logic for the Critical Experiment

In the critical experiment it must be possible to withdraw the control rods even in that case that the lower core section is not in its upper end position.
At first, initial conditions are the same as in 1.6.4.1.:
- no SCRAM signal, no warning signal, no alarm signal,
- neutron start-up source is in its container AND position signalisation works correctly,
- lower core section is in lower end position AND position signalisation works correctly,
- all control rods are in their end position (shut-down position) AND position signalisation works correctly,
- key-lock for RESA-Reset (SCRAM) was triggered.

If all conditions are fulfilled, the start-up neutron source drive will be released.

If the start-up neutron source is in its upper end position AND required minimum count rate at the neutron instrumentation was measured (about 2-3 cps) AND, additionally, the key-lock "Simulation KHZ" was triggered, than the control rods can be withdrawn even in the case that the lower and upper core sections are not connected.Also in this case, the interlock circuit for control rod movement is active, that only one control rod after the other one can be withdrawn.

1.6.5. Operational Functions

1.6.5.1. Automatic Reactor Power Control

At AKR-2, an automatic reactor power control is available with following properties:
- preset of a desired value of reactor power on the monitor screen of the control desk,
- selection of control rod for automatic power control (on the screen of the control desk),
- switch for manual/automatic power control (on the screen of the control desk),
- in the case that the selected control rod could go to its end position without obtaining the desired power value, the next control rod is automatically used for control,
- manual movement of a non selected rod is possible in the automatic mode, too, unless the end position has been reached.

1.6.5.2. Control Rod Drop

The possibility exists, to switch off manually the holding magnets of any single (or all together) control rods via soft-buttons on the monitor screens of the control desk.In that case, the control rod(s) drop into the shut-down position very fast only by gravity. For quality assurance, the drop down time is measured and can be displayed on the monitor screen of the control desk (Fig. 8).

1.6.5.3. SCRAM

Fast reactor shut-down (SCRAM, or in German: RESA) is activated in case of according criteria of the safety and control system (SUS). Additionally, SCRAM is possible at any time by manual activation (red button "Hand-RESA" on the conventional operation and signal panel left on the control desk, see Fig. 5), which has to be released after activation.

For fast reactor shut-down (SCRAM), there are two shut-down systems available in diversity:
- drop-down of the control rods (triple redudancy) and
- disconnection of the two core sections (drop-down of the lower core section)

Both shut-down mechanisms are effective only by gravity, i.e. no any additional energy supply is required.

For quality assurance, the drop-down time of the control rods and of the lower core section to reach their shut-down positions are measured and can be displayed on the monitor screen of the control desk (Fig. 8).

1.6.6. System Messages

1.6.6.1. Reactor Signalisation System

Three categories of signals are initiated according to their safety-related relevance:

- Status signals (Zustandsmeldungen) are messages, which inform the operator about the status of the system (drive ON/OFF, release of any action available / not available). At AKR-2 status signals are shown on the screens of the control desk.

- Fault signals (Störmeldungen) indicate a not specified abnormal operation in the operational or (as far as this signal is integrated in a safety alarm signal (see below)) in the safety system. At AKR-2 fault signals are shown on the monitor screens of the control desk (list of system messages, see Fig. 9) and have to be acknowledged by the operator on the screen. Fault signals will be saved in the reactor log datafile.

- Safety alarms (Gefahrenmeldungen) indicate a not specified abnormal operation in the safety system with audible and/or visual signals and must be prompted by the operator to clear the malfunction.
  Safety alarms are displayed at the conventional operation and signal panel on the control desk as follows:
  - incoming or newly excited signal (unconfirmed): fast flashing
  - still existing signal (confrmed): permanent light
  - going signal: slow flashing
  - deleted signal: dark (no light)

  Both, the safety alarms itself as well as the single alarms, from which they are generated, are additionally displayed on the screens of the control desk and filed and saved in the operational instrumentation.

1.6.6.2. Safety Alarms

1.6.6.2.1. Process-Orientated Safety Alarms

As process-orientated safety alarms all those criteria are summarised, which result in a reactor SCRAM due to exceeding specified safety thresholds or malfunctions of drives:
- "Neutron flux density > max"
  (1∪3 link of the thresholds in the wide-range channels WB1/WB2 and power-range channel LB)
- "doubling time < min"
  (1∪2 link of the thresholds in the wide-range channels WB1/2)
- "moderator temperature < min"
  (1∪2 link of the thresholds of the temperature 1/2 moderator < min)
- "reactor tank subatmospheric pressure > max"
- "any drive working without control signal"
  (contactor of a control rod OR of the lower core section OR of the neutron start-up source

closed AND accordant control signal is not available).

For any of these safety alarms reactor SCRAM (RESA) is initiated.This is displayed on the screens of the control desk as a status message. After the activation of SCRAM it is checked that all addressed units have reached their intended shut-down position (control rods and lower core section in accordant end position). If not, the collective message "Rückmeldung nicht vollzählig" (response incomplete) will be displayed.

1.6.6.2.2. Safety Alarms of I&C Units

Safety alarms of I&C units are collective messages, which combine all detected malfunctions in supervised parameters of I&C hardware and software.

- **"Schrankstörung"** (malfunction in the I&C cabinet)
  This alarm is initiated by (at least) one of the following single events:
  - "Störung Schrankstromversorgung" (malfunction in I&C cabinet power supply) - Fuse activated as a collective message for the whole cabinet. This can (depending on the fuse) result in reactor SCRAM or in non-availability of the control system or of single binary signals.
  - "Einstecküberwachung" (Plug-in supervision) - (unit(s) in the cabinet are pulled out / removed as a collective message for the whole cabinet). This can (depending on unit) also result in a SCRAM.
  - "Temperatur-/Lüfterüberwachung" (temperature and cooler supervision) - cooler failure or temperature in the TXS units too high). This results automatically in a shut-down of the corresponding computer (and furthermore to reactor SCRAM, if the computer of the reactor safety system is concerned, or to the non-availability of the control system).
  - "BASP/WDG angesprochen" ("BefehlsAusgabeSPerren/WatchDoG") - failures of the computer hardware / cycle time exceeded). This results automatically in the shut-down of the corresponding computer (and furthermore to reactor SCRAM, if the computer of the reactor safety system is concerned, or to the non-availability of the control system).

- **"Schranktür offen"** (door of the TELEPERM XS cabinet open)
  The operator is caused to close the door(s) of the cabinet, because they have to keep closed during operation due to EMC-reasons.

- **"Leittechnikstörung 1"** (malfunction in I&C system 1)
  Summary of all malfunctions as collective message, which are recognised within the computer of the reactor safety system.
  - missing antivalence in binary changeover contacts
  - errors in signal input or output in I/O-units
  - errors in sending or receiving signals via computer bus-system.

  Additionally integration of the message that the computer of the operational intrumentation is not on cyclical operation.

- **"Leittechnikstörung 2"** (malfunction in I&C system 2)
  Summary of all malfunctions as collective message, which are recognised within the computer of the operational intrumentation.

- missing antivalence on binary changeover contacts
- errors in signal input or output in I/O-units
- errors in sending or receiving signals via computer serial bus system.

Additionally integration of the message that the computer of the reactor safety system is not on cyclical operation.

The detailed localisation of malfunctions for the last mentioned two messages is preferably possible by means of the service computer. The messages which originate in the cabinet additionally activate a lamp at the top outside of the cabinet.

1.6.6.3. Fault Signals

Fault signals give detailed information on the current status of the I&C system. Fault signals are displayed on the monitor screens of the reactor control desk and visualised, logged and saved in the computer-aided operational and supervising system.

1.6.6.4. Releases / Status Messages

The information on the reactor control desk offers a wide overview about the general status of the I&C system, partly with conventional equipment technology (operation and signal panel) and partly with screens (3 monitors).

The operation and signal panel (Fig. 5) with conventional equipment technology includes:
- safety alarms class 1,
- key-switches for release and simulation "Kernhälfte in oberer Endlage" (lower core section in upper end position),
- pushbutton to confirm and delete safety alarms and to check lamps,
- analogous panel meters "Neutronenflussdichte Weitbereich" (neutron flux density wide range), "Verdopplungszeit" (doubling time),
- digital display of "Temperatur Moderator" (moderator temperature) and "Unterdruck Reaktor" (subatmospheric pressure in reactor tank).

Additional control elements in conventional techniques on the control panel:
- covered latching button for manual reactor SCRAM ("Hand-RESA"),
- key-switch for RESA-Reset,
- key-switches for releases to change operational modes of the computers.

The reactor facility is controlled and supervised by means of 3 monitor screens on the control desk, on which messages and facility pictures are displayed (Figs. 6 to 9). These pictures provide all information for reactor control and displaying of process parameters.

1.6.7. Service Equipment (Diagnostics Interface)

The service equipment is used for planning, configuration, monitoring, maintenance and modification of the reactor safety system such as:

- diagnostics,
- modification of I&C components,
- modification of parameters,
- documentation and saving of hardware and software specifications.

It can be distinguished between services without required communication with TELEPERM XS and those with access to the safety units. The following properties characterise the service equipment:
- It is not part of the safety I&C; therefore, it can be switched off.
- It is never used for reactor operation process management.
- A hierarchically arranged operating surface is provided for user prompts.
- Unauthorized access protection to the service equipment is given by administrative measures (login/logout function) and additionally by key-operated switches.
- The software is protected against modifications by setting read-only features and additionally by backup procedures on data storage (CD-ROM).
- The service computer is located in the reactor control desk behind a locked door.

1.6.8. Gateway Interface Computer

The gateway computer sends information from the reactor safety and protection system to the control and monitoring system (and never in the opposite direction!). The gateway computer is located in the reactor control desk behind a locked door.

1.6.9. Interconnection between I&C Units

Communications between the reactor protection and the operational computers in the TELEPERM XS cabinet to the gateway and service computers in the reactor operator control desk takes place via a fiber optic cable. In a media converter, the signals are converted into electric signals and transmitted to the gateway and service computers via an Ethernet switch.
Normally, all three monitor screens of the reactor control desk are connected to the gateway computer. Optionally, the monitor on the right hand side of the control desk can swapped by a switchbox to the service computer (Fig. 12).

2. Instruction for Reactor Start-up Procedure

2.1. General Remarks

1. In the operation of the AKR-2 three modes of reactor start-up procedures are to distinguish:
   - Wiederholungsstart (normal restart),
   - Anfahrexperiment (comprehensive start-up experiment),
   - kritisches Experiment (critical experiment).

   The normal restart is the start-up procedure in routine reactor operation. Restarts are possible, if nothing has changed since last reactor operation with effect on the reactivity properties of the facility, neither in the reactor itself nor in the internal experimental arrangements.The reactivity characteristics and the control rod positions in the critical state of the reactor are known.

   A comprehensive start-up experiment is necessary for a safe reactor start-up after small material or geometry changes (e.g. new installation of internal experimental arrangements), which do not require a critical experiment. The behavior and the reactivity properties of the reactor are known, i.e. the fuel loading, the reflector properties and the reactivity characteristics of the control rods.

   A critical experiment has to be carried out, if the physical parameters (e.g. critical mass, reactivity characteristics of the control rods, excess reactivity etc.) are only known from calculations. That applies in any case to the first start-up of a newly build reactor and to any further start-up after all such variations of the assembly when considerable change in the reactivity behavior is expected.

2. For any reactor start-up a valid written service instruction (BA) signed by the director of the AKR (Leiter AKR) and the radiation protection commissioner (SSB-AKR) or the corresponding deputies is required.

3. An experiment can be carried out with the same service instruction (BA) several times under the same conditions if it fits the period of validity of the BA.

4. Before each reactor start-up, a check of the proper function of the units is mandatory.

5. The preparation, the functional check of the equipment and the progress of the starting-up procedure have to be recorded in the operation logbook of the AKR-2. For that purpose a given flowchart of the checking and start-up process has to be used (Fig. 13).

6. In the comprehensive start-up experiment and in the critical experiment, the loading of the experimental channels in the core and in the reflector area as well as the measured pulse rates have to be recorded in the operation logbook.

7. When the reactor is in the supercritical state, the values of the doubling time, the corresponding control rod positions and after being in the critical state the critical control rod positions have to be recorded in the operation logbook according to the given flowchart.

8. At all changes in and at the reactor, especially
   - adding of extra fuel,
   - changes in the experimental channels,
   - movement (upwards) of the lower core section,
   - withdrawal of the control rods,
   - movement of the start-up neutron source

   the displays (especially doubling time and reactor power) have to be watched carefully.

   In case of fast power increase (in connection with constant doubling times below 30 s) the reactor has to be down-regulated. At doubling times below 10 s the reactor has to be shut-down instantly (unless automatically happened).

9. If in the critical experiment and the comprehensive start-up experiment only low pulse rates (i.e. < 5 cps) are measured, then for each pulse channel all pulse rates have to be determined as the average from at least 3 single measurements. A valid pulse rate, which shall be measured only once, must have at least 15 cps.

10. The correct sequence of operational steps in the start-up procedure is assured by the protective logic (Fig. 11). By means of soft-buttons on the screens of the control desk first step is to move the start-up neutron source to its upper end position, afterwards to do the same with lower core section. The successful execution of these steps is required to let the reactor safety system give release for withdrawing the control rods. This is possible not earlier until the start-up neutron source AND the lower core section have reached their respective end positions AND a pre-defined minimum count rate is measured at the neutron flux channels AND the key-lock "Anlassfreigabe" (start-up release) on the control desk is activated. Thus, operation errors, which affect the safety of the reactor, are impossible.

11. Too fast power increase (i.e. too short doubling time) or too high absolute power in the start-up procedure result in an automatic shut-down of the reactor by the safety system.

12. The pictures on the monitor screens of the control desk show a symbolic illustration of each drive and each control command, which can be addressed by mouse click on soft-buttons. At the same time (according to the context of the picture), releases, current positions and end positions of the drives as well as measured doubling time or pulse rates are displayed in order to see the success and the consequences of the action.

13. All operations have to be done on the control desk manually. Operational elements are conventional keys and pushbuttons as well as the computer-aided operational and supervising system which requires actions by keyboard or mouse on soft-buttons on the screens. Only the (operative classified) power control and the (safety related) fast reactor shut-down (SCRAM) as well as the optical and acoustical alarm signals in cases of failures work automatically.

14. On one of the control desk screens, the list of system messages can be displayed as a message sequence (example see Fig. 9). All fault and alarm signals are listed in this table in the order of their appearance (and disappearance).Coming as well as going messages are registered. Coming messages are announced also acoustically, they have to be confirmed by the reactor operator.

Additionally, alarm signals are displayed on the conventional alarm and signal panel of the control desk and are acoustically announced. Confirmation also of these signals by the reactor operator is mandatory. Normally, all alarm signals result in a SCRAM initiated by the reactor safety system.

15. Since the delay of the subatmospheric pressure control in the ventilation system in the reactor hall can be up to 20 min, the reactor operation may start not earlier than having passed this delay time after switching on the ventilation, and if, additionally, LED H13 on the signal panel does not indicate any failure.

2.2. Start- and Stop-Procedure of the Safety and Control System

For starting the safety and control (SUS) system, there is a combined mains switch on the control desk to switch on the power supply for the TELEPERM XS cabinet and the control desk of the reactor. The gateway computer (located in the control desk of the reactor), which connects to the safety and control system, and in which the operation and supervising system is integrated, has to be started afterwards manually. The computers in the TELEPERM XS cabinet boot automatically when connected to the power supply. After booting, the reactor protection is in a state, in which the signal "Reaktor abschalten (RESA)" (= SCRAM) is activated and the electro-magnets for the control rods and the lower core section are disconnected. The reactor is shut down.

The software of the computer-aided operational and supervising system of the AKR-2 is programmed in the programming language WinCC.

In detail the safety and control system is put into operation as following:

1. Switch-on the main switch on the right-hand side of the control desk (no. 10 in Fig. 4). Hereby, the power supply for both the I&C cabinet (24V DC) and the computers in the control desk (230V AC) is switched on.
2. The TELEPERM XS system in the cabinet starts running automatically.
3. The computers (gateway and service device) have to be switched on manually. The gateway computer runs automatically until WinCC software is executed. The plant video display is shown on all three monitor screens.
   While WinCC is running the access to the operating system level (Taskmanager, Windows-Explorer etc.) and to quit WinCC by unauthorised action is blocked. This is only possible after entering a pre-defined username and a password and is only permitted to the staff. When clicking a click button the message “Keine Berechtigung” (no authorisation) is displayed. Pressing <STRG+Tab+n> an entry mask appears, in which the login and the password is to enter (only permitted to the staff). With the correct entry the click buttons are unblocked for two minutes.
4. Push the <STRG>-key of the right-hand keyboard on the control desk twice quickly in order to switch the right monitor screen to the service device.
5. Login at the service device (any user but root).
6. Start the Service-Monitor-Server (SMS) by double click on the button “Start SMS”. This is required, because TXS, while booting, checks the communication to the gateway and to the service device. The connection is accepted to be correct not earlier than confirmation signals from the computers (via the SMS) arrive the TXS processing units. Without successful confirmation the RESA(=SCRAM)-signal cannot be reset.

7. Afterwards the SMS could be stopped and the service device could be shut down. But reasonably the SMS should be left running in order to have available possible failure messages (e.g. telegram failure), which are recorded in the log-datafile of the SMS, for a long term analysis.
8. Press the <STRG>-key on the right-hand keyboard twice quickly to switch the right monitor screen back to the gateway.

The safety and control system is switched off like following:
1. Press the <STRG>-key on the right-hand keyboard twice quickly to switch the right monitor screen to the service device.
2. Quit the SMS (i.e.enter <q> into the window of the SMS + <return>)
3. Quit WinCC:
   By pressing <STRG+Tab+n> an entry mask appears, in which the username and the password has to be entered (only permitted to the staff). With the correct entries the click buttons are unblocked for two minutes.
4. Shut down the gateway interface computer and the service device in the reactor control desk (any order).
5. Switch off the mains switch on the right side of the reactor control desk.

2.3. Preparation for Reactor Start-up

For starting the reactor, the signal “Reaktor abschalten (RESA)” (fast reactor shut-down (SCRAM)), which is saved in the safety system, has to be reset. This is done with a key-switch on the conventional operation and signal panel of reactor control desk, and only successful in that case if no shut-down criteria, no alarm signals or operational related failures are queued. After successful reset of the RESA (SCRAM) signal, the holding magnets for the control rods and the lower core section are supplied with power and are coupled.

The pre-defined operation actions required for starting the reactor are effected by soft-buttons on the monitor screens (Figs. 6 and 7) on the reactor control desk. The required and possible sequence of the reactor operator’s actions is pre-defined technically by the protective logic (Fig. 11) and administratively by a check list for reactor start-up and operation procedure (Fig. 13) in the operation logbook and is supervised by the reactor safety system. Operation systems are blocked until the accordant release conditions are existent.

The preparation for the reactor start includes the check of the radiation protection equipment:
- proper function of the hand-foot-clothes-contamination-monitor (HFK),
- personal equipment with film batches and/or dosemeters for direct reading,
- availability and proper function of the gauged portable, battery-powered X-ray and gamma dosemeter,
- availability and proper function of the portable, battery-powered neutron dose rate meter,
- operational reliability of the stationary gamma local dose rate devices in the reactor hall (green panels lighting).

2.4. Safety Check of the Fast Shut-down System

- The reactor facility is started according 2.2. and prepared for start-up according 2.3.
- The neutron source is lifted to its upper end position at lower core section. The position of the source is displayed on the monitor screen, additionally the arrival at the end position is signaled.
- The lower core section is lifted out of its lower end position. The position of the lower core section is displayed on the monitor screen. The movement has to be stopped, when the lower core section has left its lower end position, which is signaled on the monitor screen.
- On the conventional operation and signal panel of the reactor control desk, RESA (=SCRAM) signal is initiated manually by pressing the red pushbutton "Hand-RESA".
- On the monitor screen, it has to be checked that the lower core section has dropped down reliably (confirmation by showing the light signal for reaching the lower end position).

For the critical experiment, another extended safety check is necessary:
- The proper shut-down function of the lower core section is checked by the described safety check according 2.4.
- Additionally to the previous procedure, also the correct drop of the control rods has to be checked. For this purpose, the protective logic signal "Kernhälften zusammen" (core sections joined) has to be simulated by turning the key-switch "Simulation KHZ"" (simulation of core sections in joined state) on the conventional operation and signal panel of the control desk. This key-switch administratively ensures that switching over to this operating state can only be done consciously.
- All three control rods are moved out from their internal end position one after another. The movement has to be stopped, when the corresponding control rod has left its end position, which is signaled on the monitor screen.
- On the conventional operation and signal panel of the reactor control desk, RESA (SCRAM) signal is initiated manually by pressing the red pushbutton "Hand-RESA".
- On the monitor screen, it has to be checked that all control rods have dropped down reliably (confirmation by showing light signals for reaching their end positions).

After this safety check, the facility must not switched off until finishing the critical experiment.

2.5. Normal Restart of the Reactor

1. Before beginning any restart, the reactor facility has to be prepared according chapters 2.2. and 2.3., and the safety check has to be done according to chapter 2.4. The results of the safety check have to be recorded in the operation logbook according the flowchart stamp print (Fig. 13). If all required conditions are met, the reactor can be started.
2. The neutron source is lifted to its upper end position at lower core section. The position of the source is displayed on the monitor screen, additionally the arrival at the end position is signaled (end position about 800).
3. After the source is completely in its upper end position and the required neutron flux is measured, the lower core section is lifted. The position of the lower core section is shown on the monitor screen. While lifting the lower core section, the displays of the doubling time and the power meter channels have to be watched carefully. The arrival at end position is signaled (end position about 506).
4. With the key-switch of the starting release the drives of the control rods are unblocked. The rod drives are connected in such a way that only one single rod can be selected and moved

in pull-out direction. The sequence of pulling out is arbitrary (if not indicated different administratively in the service instruction), because each rod approximately has the same reactivity equivalent.

5. By pulling out the control rods the reactivity change has to be limited. If the control rod characteristic is not known, a rod must not be lifted more than 5 cm (about 500 digits) at a time. Between each step a break for at least 1 min must be maintained. The power and doubling time meters have to be watched carefully at any time.
6. In case of a constant doubling time combined with a permanent power increase the reactor is supercritical. Now, the rod position may only be changed in such a way that the doubling time does not fall shorter than 30 s (acoustic alert at 20 s, automatic RESA (SCRAM) at 10 s).
7. The start neutron source can be run out manually at a power exceeding 0.25 W. A small influence on the reactor behavior has to be expected. The possibly resulting reactivity change must be compensated by adequate movement of the control rods, so that the reactor power still increases with a doubling time >30 s.
8. When the intended reactor power given in the service instruction is approaching, the doubling time is to control to "infinite" (i.e. display of the doubling time at 1000 s) to adjust the critical state at the required power.
9. The restart is finished by writing the values of the control rod positions, the reactor power and the time of reaching the critical state into the flowchart stamp in the operation logbook (Fig. 13).

2.6. Protocol Records in the Operation Logbook

Detailed records of all operations at nuclear facilities are mandatory. These records are important documents. All entries into the operation logbook have to be done well legible and with ink or ball-point pen. The sequence of all operations must be reproducible by means of the logbook. Therefore, all actions at the facility have to be instantly registered with date and time (according the clock at the reactor facility) and signature.

This is relieved by the pre-defined algorithm of a flowchart control stamp in the operation logbook (Fig. 13). Entries, which are not pre-defined in the stamp, are written down formlessly and manually in chronological order below the stamp in the operation logbook with date and signature of the reactor operator.

For each experiment at the reactor, a corresponding service instruction (BA) with the respective date is required, from which the number of the instruction, the name of the experiment and the name of the responsible person for the experiment have to be copied into the flowchart stamp in the operation logbook.

Before any operation of the AKR-2, the dosemeter equipment of the staff and all other participating persons as well as the proper function of the instruments and the systems necessary for fast reactor shut-down (SCRAM) have to be checked. The result is to be noted in the stamp (time of check and in case of the positive result "i.O." (= o.k.)) and signed by the reactor operator.

Also the core temperature displayed on the conventional operation and signal panel (Fig. 5) has to be checked and noted in the logbook. All steps of reactor start-up procedure are recorded during or instantly after their execution.

Usually, the end of any reactor experiment is the shut-down of the facility. The time of this action has also to be recorded in the logbook. The log is finished with the signature of the operator.

3. Fields of Application of the AKR-2

3.1. Education and Training

The main purpose of AKR-2 and its design basis was and is the education of students in nuclear and reactor physics, in nuclear engineering as well as to teach fundamental knowledge and rules in radiation protection and radiation dosimetry (see chapter 4).

3.2. Application in Research Projects

Due to its physical characteristics, a zero power reactor as the AKR-2 offers only limited possibilities for research. However, it can be used for all those projects where high neutron fluxes are not required but variable operational conditions and low costs are requested, e.g. investigations on sophisticated neutron detectors, development of radiation measuring techniques, radiation spectrometry in mixed neutron-photon fields.

3.3. Information Centre for the Public

One of the great advantages of small, low power training reactors on the campus of a university is their central location and that not such strict admittance restrictions exist as in many other nuclear installations. Hence, it is suitable for using these facilities also as information centres for groups or individuals of the interested public. Besides the transfer of fundamental knowledge, it is of high importance to enhance the quality of the public discussion about nuclear energy.

4. Examples of Typical Fundamental Training Experiments at AKR-2

A variety of exercises in fundamental neutron, reactor, nuclear and radiation physics as well as in radiation protection were developed at AKR. For any exercise, scripts are available and can mostly be downloaded from the AKR-website describing the theoretical basis of the experiment and providing practical procedures, values to be determined, evaluation procedures as well as how to discuss results etc. In advance of any practical work, the preparation of the students to the special exercise is tested by means of a PC-based check list. Results of these tests are used by the lecturer to identify deficiencies in knowledge or in preparation and to correspondingly concentrate efforts in colloquia.

4.1. Reactor Start-up Procedure

This is the basic exercise for all programs at the reactor and in most cases the first contact of the students with the facility. The exercise comprises start-up of the reactor, the adjustment of the critical state at various power levels, changes of the power level (increasing, decreasing), various inspections and safety checks of the instrumentation and control units.

Included in that part is the study of fundamental reactor physics effects as:
- correlation between reactivity, reactor period (or doubling time) and reactor power,
- subcritical multiplication of the neutron start-up source,
- prompt reactivity steps,
- stable positive reactor period with exponential power increase in supercritical reactor state,
- correlation between prompt and delayed neutrons,
- control of stable reactor power,
- independence of control rod position in critical reactor state on absolute power level (if effects of temperature and fuel burnup are negligible as in a zero power reactors),
- influence of the neutron start-up source on the critical reactor,
- dependence of neutron and gamma dose rate on the reactor power and on the distance from the reactor.

Usually, this part is also the first close contact of the students with the special rules of radiation protection measures (changing clothes, wearing overshoes and dosemeters, rules of conduct according to the ALARA-principle, check of contamination when leaving the radiation protection area etc.).

4.2. Control Rod Calibration

For safe operation of a nuclear reactor it is of high importance to know:
- reactivity values of all control rods in dependence on their position,
- excess reactivity,
- shut-down reactivity brought about by the control rods.

In the exercise position-dependent reactivity values of all control rods are determined by measurements of the stable reactor period in combination with the reactivity compensation method. Physical background of the evaluation method is the INHOUR-equation.

Results of the exercise are the integral reactivity curves (total reactivity) as well as the differential reactivity curves (characterisation of the efficiency of a control rod in a given position). Furthermore, the excess reactivity is calculated as the maximum available reactivity to be released by withdrawal of all control rods, starting from the critical core. In contrary, the shut-down reactivity of the control rods is calculated as the negative reactivity value when all control rods are in their shut-down positions.

4.3. Critical Experiment

The critical experiment is a check of proper fuel loading (critical mass) and core configuration of a nuclear reactor. At AKR-2 with its well sealed core no exchange of fuel elements is feasible in student's exercise. Instead, a stepwise variation of the distance of both core sections is carried out leading to a corresponding variation of the subcritical multiplication of the neutron source. For any step, the multiplication factor k of the reactor and the corresponding reactivity ρ of the system are iteratively calculated. The result of this kind of critical experiment is the critical distance between both core sections.

4.4. Adjoint Flux Function

The intention of the exercise is to investigate influences on reactivity by introducing samples of different materials (e.g. PVC as a neutron absorber, PE or graphite as typical neutron scattering materials) into the central irradiation channel of the reactor. The samples are shifted stepwise inside the channel and the reactor response is measured. The reactivity $\rho(x)$ is evaluated in dependence on the distance x of the sample from the core centre and compared to calculations.

4.5. Measurement of Radial Neutron Flux Distribution

The knowledge of space dependent radial and axial neutron flux distribution $\Phi(r)$ and $\Phi(z)$, respectively, is of high importance in any reactor facility, e.g. in research reactors as entry value for many experiments (neutron source strength in irradiations) or in power reactors as heat source distribution. In the experiment, the flux distribution of thermal neutrons $\Phi_{th}(r)$ is measured in radial direction over the diameter of the reactor in the horizontal plane of the central experimental channel by means of neutron activation technology in Mn activation samples.

4.6. Pile-Oscillation Experiment

While the exercise "control rod calibration" and "adjoint flux function" are based on the measurement of the stable reactor period (i.e. in the equilibrium state between prompt and delayed neutrons) pile oscillation experiments require a more detailed way of looking at the theoretical basics of reactor kinetics including the solution of point kinetic equations for time dependent reactivities $\rho(t)$. By means of a mechanical pile-oscillator fitted to the central experimental channel of the reactor the function x(t) as sample position in dependence on time is measured using a multichannel analyser which is operated in multi-scaling mode. This function is weighted with the result of the adjoint flux function $\rho(x)$ in order to get $\rho(t)$. The experimental result will be compared with an independent calculation.

4.7. Demonstration of Neutron Activation and Decay of Various Radioisotopes

The production of radionuclides by neutron capture is one important application of many research reactors. Of course, in the low neutron flux density of a zero power training reactor no real production of nuclides is possible in industrial scale, but the effect of the production of radioactive nuclei itself can be demonstrated as well. Activation, saturation and subsequent decay can be investigated in dependence on reactor power (i.e. neutron flux density), time, capture cross section, half live time, sample size etc. Students get the experience that materials are activated in the neutron field but they acknowledge in the same way that activity decreases subsequent to the irradiation according to the law of radioactive decay. Experimental results are compared to calculations on relative and absolute scale.

4.8. Identification of Radionuclides by means of High Resolution Gamma Spectroscopy

High resolution gamma spectrometry is one of the mostly used techniques for non-ambiguous identification of radionuclides. Physical properties of different kinds of detectors (e.g. simple traditional NaJ-scintillator/photomultiplier-combinations with their high efficiency but comparably poor energy resolution and actual HPGe semiconductor detectors) and required features of amplifiers and multichannel analysers are investigated. The HPGe-spectrometer is calibrated in its energy scale as well as with respect to absolute energy dependent efficiency using a set of well known calibration sources. The procedure of neutron activation analysis (NAA) is demonstrated by activation of "unknown" samples in an irradiation channel of the reactor and subsequent identification of the isotopic composition qualitatively and quantitatively. Traces of radionuclides can be measured and identified in samples taken from natural environment (e.g. K-40) or industrial goods (e.g. ancient tiles or watches). Another issue of NAA is the use of activation foils for the determination of absolute neutron flux densities in different energy ranges at nuclear facilities according to the conversion of the formula for activation with respect to the neutron flux as unknown value.

4.9. Radiation Protection and Shielding

In this experiment, the students obtain knowledge of dosimetric units and of dosimetric limits defined in the German Radiation Protection Ordinance. Characteristics of ionising radiation are highlighted and from these characteristics essential principles of radiation protection are derived and deepened. The student gets experience in the correct use of suitable kind of measuring devices for the different types of radiation and measuring conditions as well as in the evaluation of measured values. Fundamental rules of radiation protection according to the ALARA-principle are demonstrated experimentally such as: keep distance, use shieldings, shorten time in radiation fields.
The fundamental physical dependences of dose rates or of total doses on these parameters are explored. Efficiencies of several shielding materials as Pb, Fe or different qualities of concrete are investigated and shielding coefficients are derived and compared with data in literature. Using these data, required thicknesses of shieldings can be calculated and experimentally verified in order to get defined dose rates meeting given conditions at the reactor.

4.10. Radiation Measurement Techniques

Fundamental characteristics of different kinds of radiation detectors are explored with special attention to gas-filled ionisation detectors like ionisation chambers, proportional and Geiger-Mueller-counters. The dependence of the counting rate on high voltage is measured and discussed with reference to the adjustment of suitable parameters for proper application. Students become familiar with the statistical character of any measured value being essential for assessment of accuracy (Gaussian-distribution of measured data, arithmetic mean value, statistical uncertainties, standard deviation). Attenuation curves are measured for different materials and linear as well as mass attenuation coefficients are determined and discussed.

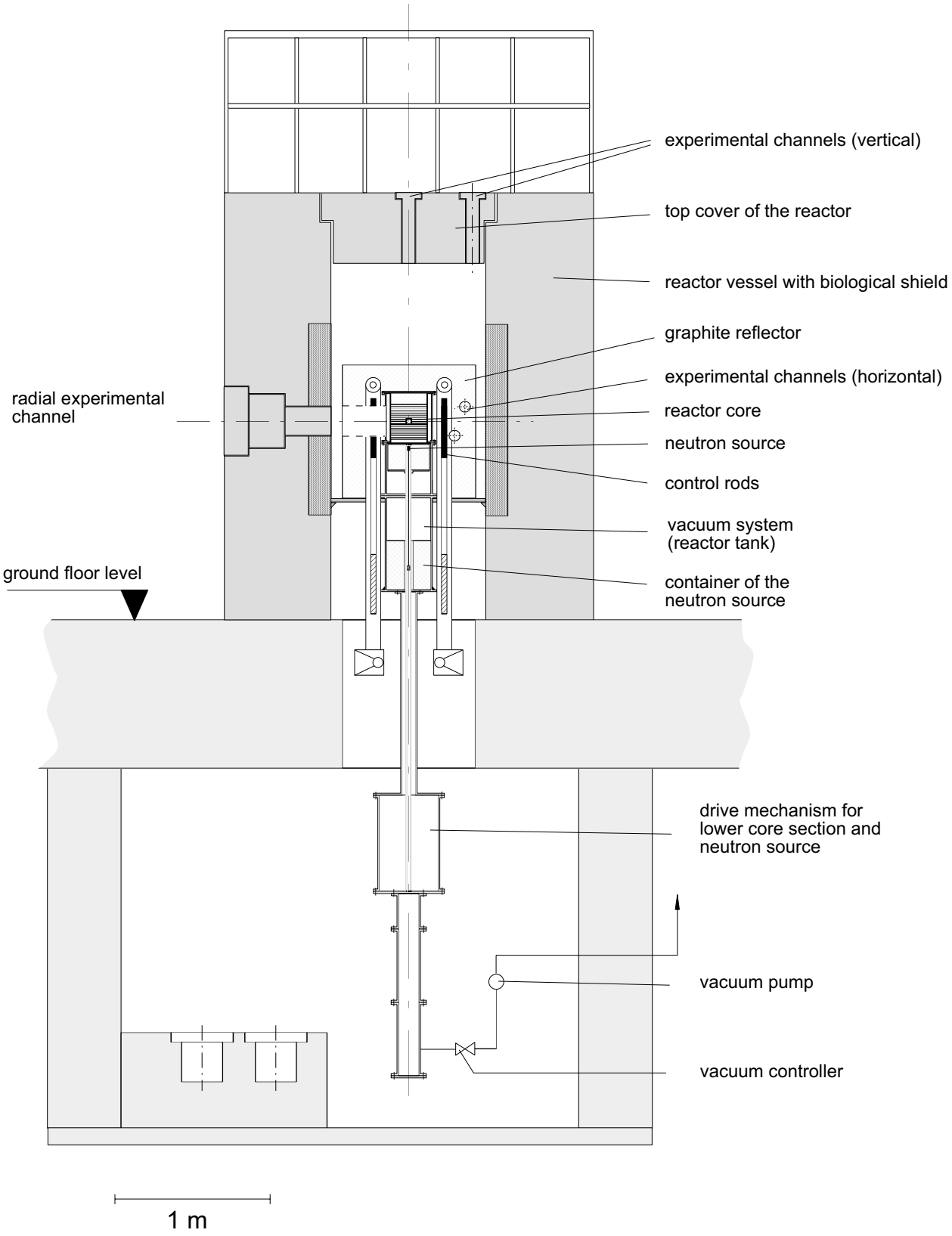


Fig. 1, Vertical Cross Section of the Training and Research Reactor AKR-2

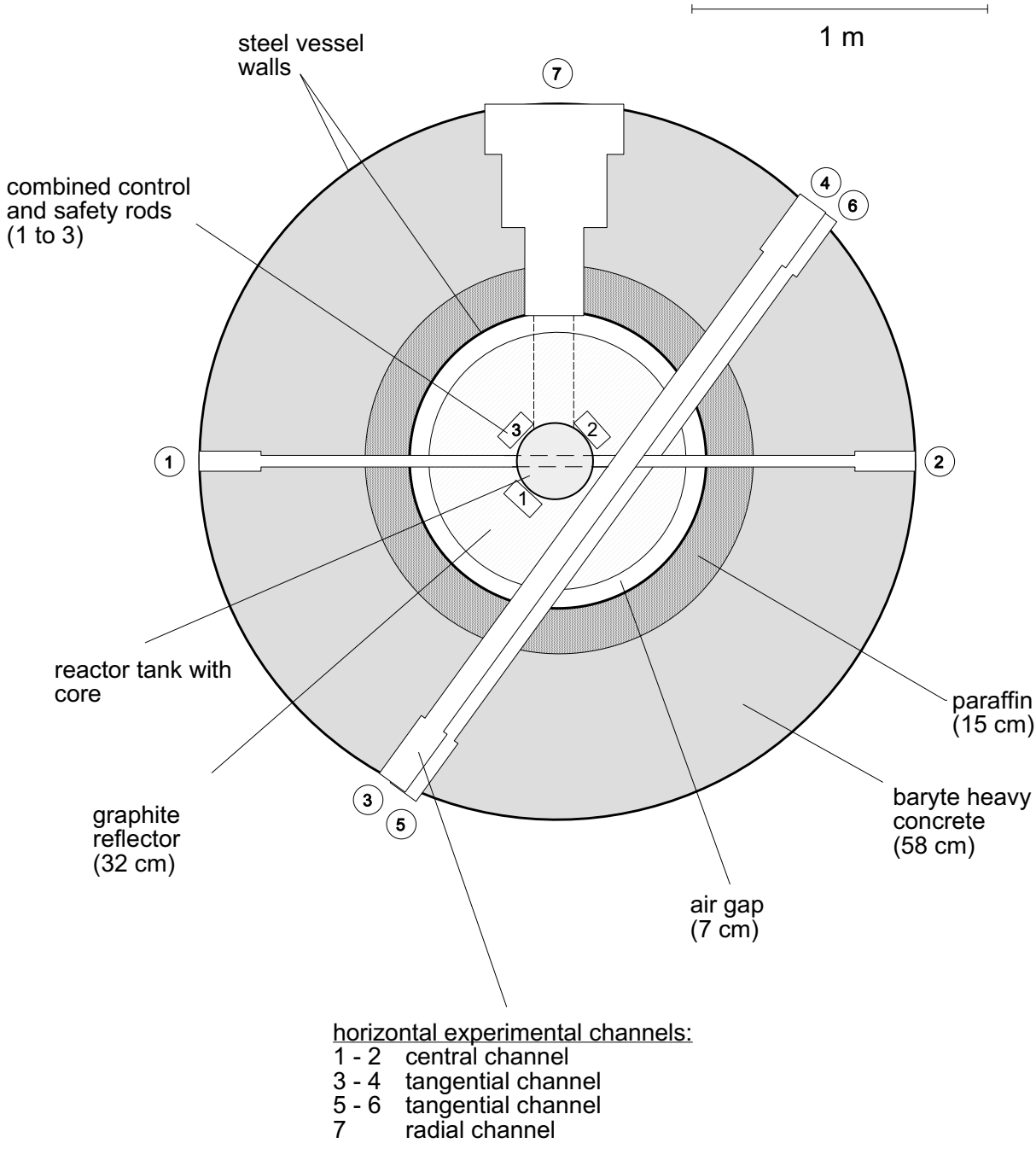


Fig. 2, Horizontal Cross Section of the Training and Research Reactor AKR-2 at Core Level

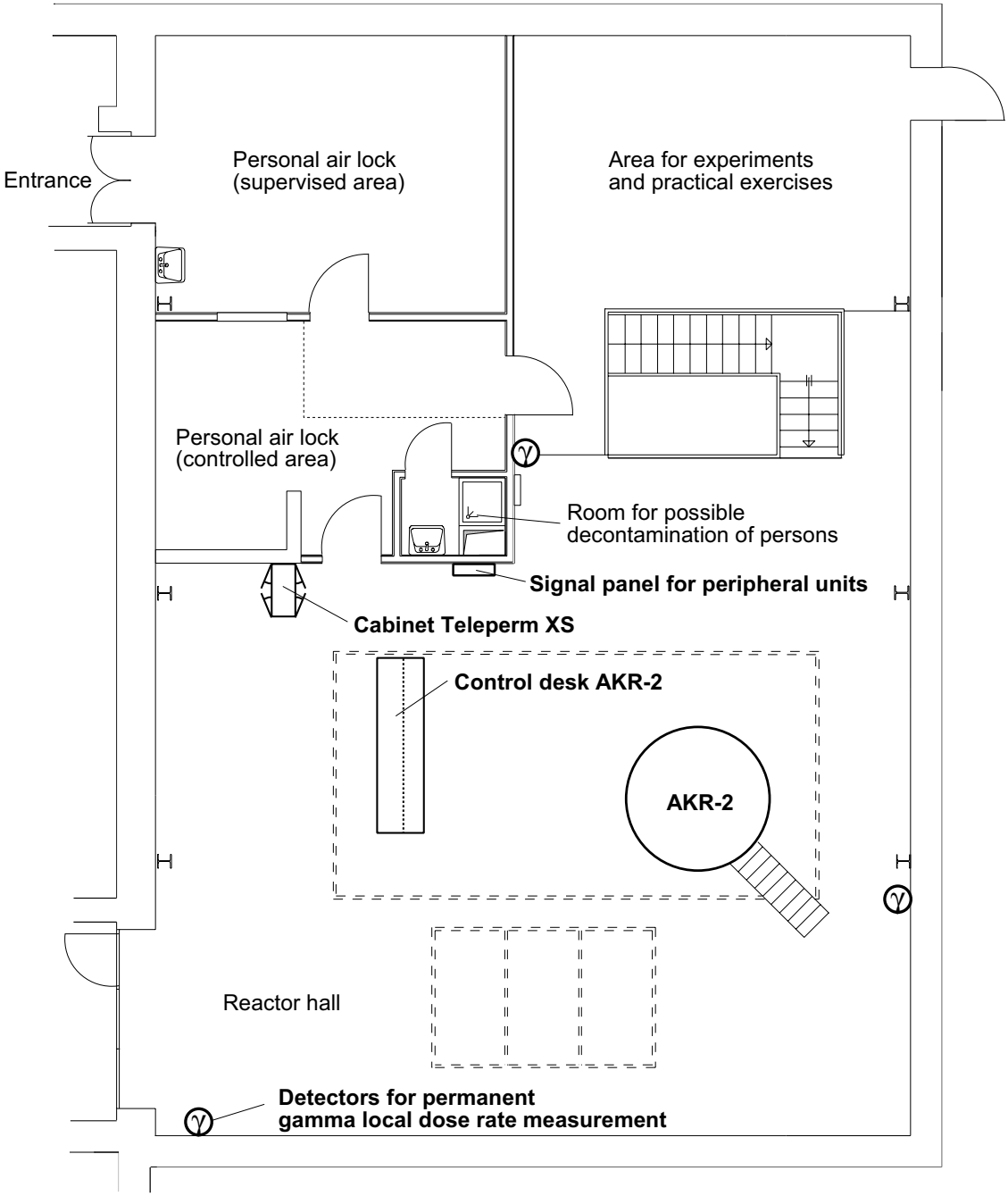
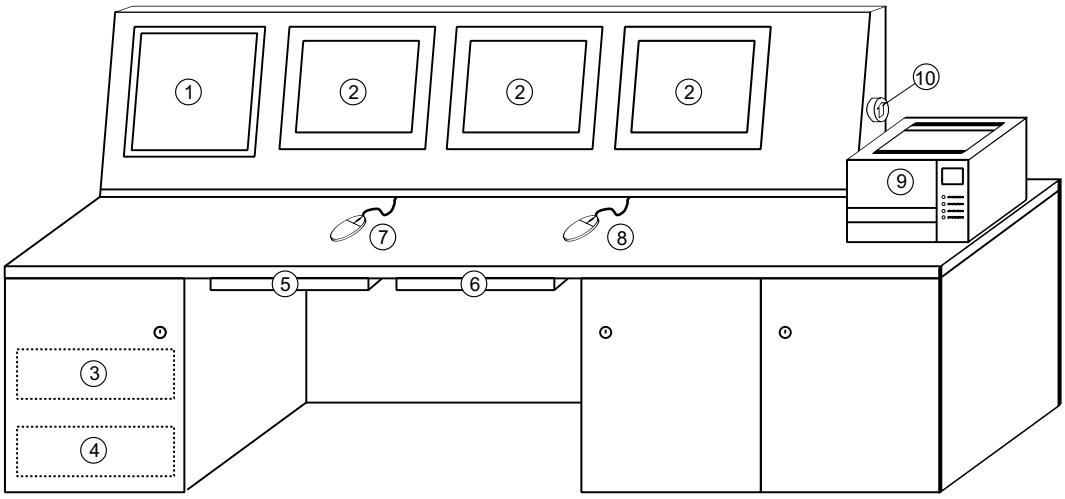


Fig. 3, Cross Section of the Reactor Hall AKR-2



1 Conventional Operation and Signal Panel

2 Monitor screens

3 Service computer (Diagnostics Interface)

4 Gateway Interface Computer

5 Keyboard for Gateway Interface Computer

6 Keyboard for Service Computer

7 PC-Mouse for AKR-2 Control

8 PC-Mouse for Service Computer

9 Hardcopy Printer

10 Mains Switch (for Electrical Power Supply of Reactor)

Fig. 4, Control Desk of AKR-2

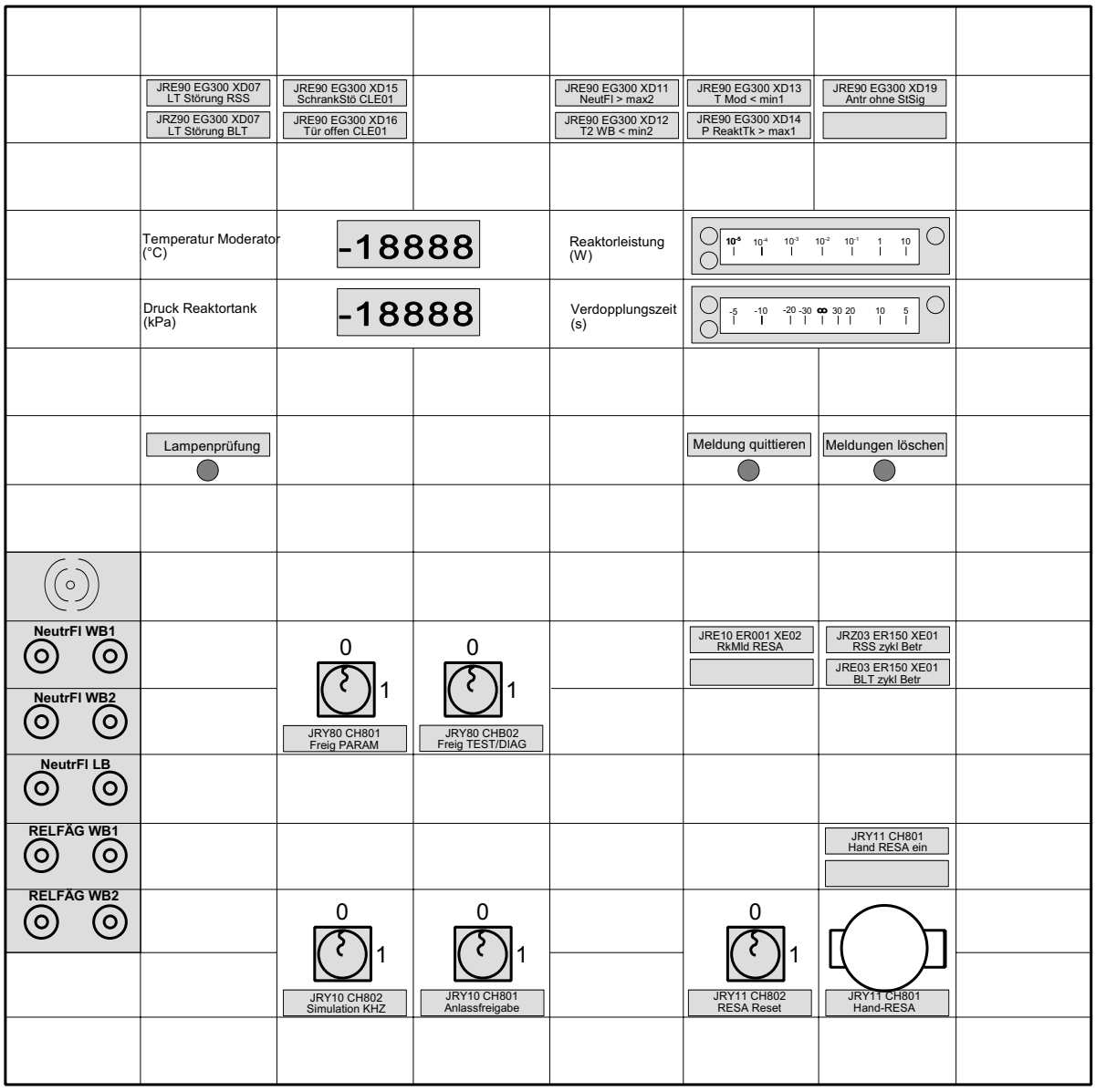


Fig. 5, Conventional Operation and Signal Panel

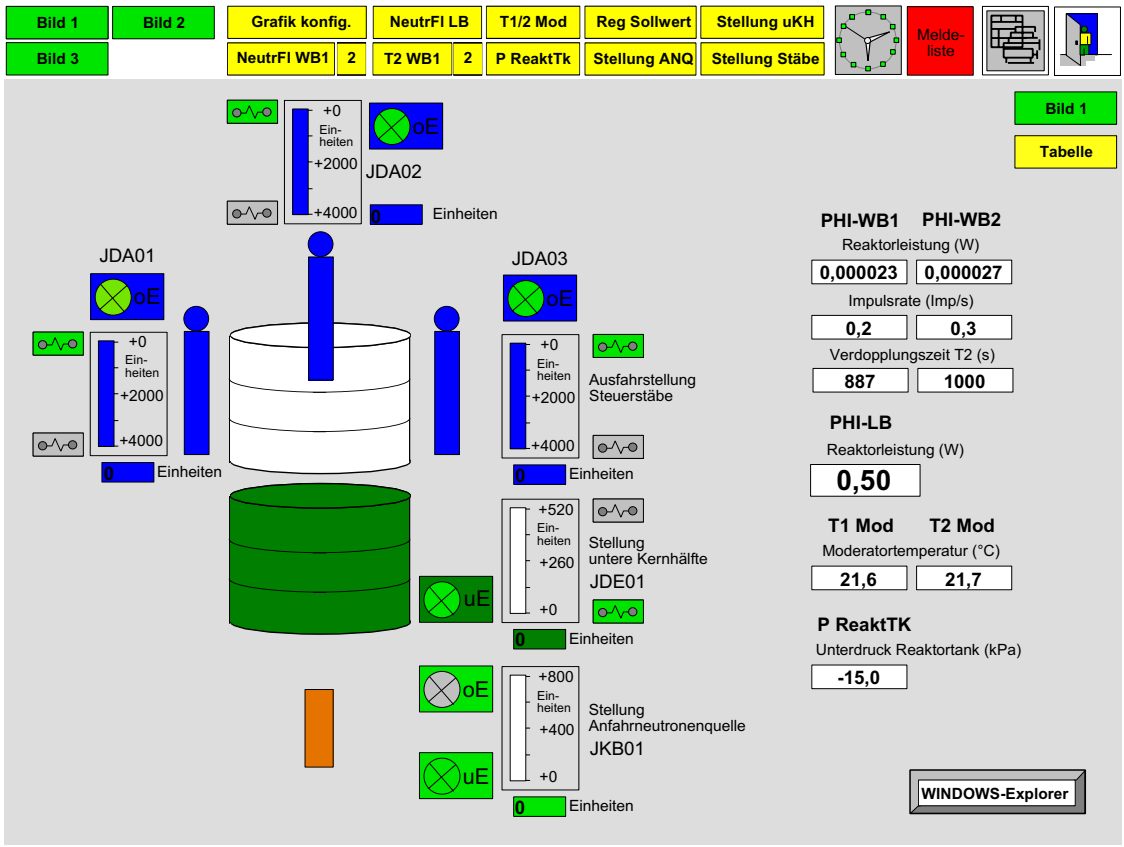


Fig. 6, Monitor Screen, Picture 1

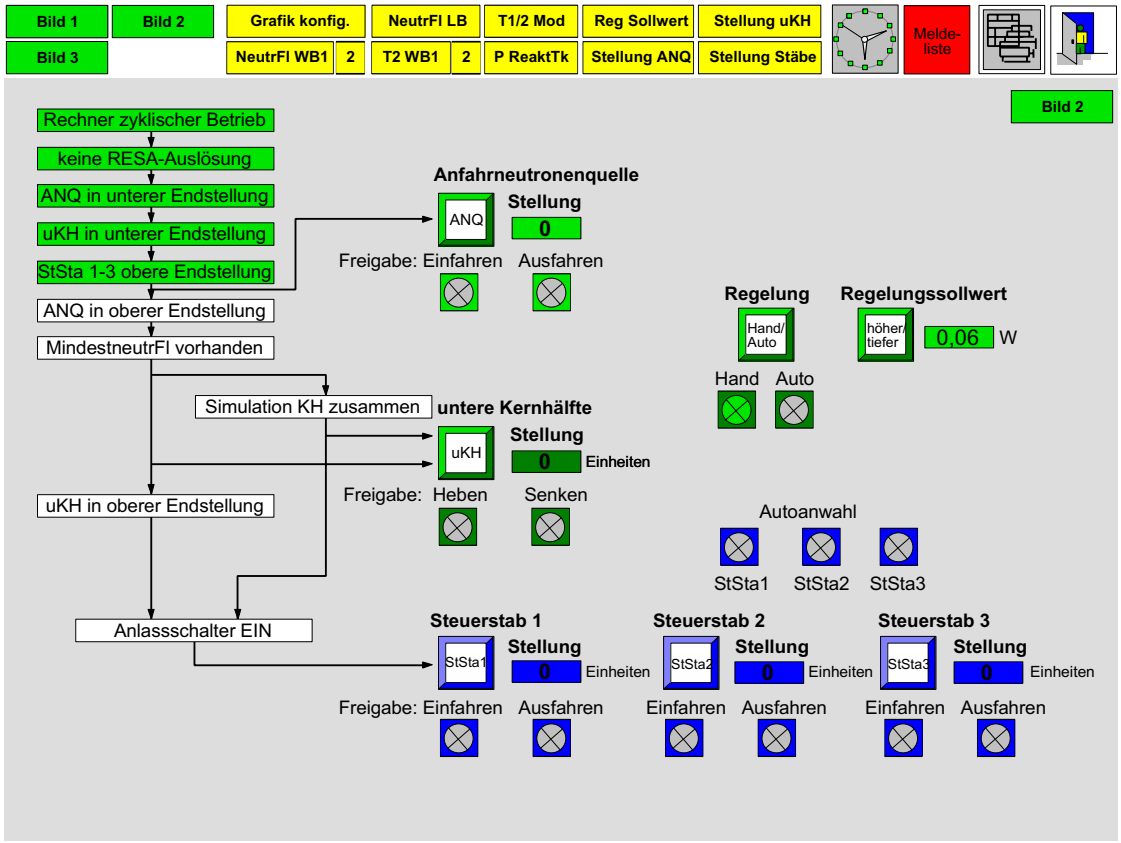


Fig. 7, Monitor Screen, Picture 2

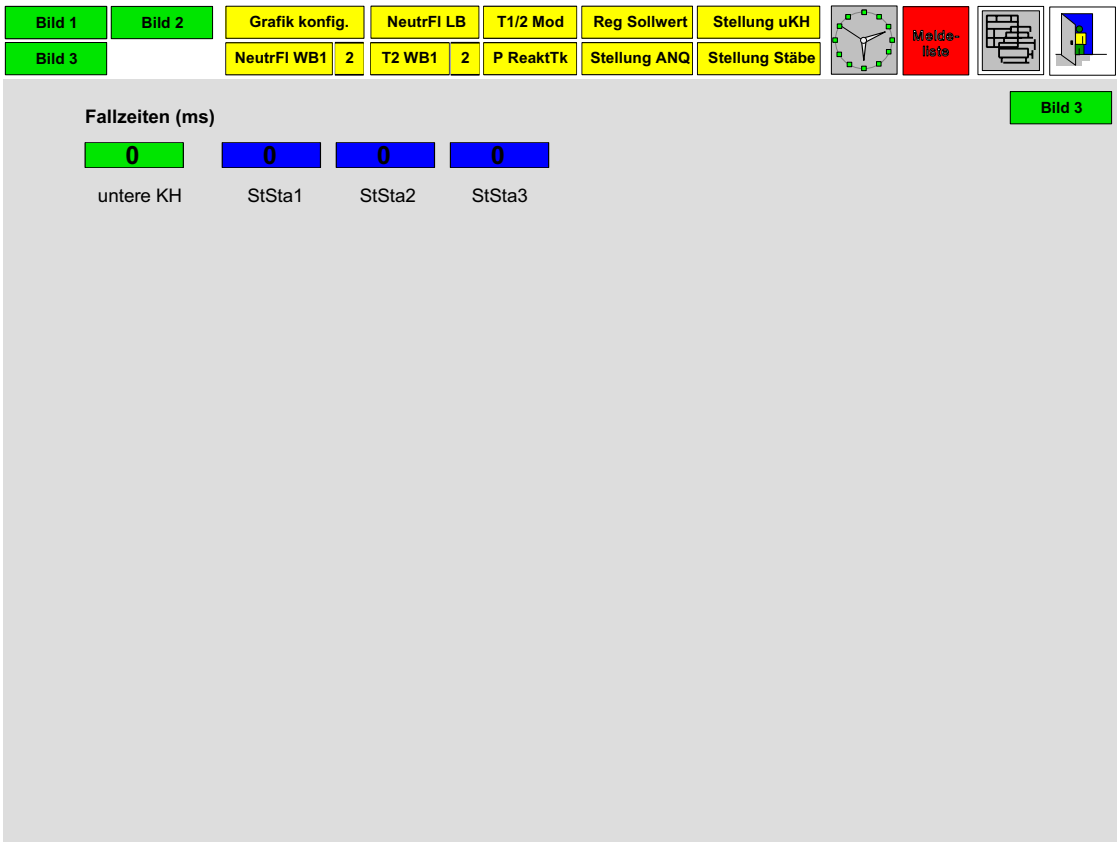


Fig. 8, Monitor Screen, Picture 3

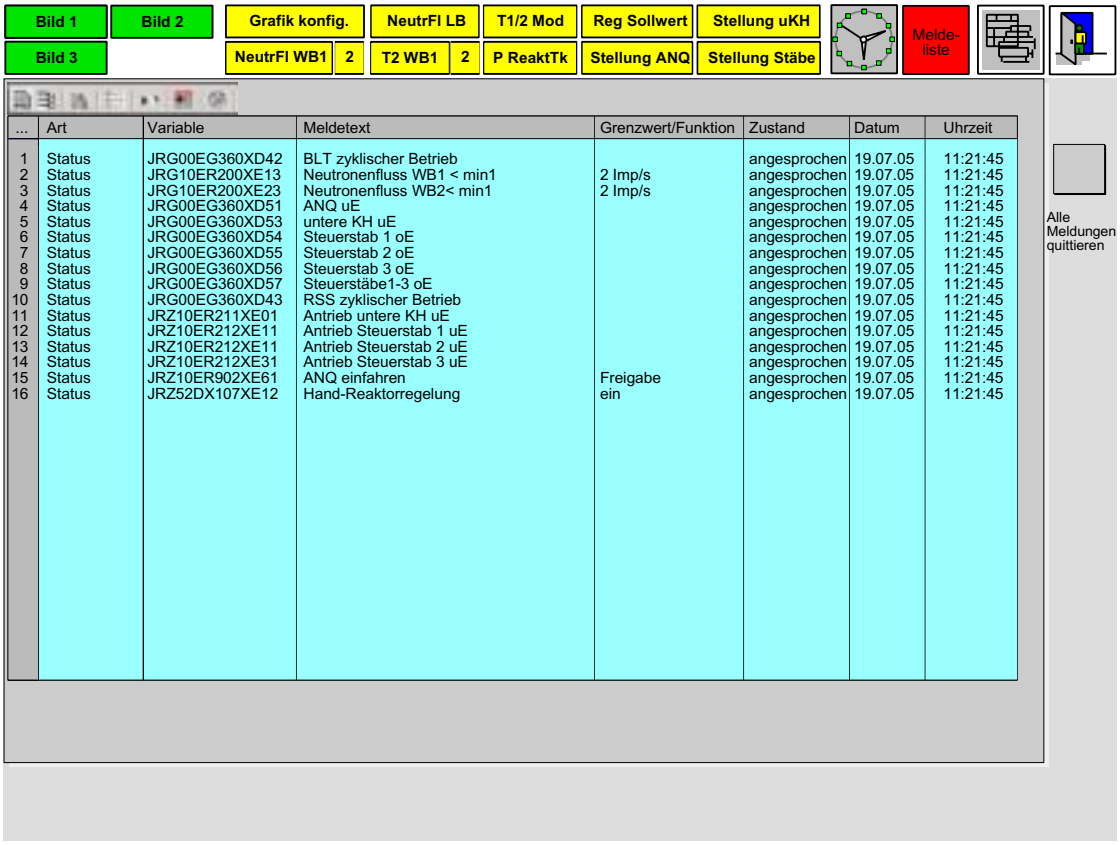
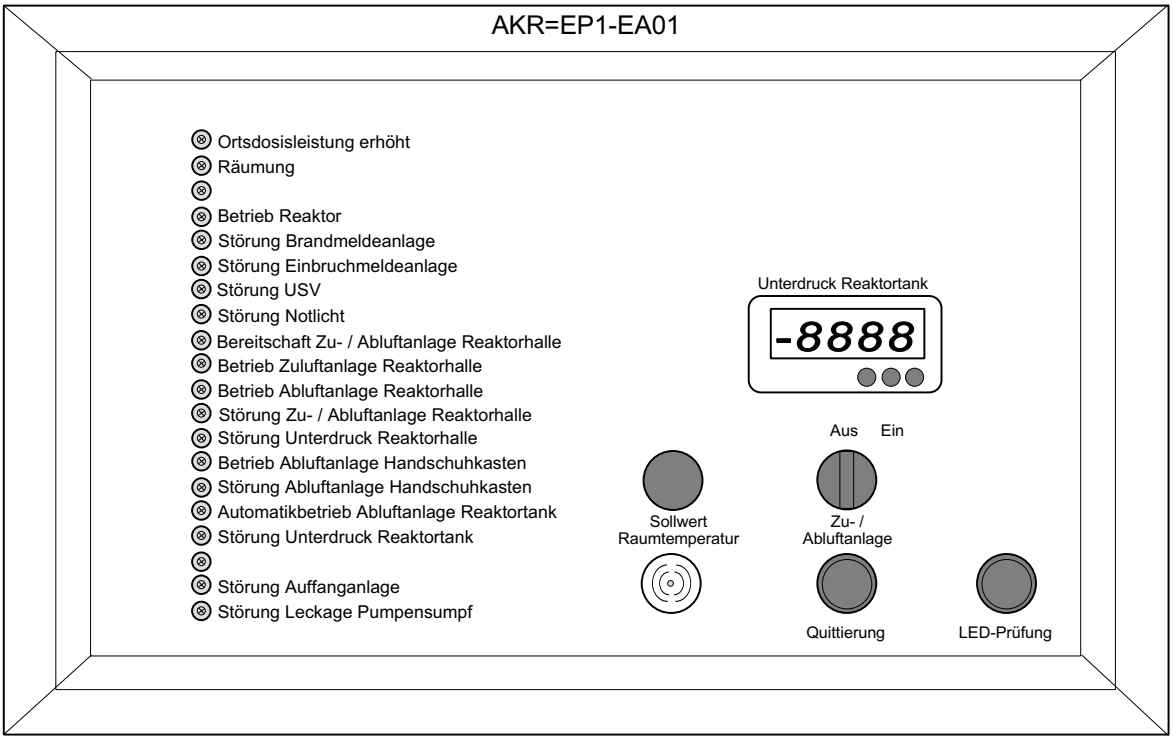


Fig. 9, Monitor Screen, List of System Messages (example)



Indicators:

Ortsdosisleistung erhöht	Local dose rate above threshold
Räumung	Evacuation
Betrieb Reaktor	Reactor in operation
Störung Brandmeldeanlage	Fault in fire alarm system
Störung Einbruchmeldeanlage	Fault in alarm system for detection of un-authorised access
Störung USV	Fault in uninterruptible power supply
Störung Notlicht	Fault in emergency light
Bereitschaft Zu-/Abluftanlage Reaktorhalle	Standby of ventilation in reactor hall
Betrieb Zuluftanlage Reaktorhalle	Operation of supply air system in reactor hall
Betrieb Abluftanlage	Operation of extracted air system in reactor hall
Störung Zu-/Abluftanlage Reaktorhalle	Fault in ventilation system in reactor hall
Störung Unterdruck Reaktorhalle	Fault in subatmospheric pressure in reactor hall
Betrieb Abluftanlage Handschuhkasten	Operation of extracted air system in glove box
Automatikbetrieb Abluftanlage Reaktortank	Automatic operation of subatmospheric pressure system in reactor tank
Störung Unterdruck Reaktortank	Fault in subatmospheric pressure system in reactor tank
Störung Auffanganlage	Fault in drain pan for possibly contaminated water
Störung Leckage Pumpensumpf	Fault in pump sump on basement floor
Unterdruck Reaktortank (kPa)	Value of subatmospheric pressure in reactor tank (kPa)

Keys / Switches:

Sollwert Raumtemperatur	Desired value of room temperature in reactor hall
Zu-/Abluftanlage Ein/Aus	Ventilation system in reactor hall ON/OFF
Quittierung	Acknowledgement of signals
LED-Prüfung	Functional check of LED-lights

Fig. 10, Signal and Key Panel for Peripheral Units

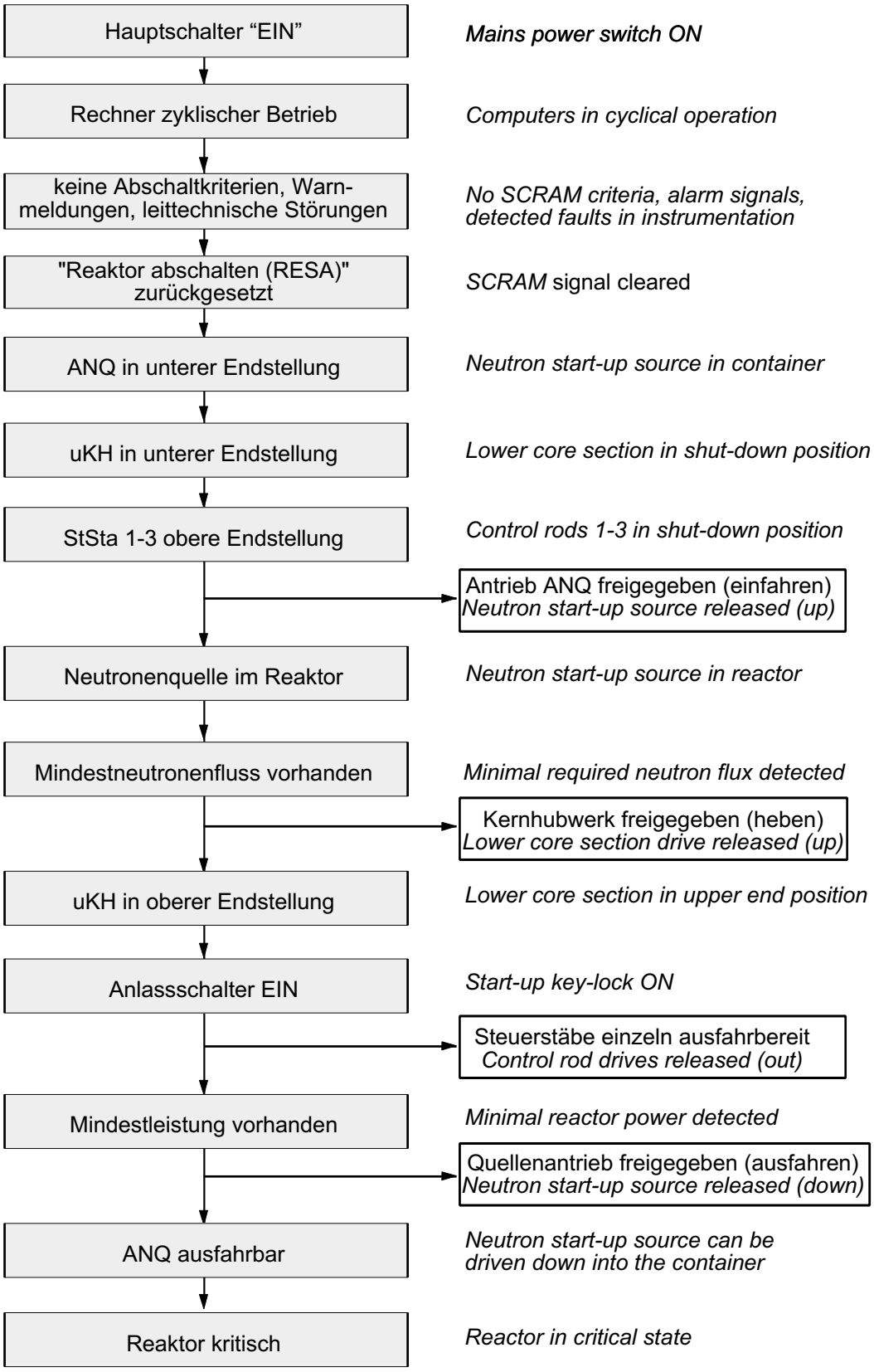


Fig. 11, Protective Logic for Reactor Start-up Procedure

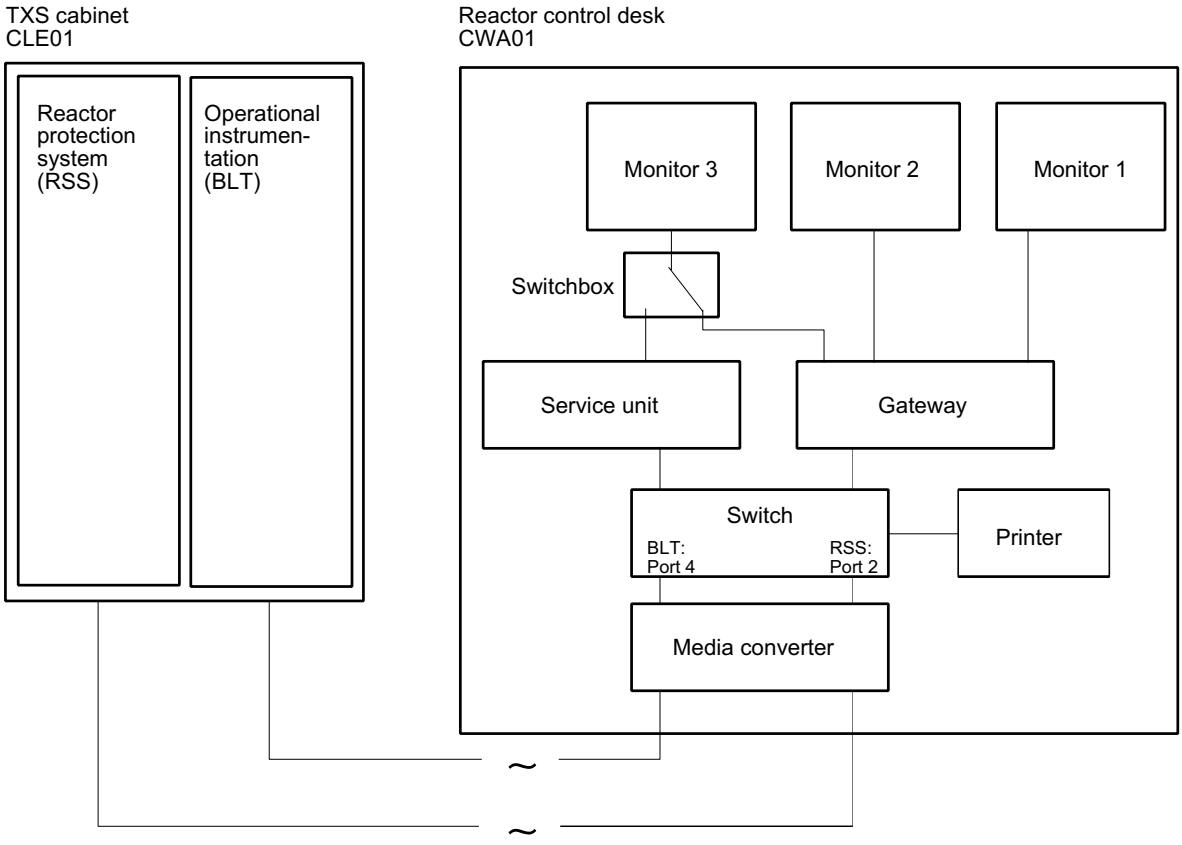


Fig. 12, Combination of the Safety and Control System (SUS)

Datum:			BA:	
Versuch:			Versuchsleiter:	
Überprüfung dosimetrische Einrichtung:			Uhr	
Überprüfung nichtnukleare Einrichtung:			Uhr	
Quellenantrieb:	Kernhubwerk:		Totalabschaltung:	
Beladung Brennstoff vom:			Experimentierkanäle vom:	
Bemerkungen:			Unterschrift:	
Temperatur Spaltzone:				
Quelle eingefahren:	Uhr		Kernhälften zusammen:	Uhr
Reaktor überkritisch:	Uhr		Verdopplungszeit::	s
Stab 1:		Stab 2:	Stab 3:	
Reaktor kritisch:	Uhr		Leistung:	W
Stab 1:		Stab 2:	Stab 3:	
Abschaltung:	Uhr			
Anlage AUS:	Uhr		Unterschrift:	

Fig. 13, Check List for Reactor Start-up and Operation

Comments to the check list of start-up procedure and reactor operation
(translation of and remarks to Fig. 13)

Datum :	actual date
BA :	no. of actual service instruction
Versuch :	experiment to be carried out (according to operational instruction)
Versuchsleiter :	person being responsible for the experiment
Überprüfung dosimetrische Einrichtung :	availability and proper operation of dosimetric devices
Überprüfung nichtnukleare Einrichtung :	availability and proper operation of conventional instrumentation
Quellenantrieb :	test of proper operation of neutron source drive (o.k.)
Kernhubwerk :	test of proper operation of drive and fast shut-down actions for lower core section (o.k.)
Totalabschaltung :	test of manual fast total shut-down action (SCRAM), released by pressing the red button on control desk (o.k.)
Beladung Brennstoff vom :	date of last fuel reloading in the core
Experimentierkanäle vom :	date of last change in configuration of the experimental channels
Bemerkungen :	remarks (optional)
Unterschrift :	signature of the operator
Temperatur Spaltzone :	measured temperature inside the core
Quelle eingefahren :	neutron source in core (time), digital position 800
Kernhälften zusammen :	lower core section lifted (time), digital position 506±1
Reaktor überkritisch :	reactor in supercritical state (time) (characterised by constant doubling time < ∞)
Verdopplungszeit :	measured constant doubling time in supercritical state
Stab 1, 2, 3 :	digital positions of control rods S1, S2, S3
Reaktor kritisch :	reactor in critical state (time) (characterised by constant doubling time = ∞)
Leistung :	corresponding reactor power, in Watt
Stab 1, 2, 3 :	digital positions of control rods S1, S2, S3
Abschaltung :	reactor shut-down (time)
Anlage AUS :	mains off (time)
Unterschrift :	signature of the reactor operator

Glossary (German / English)

Alarmmeldung	alarm signal / safety alarm
Betriebliche Leittechnik	operational I&C system, operational instrumentation
Betriebsanweisung (BA)	service instruction
Betriebsfunktionen	operational functions
Betriebsjournal	operation logbook
Freigabe	release
Gateway	Gateway interface computer
Leistungsmesskanal	(linear) power-range channel
Meldeliste	list of system messages
Meldetableau	(conventional) operation and signal panel
Nukleare Instrumentierung	nuclear instrumentation
rechnergestütztes Bedien- und Beobachtungssystem	computer-aided operational and supervising system
RESA	fast reactor shut-down (SCRAM)
Schlüsselschalter	key-lock
Service-Gerät	service equipment (diagnostic interface)
Sicherheitsleittechnik	reactor protection
Steuerpult	control desk
Störmeldung	fault signal
SUS	safety and control system
Systemsignalisation	system messages
Verriegelungskreis	protective logic
Weitbereichsmesskanal	(logarithmic) wide-range channel
Zustandsmeldung	status signal / status message

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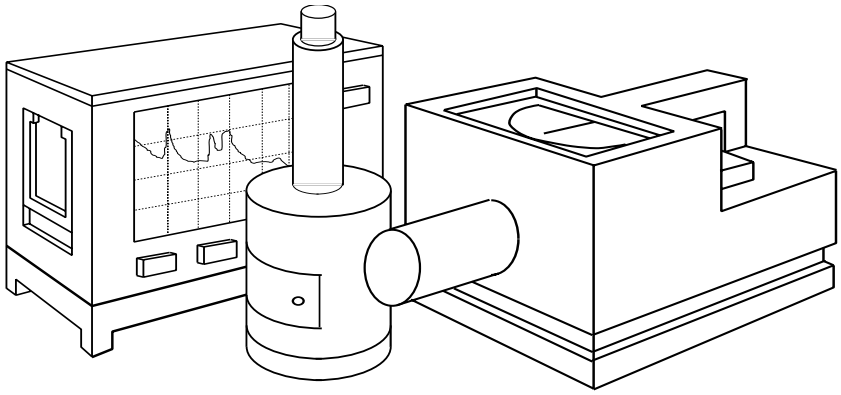
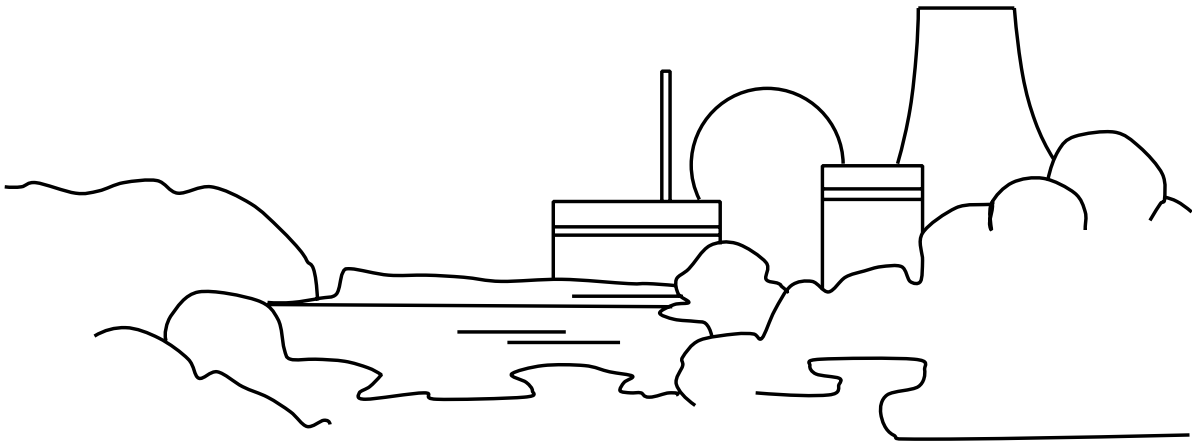
Institute of Power Engineering
Training Reactor



Reactor Training Course

Experiment

"Reactor Start-up Procedure"



Instruction for
Experiment "Reactor Start-up"

Content:

Figures:

(issued: March 2015)

1. Motivation

With help of training reactor AKR-2 of the Technical University Dresden it can be studied:
- the principle design of a thermal nuclear reactor,
- the function of main components required for a controlled nuclear fission chain reaction,
- basic design of a digital reactor instrumentation and control (I&C) system.

In order to understand the typical control behaviour, the results of the reactor theory which can be derived from the kinetic reactor equations should be known. By means of reactor power control display and of the reactor period readings the reactor status should be certainly recognised.

2. Tasks

1. The safety check of the reactor has to be carried out and recorded in the operation logbook in accordance with the pre-defined check list.
2. The reactor has to be put into operation by a normal restart and should become critical at that thermal power given in the service instruction (BA) or by the supervisor.
3. For qualitative investigation of the behaviour of a zero-power reactor, power changes have to be carried out according to instructions by the supervisor.
4. The gamma-ray dose rate has to be measured in dependence on thermal reactor power at selected test points in the reactor hall.
5. Critical control rod positions have to be determined and discussed in dependence on the thermal power of the reactor.

3. Types of Reactor Start-up

Starting-up a nuclear reactor means to generate a controlled nuclear fission chain reaction. This process needs to be controlled clearly and handled safely at all times.

The start-up of a nuclear power plant reactor, i.e. the restart from shut-down condition to high thermal power (> 1000 megawatts) is an extraordinary complex process because of the necessary heat removal. The basics of reactor physics can be studied more easily at so-called **Zero-Power Reactors** like the AKR-2.

Zero-power reactors are operated only at such low power (watts to kilowatts) that power effects like
- fuel burn-up,
- poisoning or
- temperature effects

can be virtually neglected during the time of the training. Hence, every start of such a zero-power reactor corresponds, regarding its nuclear part, to a cold start-up of a power reactor.

Any start-up of zero-power reactors can be classified into three categories:
- normal restart,
- comprehensive start-up experiment,
- critical experiment.

The **normal restart** is the start-up in routine reactor operation. Restarts are possible, if nothing has changed since last reactor operation with effect on the reactivity properties of the facility, neither in the reactor itself nor in the internal experimental arrangements.The reactivity characteristics and the control rod positions in the critical condition of the reactor are known.

A **comprehensive start-up** experiment is necessary for a safe reactor start-up after minor material or geometry changes (e.g. after the installation of new internal experimental arrangements) which do not require a critical experiment. The behaviour and the reactivity properties of the reactor are known, i.e. the fuel loading, the reflector properties, and the reactivity characteristics of the control rods.

A **critical experiment** has to be carried out, if the physical parameters (e.g. critical mass, reactivity characteristics of the control rods, excess reactivity etc.) are known from calculations only. This applies, in any case, to the first start-up of a newly build reactor and to any further start-up after variations of the assembly which let expect a considerable change in the reactivity behaviour.

4. Theoretical Background

At zero-power reactors, the neutron flux density Φ (unit: neutrons/(cm² · s)) is the quantity to be controlled. It is proportional to the neutron density (unit: neutrons/cm³) as well as to the reactor power (unit: watt) and to the total number of the neutrons N in the reactor.

Normally, the shut-down reactor contains almost no free neutrons. Its state regarding neutron reproduction is characterised by the multiplication factor **k < 1**. This **subcritical condition** is maintained by material and/or geometric conditions in the core, e.g. due to addition of neutron absorbers, removal of moderator or reflector material, separation of the core into subcritical masses, etc. By gradually eliminating these conditions, the reactor state can be converted to the **critical condition (k = 1)** or to a **supercritical condition (k > 1)**.

In this process it is remarkable, that the neutron flux density rises by several orders of magnitude, i.e. from a few neutrons up to many millions of neutrons per second and cm² depending on the eventual power. Therefore, certain start-up rules have to be kept to avoid start-up incidents.

These rules avoid
- mistakes of the reactor operators resulting from misinterpretation of the reactor state,
- too high reactivity change rates,
- a too high excess reactivity ($\rho_{excess} > ß$).

Since criticality can be achieved also at very low power levels, there is the danger of exceeding the critical point already at very low neutron flux densities and without the reactor operator being aware of it. Because of the statistical fluctuations at low neutron flux densities, measurements require detectors with a sluggish time response. The time delay between the occurrence and the display of flux density changes may cause the operator to assess the reactor state incorrectly and possibly to increase the multiplication factor to inadmissible supercritical ranges. The resulting very fast increase of the neutron flux density might be recognised only when intervention is

already impossible. Then, harm to the operational staff and damage to the facility could be a consequence.

Thus, following prerequisites ensure a safe reactor start-up:

- The reactor must be started from an adequately high value of the neutron flux density. The initial neutron flux density has to be raised such that the neutron detectors deliver proper values. Usually, at zero-power reactors, an artificial (external) neutron source is used for controlling the subcritical condition. The subcritical multiplication of the source neutrons provide the reqiured values of the flux density for the neutron detectors of the reactor instrumentation. At high power reactors, also neutrons from spontaneous fission and (γ,n)-reactions may be sufficient for this purpose.
- Limitation of the positive reactivity change rate:
  Only those positive values of the reactivity change rate are allowed that
  - the reactor can be controlled safely manually during normal operation and
  - the I&C-system of the reactor is able to shut-down the reactor reliably if required (international recommended limit: $(d\rho/dt)_{max} = 10^{-4}\ s^{-1}$).
- Limitation of the excess reactivity:
  The excess reactivity is limited to values of $\rho_{excess} < ß$ (at AKR-2: $\rho_{excess} \approx 0.3\ \%$).

4.1. Prompt and Delayed Neutrons

Fission of nuclei of the reactor fuel (e.g. U-235) releases fast neutrons. These neutrons are released either immediately after the nuclear fission as the so-called **prompt neutrons** or originate from a special case of radioactive decay as the so-called **delayed neutrons**.

The average time between the birth of a fast neutron and the next nuclear fission after slowing down and diffusion with the formation of a new generation of fission neutrons is called **neutron life time l**.

The neutron life time consists of the moderation time (from fast to thermal energy, about 10^{-5} s), the diffusion time (entering of the neutron into the fissionable nucleus, about 10^{-4} s) and the reaction time (nuclear fission, about 10^{-15} s).

Consequently, the total time is determined by the longest part, which is the diffusion time.

All neutrons that arise within one neutron life time are considered as one neutron generation. These neutrons vanish by leakage out of the reactor ($N_{leakage}$) and by absorption (N_{abs}) within the reactor core. Partly, absorption causes new fissions (provided that the absorption is in the fuel).

At the time t + l, a new neutron generation has been produced (N_{gen}). Therefore, the multiplication factor k can be written in the following form:

$$k = \frac{N_{gen}}{N_{abs} + N_{leakage}} = \frac{number\ of\ neutrons\ at\ time\ t + l}{number\ of\ neutrons\ at\ time\ t} \qquad \textbf{(1)}$$

Consequently, the number of neutrons per time unit being in the reactor is the balance of those neutrons being produced ($k \cdot N / l$) and those disappearing ($- N / l$). Hence, the change of the number of neutrons is given by:

$$\frac{dN}{dt} = k \frac{N}{l} - \frac{N}{l} = N \frac{k - 1}{l} \qquad \textbf{(2)}$$

or

$$\frac{dN}{N} = \frac{k - 1}{l} dt \qquad \textbf{(3)}$$

which is solved by

$$N(t) = N_0 \cdot e^{\frac{k-1}{l} \cdot t} = N_0 \cdot e^{\frac{t}{T}} \qquad with\ T = \frac{l}{k - 1} \qquad \textbf{(4)}$$

where T is called **reactor period**. The reactor period is that time interval during which the neutron number N (or in same way **Φ**, n or P) increases by a factor of e (≈ 2.71). Because it is more convenient to determine a change for a factor of 2 (instead of e), it is often common practice to use the **doubling time** T_2 instead of reactor period. Reactor period and doubling time are connected with each other by the simple relation:

$$T_2 = ln2 \cdot T \qquad \textbf{(5)}$$

Thus, according to equation (4), the time behaviour of a reactor follows always an exponential function.

For the assumption that only prompt neutrons would exist in a nuclear reactor (with typical life times $l \approx 10^{-4}$ s), a change of the multiplication factor from 1 to 1.001 would cause an enormous increase of the neutron number and consequently of the reactor power within only one second by a factor of

$$\frac{N(t + 1s)}{N(t)} = e^{\frac{1.001 - 1}{0.0001\ s} \cdot 1s} \approx 22000 \qquad \textbf{(6)}$$

Such dynamics would be virtually uncontrollable. A nuclear reactor with a <u>controlled</u> chain reaction would not be feasible.

The control of the chain reaction is possible only due to the existence of the delayed neutrons, whose properties are summarised in Tab. 1. Delayed neutrons arise in 6 groups (from 6 groups

of radioactive precursor nuclei as a very special case of nuclear decay) with half-lives $T_{1/2}$ between 0.23 s and 56 s. From fission of U-235 only a fraction of ß = 0.641 % is released as delayed neutrons. The remaining 99.359 % are prompt neutrons.

Group i	pre-cursor	average kinetic energy / keV	$T_{1/2}$ / s	$\lambda_i=\ln 2/T_{1/2}$ / s^{-1}	fraction $ß_i$ compared to all fission neutrons / %	relative fraction $a_i = ß_i / ß$	neutrons per 10^3 fissions (absolute value)
1	Br-87	250	55.72	0.0124	0.021	0.033	0.52
2	J-137	560	22.72	0.0305	0.140	0.219	3.46
3	Br-89	430	6.22	0.111	0.126	0.196	3.10
4	?	620	2.30	0.301	0.253	0.395	6.24
5	?	420	0.61	1.14	0.074	0.115	1.82
6	?	-	0.23	3.01	0.027	0.042	0.66
total					0.641	1.000	15.80

Tab. 1, Properties of delayed neutrons caused by fission of U-235 /Reactor Physics Constants, ANL-5800/

The fraction of delayed neutrons compared to all fission neutrons can also be derived using the last column of Tab. 1. Considering an average of 2.47 released neutrons per fission of a U-235 nucleus, it results ß = 15.80/1000/2.47 = 0.0064 = 0.64 %.

The value ß can vary between 0.5 % and 0.7 % depending on the fuel enrichment, on the moderator temperature, and in particular on fuel burn-up. At a given nuclear reactor, the fraction of delayed neutrons is called $ß_{eff}$.

For reasons of simplicity, the 6 groups of delayed neutrons can be approximately condensed to one single group with averaged parameters. Thus, using the values of Tab. 1, the average life time $l_{delayed}$ of the delayed neutrons, can be written as

$$l_{delayed} = \frac{1}{\lambda} = \frac{1}{\beta} \sum_{i=1}^{6} \frac{\beta_i}{\lambda_i} = 13.00\ s \qquad \textbf{(7)}$$

This corresponds to an average decay constant $\lambda = 0.0769\ s^{-1}$.

For prompt and delayed neutrons together, an effective life time can be calculated by

$$\bar{l} = l_{delayed} \cdot 0.00641 + l_{prompt} \cdot 0.99359 = 0.083\,s \qquad \textbf{(8)}$$

If the power increase within one second is now calculated again according to (6), taking into account a change of the multiplication factor from 1 to 1.001, it results

$$\frac{N(t + 1\,s)}{N(t)} = e^{\frac{1.001 - 1}{0.083\ s} \cdot 1\,s} \approx 1.01 \qquad \textbf{(9)}$$

i.e. an increase of reactor power by only 1 %, which can be controlled without any difficulties.

4.2. Subcritical reactor

It can be assumed that the reactor does not contain any free neutrons at the beginning of start-up ($n_{(t=0)} = 0$). At the time t = 0, a neutron source shall be inserted into the reactor. After t = l (l = life time of a neutron generation), the source has produced an average neutron source density $n_{Source} = S \cdot l$. After another neutron generation life time l, this value has increased by the factor k according to equation (1) while, at same time, the source has released additional $n_{Source} = S \cdot l$ neutrons. Hence, one has

$n(t=l) = n_1 = S \cdot l$
$n(t=2 \cdot l) = n_2 = S \cdot l + S \cdot l \cdot k$

After another life time, n_2 has got multiplied by k again while at same time, the source has released additionally $n_{Source} = S \cdot l$ neutrons, resulting in

$n(t=3 \cdot l) = n_3 = S \cdot l + (S \cdot l + S \cdot l \cdot k) \cdot k$

or in general

$n(t) = S \cdot l \cdot (1 + k + k^2 + k^3 + \ldots)$

This is a geometric series that has the limit (in given case of a subcritical reactor with k < 1)

$$n(t = \infty) = S \cdot l \frac{1}{1 - k} \qquad \textbf{(10)}$$

The factor

$$M = \frac{1}{1 - k} \qquad \textbf{(11)}$$

is called the **subcritical multiplication factor**.

Equation (10) has the following consequences:
- The subcritical reactor acts as a neutron amplifier (for the neutrons released by the source).
- For the critical reactor (k = 1), after inserting a neutron source, the neutron density would increase linearly up to infinity. That is because after every single fission, exactly one neutron

is left for a new fission, while the neutron source constantly adds more neutrons. Therefore, all neutrons from the source sum-up in linear dependence.
However, this source effect can be noticed only at very low reactor powers since already at a power level of 1 W the neutrons being added by the source contribute only a small fraction of the total neutron contents in the facility.

For a subcritical reactor with a neutron source inserted, the **time dependence of the neutron density** is described by:

$$n(t) = \frac{S \cdot l}{1 - k} \cdot \left(1 - e^{-\frac{(1-k)}{l} t} \right) \qquad \textbf{(12)}$$

Because of the negative exponent, the neutron density in the reactor asymptotically approaches the value:

$$n_{\infty} = \frac{S \cdot l}{1 - k} = \frac{S \cdot l^{*}}{-\rho} \qquad \textbf{(13)}$$

with $l^{*} = l / k$ und $\rho = (k - 1) / k$. A comparison with equation (11) shows that the neutrons of the source have been amplified by the subcritical multiplication factor. For values of k approaching 1, the subcritical amplification gets larger and larger. On the other hand, according to equation (12), the time increases more and more until the approximation to the asymptotic limit value of the neutron density (see also Fig. 1)

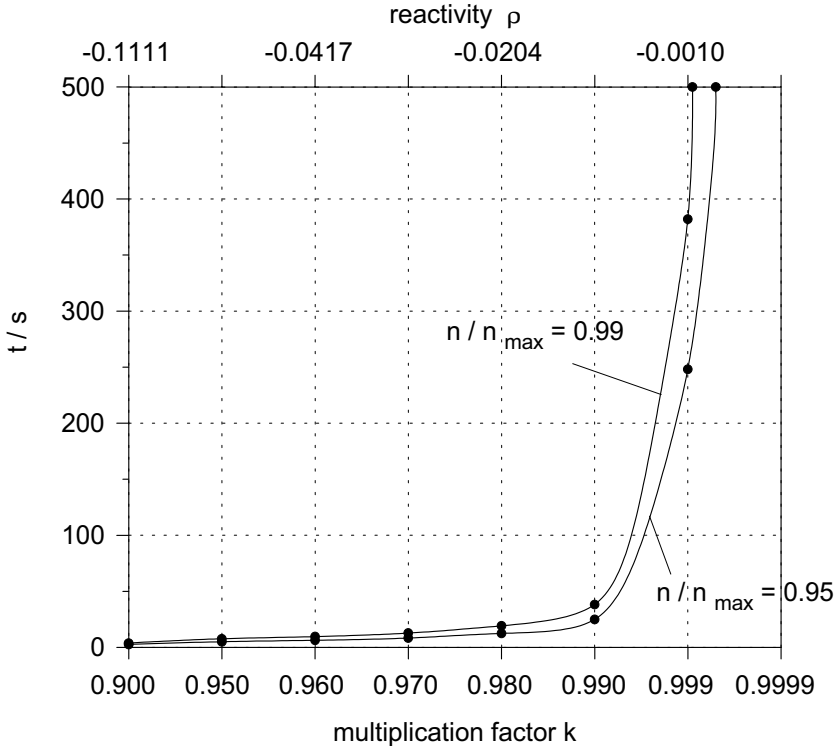


Fig. 1, Times required for 95 % and 99 % approximation to the maximum asymptotic neutron density n_{max} in dependence on the multiplication factor k and the reactivity ρ

4.3. Supercritical reactor

With ß being the fraction of delayed neutrons arising from nuclear fission, the number of prompt neutrons being produced in the reactor within one neutron life time is

$$n_{prompt} = k \cdot n - \beta \cdot k \cdot n = n \cdot k \ (1 - \beta) \qquad \textbf{(14)}$$

For the change of the prompt neutron density within one neutron life time one has:

$$\frac{dn_{prompt}}{dt} = \frac{n \cdot k \ (1 - \beta) - n}{l} = \frac{n}{l} \ [k \cdot (1 - \beta) - 1] \qquad \textbf{(15)}$$

At the same time, delayed neutrons $n_{delayed}$ result from radioactive decay of certain radionuclides (delayed neutron precursors) which have been produced in previous fissions. The number of delayed neutrons can be calculated by means of the law of radioactive decay (considering all 6 groups of delayed neutrons):

$$\frac{dn_{delayed}}{dt} = \sum_{i=1}^{6} \lambda_i \cdot C_i \qquad \textbf{(16)}$$

Consequently, the total change of the neutron density within one neutron life time is:

$$\frac{dn}{dt} = \frac{dn_{prompt}}{dt} + \frac{dn_{delayed}}{dt} + S = \frac{n}{l} \ [k \cdot (1 - \beta) - 1] + \sum_{i=1}^{6} \lambda_i \cdot C_i + S \qquad \textbf{(17)}$$

Additional application of $l^* = l / k$ und $\varrho = (k - 1)/k$ to equation (17) gives the so-called **reactor kinetic equations**:

$$\begin{aligned} \frac{dn}{dt} &= \frac{\varrho - \beta}{l^*} \cdot n + \sum_{i=1}^{6} \lambda_i \cdot C_i + S \\ \frac{dC_i}{dt} &= \frac{\beta_i}{l^*} \cdot n - \lambda_i \cdot C_i \qquad (i = 1, \ldots, 6) \end{aligned} \qquad \textbf{(18)}$$

with $n \cdot ß_i / l^*$ being the number of generated delayed neutron precursors and $\lambda_i \cdot C_i$ being the number of precursors decaying under emission of one neutron. Equations (18) are a system of 7 coupled differential equations. A quite simple approximated solution is achievable by summarizing the 6 groups of delayed neutrons in only one single group with the following averaged values (see also chapter 4.1.):

$$\beta = \sum_{i=1}^{6} \beta_i \qquad \frac{1}{\lambda} = \frac{1}{\beta} \sum_{i=1}^{6} \frac{\beta_i}{\lambda_i} \tag{19}$$

Thus, the system of differential equations will reduce to only 2 coupled differential equations:

$$\begin{aligned} \frac{dn}{dt} &= \frac{\rho - \beta}{l^*} \cdot n + \lambda \cdot C + S \\ \frac{dC}{dt} &= \frac{\beta}{l^*} \cdot n - \lambda \cdot C \end{aligned} \tag{20}$$

For a reactivity jump ($\rho = 0$ für $t < 0$ und $\rho = \text{const}$ für $t \geq 0$) and if additionally the source neutrons S are neglected, equation (20) results in

$$n(t) = n_0 \left[\frac{\beta}{\beta - \rho} e^{\frac{\lambda \cdot \rho}{\beta - \rho} \cdot t} - \frac{\rho}{\beta - \rho} e^{-\frac{\beta - \rho}{l^*} \cdot t} \right] \tag{21}$$

By means of equation (21) two important reactor conditions can be discussed:

4.3.1. Delayed supercritical reactor ($\rho < ß$)

In this case, the exponent of the second term in equation (21) is negative and has, because of $l^* \approx 10^{-4}$ s , a large absolute value. Consequently, this term vanishes within a few seconds and only the first term in equation (21) remains after short time.

$$n(t) = n_0 \frac{\beta}{\beta - \rho} e^{\frac{\lambda \cdot \rho}{\beta - \rho} \cdot t} \tag{22}$$

Therefore, it can be seen that a sudden increase of the reactivity results in a sharp rise of the neutron density, which is called **prompt jump** as it is caused by prompt neutrons. The height of this jump is

$$\frac{\Delta n_{prompt}}{n_0} = \frac{\beta}{\beta - \rho} - 1 = \frac{\rho}{\beta - \rho} \tag{23}$$

Considering prompt neutrons only the reactor is subcritical (since ρ < ß). The neutron density increases only due to subsequent addition of delayed neutrons, and that according to the stable reactor period T_s (see Fig. 2). The value of the **stable reactor period** can be derived from equation (22) to

$$T_s = \frac{\beta - \rho}{\lambda \cdot \rho} \qquad \textbf{(24)}$$

This description of the reactor period, which was derived for the approximation of only one averaged group of delayed neutrons, is sufficiently precise for small values of the reactivity (ρ < 0.001).

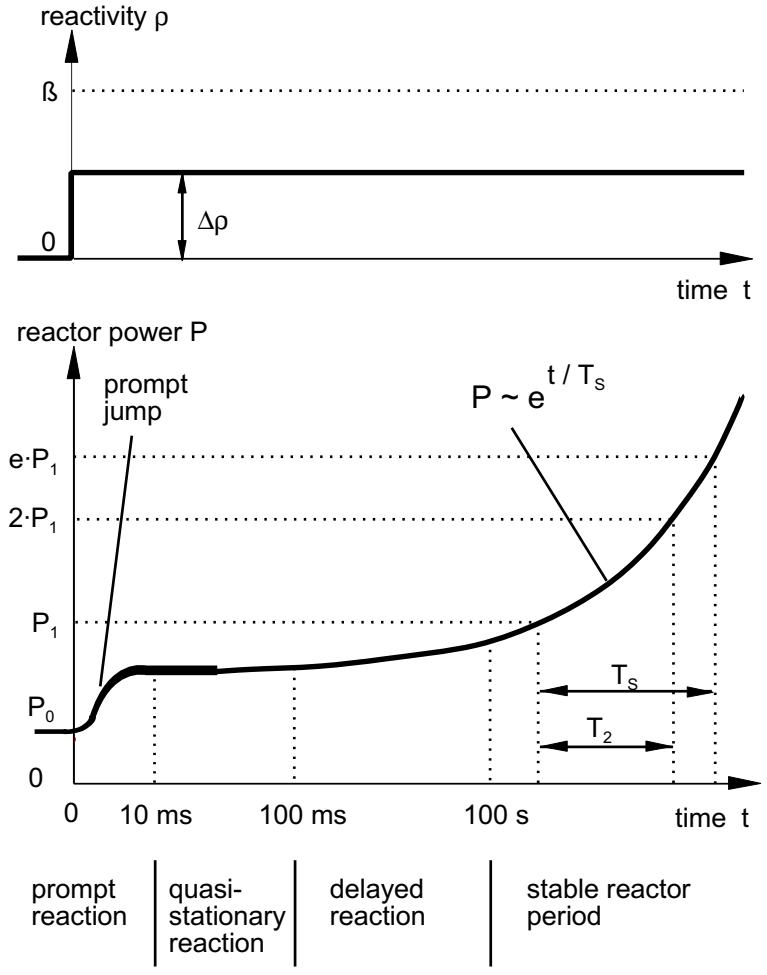


Fig. 2, Dependence of reactor power on time following a positive reactivity jump (0 < ρ < ß)

4.3.2. Prompt Supercritical Reactor (ρ > ß)

In this case, both, the second term in equation (21) becomes positive as well as its exponent. Consequently, the neutron density rises very fast (because of $l^* \approx 10^{-4}$ s). The reactor is supercritical only by the prompt neutrons. The reactor period would be in the order of milliseconds. Thus, the reactor power would increase so fast that the control rods could not be used reasonably for reactor control. This case is the accident of an uncontrollable power excursion and must never occur. Therefore, it is of high importance to keep ρ < ß at any time and under any circumstances.

Table 2 summarises possible reactor conditions, resulting reactor power behaviour and the corresponding multiplication factors k and reactivity values ρ.

Because of the exceptional safety relevant importance of the transition from the controlled reactor to the uncontrollable reactor (i.e. from the delayed supercritical condition to the prompt supercritical condition), a particular artificial unit ρ' has been introduced for describing the reactivity. By relating the reactivity ρ to the fraction of delayed neutrons ß the reactivity ρ' = ρ / ß is defined with the advantage that in case of safe delayed supercritical reactor ρ' remains smaller than 1. For distinguishing the two definitions of reactivity, the quantity ρ' has been added by the arbitrary unit $ (Dollar, 1 $ = 100 Cents). Thus, the transition from the delayed supercritical condition to the prompt supercritical condition occurs at the impressive value of ρ' = 1 $.

Reactor Condition	**Reactor Power Behaviour**	**Multiplication Factor k**	**Reactivity**	
			ρ = (k - 1) / k	ρ' = ρ / ß [$]
subcritical		< 1	< 0	< 0
critical		= 1	= 0	= 0
(delayed) supercritical		1 < k < 1+ß	0 < ρ < ß	0 < ρ' < 1 $
prompt supercritical		k ≥ 1+ß	ρ ≥ ß	ρ ≥ 1 $

Tab. 2, Summary of possible reactor conditions with corresponding reactor parameters

4.4. Influence of Reactivity Change Rate

For the subcritical reactor, differentiation of equation (13) gives the neutron density change rate as a function of the reactivity change rate:

$$\frac{dn}{dt} = \frac{S \cdot l^*}{\rho^2} \cdot \frac{d\rho}{dt} \qquad \textbf{(25)}$$

With the general definition of the reactor period

$$T = \frac{n}{(dn/dt)} = \frac{P}{(dP/dt)} \qquad \textbf{(26)}$$

and using equation (25), the reactor period for the subcritical reactor results in:

$$T = \frac{-\rho}{(d\rho/dt)} \qquad (\ \rho < 0\ ) \qquad \textbf{(27)}$$

Considering an infinitely slow approach to criticality, i.e. $\lim(d\rho/dt) \rightarrow 0$, the power of the subcritical reactor is given by the subcritical amplification value according to equation (13), at all times. Criticality would be achieved only after infinitely long time (Fig. 3). However, in the meantime, the reactor power had been grown to infinity.

On the other hand, for limited positive values of dρ/dt, the reactor becomes critical at limited values of the reactor power whereby a higher reactivity change rate corresponds to a lower reactor power when achieving criticality. In the deep subcritical reactor condition, the reactivity change rate has virtually no effect on the change rate of the neutron flux density.

For adding positive reactivity with a constant rate, i.e. dρ/dt = const, the reactor period decreases with linear correspondence to the remaining amount of negative reactivity in the reactor.

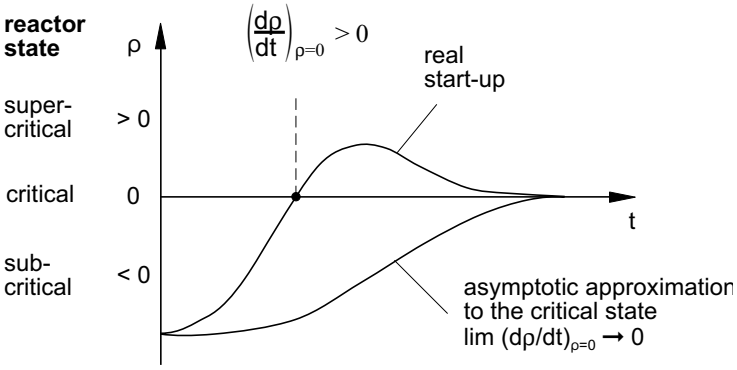


Fig. 3, Behaviour of the reactivity for an infinitely slow reactor start-up and for a real reactor start-up

The more the reactor approaches to its critical condition, the more the neutron density (and reactor power and reactor period as well) depend on the reactivity change rate.

The reactor start-up would be very time consuming, if the operator would wait for the approximation to the (subcritical) asymptotic limit of the reactor power n_∞ after each increase of reactivity. Instead of this, it is the common procedure of a normal reactor restart to rise the criticality first onto the supercritical reactor condition and then adjust criticality afterwards at the desired power level (compare to Fig. 3).

When approaching the desired reactor power, the (negative) reactivity change rate is continuously lowered to values that are appropriate for the respective differences of the actual reactor power and the desired reactor power. The reactor is critical (k = 1), if the stable reactor period is infinite ($T_S = \infty$).

5. Procedure of the Experiment

5.1. Design and Operation of the AKR-2

The AKR-2 is a homogeneous thermal zero-power reactor. A detailed description of its design and operation is given in Ref. /2/. A basic functional layout of the AKR-2 facility is shown in Fig. 4. For safety reasons, the cylindrical core (diameter 250 mm, height 275 mm) consists of two separate sections. Each section contains still the initial fuel loading as it was assembled stepwise from the disk-shaped fuel elements having various thicknesses in the critical experiment during commissioning of the reactor in 1978.

The fuel elements consist of a homogeneous mixture of nuclear fuel (uraniumoxide, enriched to 19.8 % U-235) and the moderator (polyethylene). The critical mass of the core is about 790 g of U-235. The core is surrounded on all sides by a graphite reflector of about 30 cm thickness.

Both, the upper and the lower core sections are each hermetically enclosed in an aluminium container. A second, larger gas-tight reactor tank encloses the two core sections and parts of the reflector. The pressure inside the reactor tank is lowered by (8…18) kPa compared to the ambient atmospheric pressure. This subatmospheric pressure barrier prevents an uncontrolled leakage of radioactive fission products even in the unlikely case that all the other internal retention barriers would fail.

In the shut-down condition of the reactor, the lower core section is lowered by about 50 mm. When starting the reactor, it is moved upwards by a drive mechanism and a threaded spindle including an electromagnetic holder of the core section until close contact with the upper core section is given. Prior to this, inside the spindle, the start-up neutron source (^{241}Am-Be, source strength $2.2 \cdot 10^6$ s^{-1}) is driven up to the lower side of the core.

Three control rods made from cadmium are available for both, the control of the reactor and for safety shut-down. The positions of the neutron source, of the lower core section and of the three control rods are displayed digitally and analogously on the monitor screens at the control desk.

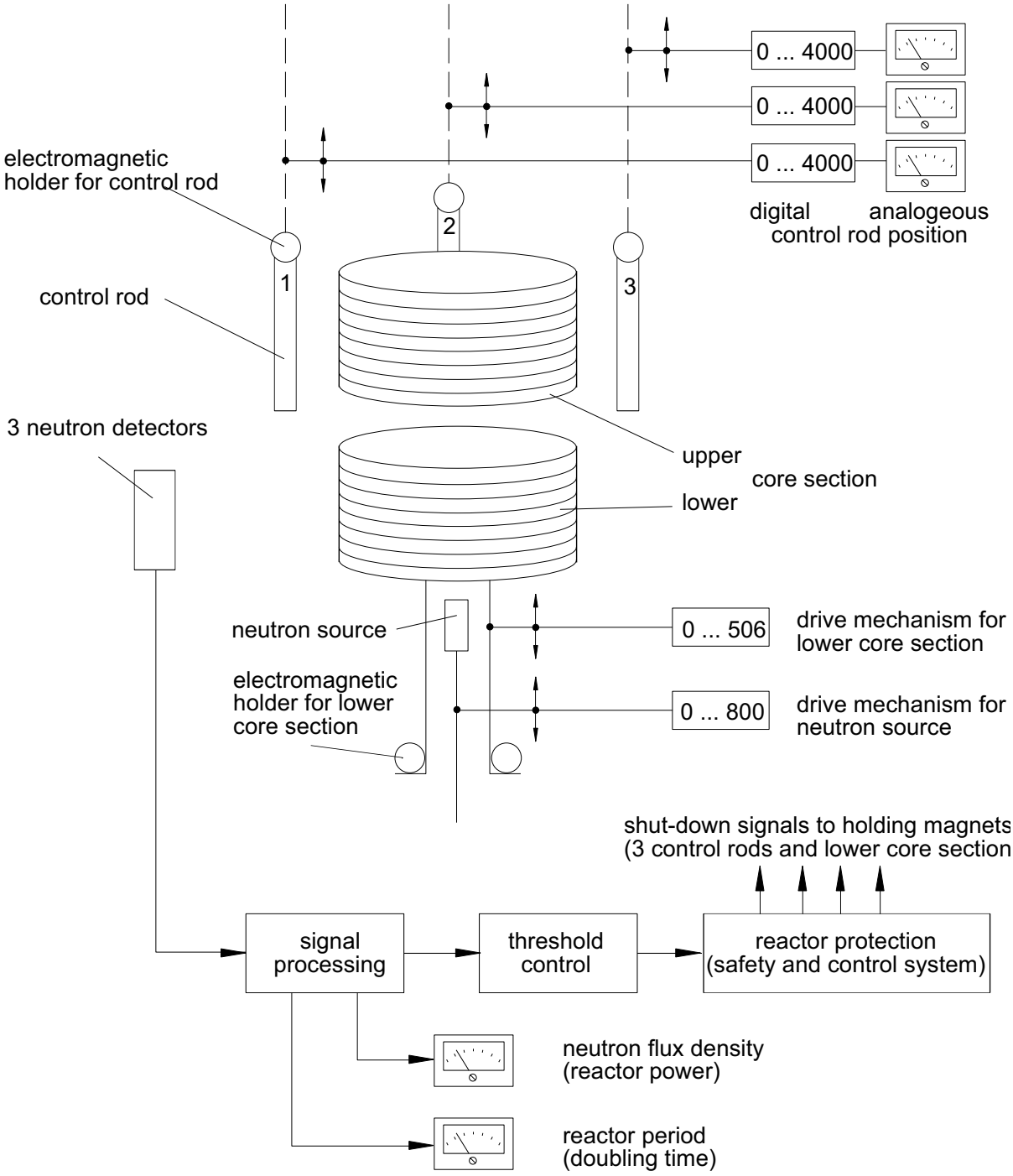


Fig. 4, Functional layout of the AKR

Three neutron detectors of different types and sensitiveness are used for measuring the neutron flux density in the reactor and in this way for supervising the operating condition of the reactor. They provide electrical signals (pulse densities, currents) that are proportional to the neutron flux density and are used, in consequence, for calculation of reactor power and reactor period.

These signals are supervised with safety threshold monitors. When exceeding the pre-defined thresholds, the safety and control system (SUS-System) gives shut-down (SCRAM) signals that release the holding magnets of the control rods and of the lower core section. These magnets lose their electricity supply and consequently the control and safety units (i.e. lower core section and all three control rods) drop down and interrupt the nuclear chain reaction, i.e. the reactor is shut-down.

5.2. Normal Restart

It must be definitely excluded that inadmissible operating conditions and disturbances during the start-up of a reactor due to improper operation can occur. Therefore, the instrumentation is designed in such a way that necessary safety requirements are kept automatically and undue actions of the operator have no effect or lead to automatic reactor shut-down.

During the reactor start-up, the protective logic ensures the correct sequence of actions, i.e. the start-up procedure can only succeed if a certain pre-defined sequence of starting conditions is maintained and by strictly following the given necessary actions.

A flow chart of the start-up procedure is given in Fig. 5.

The processing and completion of the particular steps is indicated on the monitor screens of the control desk. The operating condition of the reactor can be concluded from the time behaviour of the reactor power and the reactor period (or doubling time). Warning and alarm signals inform the operator about inadmissible operational parameters both, visually and acoustically.

The safety and control system (SUS) supervises compliance with the given threshold conditions at any time, already during the reactor start-up. It causes automatic reactor shut-down (SCRAM) under the following conditions:
- any fault in the protective logic
- neutron flux density > 120 % in 1 of the 3 neutron measuring channels
- doubling time of the reactor power < 10 s in 1 of 2 neutron measuring wide-range channels.

Before every start-up of the reactor, all electronic and mechanical parts that are essential for a safe operation have to be checked.

The procedures of both, the functional check and the start-up procedure have to follow fixed sequences (see Ref. /2/). The results have to be documented in the operation logbook.

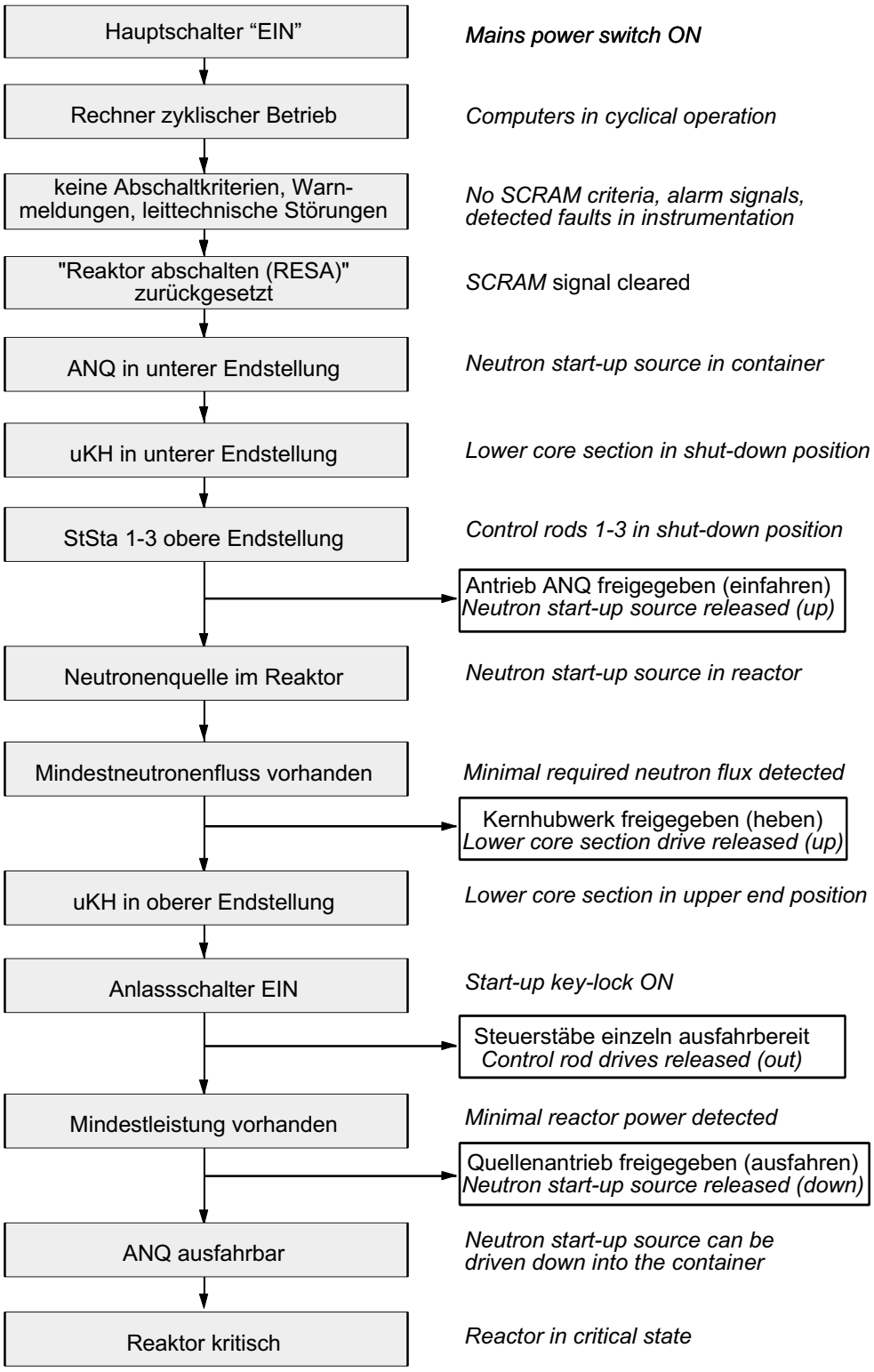


Fig. 5, Flow chart of the start-up procedure (protective logic)

Consequently, the reactor start-up has to be carried out in the following order:

1. Information about pre-defined conditions of the subsequent reactor start-up by reading the adequate service instruction (BA) and about previous conditions of reactor operation, especially about corresponding critical control rod positions, by reading the entries in the operation logbook.

2. Functional check in accordance with test instruction. The reactor is free to be started-up only at full availability of all components of the system!

3. Reactor start-up in accordance with the check list for reactor start-up and operation.

Remarks:
- The neutron source can be withdrawn at reactor power values > 0.25 W.
- If the reactor power has about 80 % of the desired power level, the (positive) reactivity has to be reduced by moving cautiously the control rod(s) in direction to the core until the desired reactor power is obtained with a reactor period of $T_S \approx \infty$.

5.3. Power Change

Power change means any intended rise or reduction in the reactor power.

Power rise:

A rise in the power is achieved by drawing out one or several of the control rods with corresponding increase of reactivity. The rods can be pulled out only stepwise and one after another.
The required change in the rod position can be pre-determined with the help of the rod reactivity characteristics. The reactor doubling time should not fall below 30 s.
If the rod reactivity characteristics are not available, the control rods have to be moved in accordance with the display for the doubling time, i.e. before obtaining the value of 30 s, the movement of the control rods has to be stopped.

Power reduction:

A reduction in reactor power is obtained by moving the control rods into the core with corresponding decrease of reactivity to negative values. The value of the resulting negative reactor period is not safety-relevant. Therefore, all three control rods can be moved simultaneously.

5.4. Determination of the Operation Condition

The effects of changes in the positions of the neutron source, of the lower core section and of the control rods on the reactor can be expressed by means of one common physical quantity, i.e. the **reactivity** ρ (t). The reactivity is influenced by these changes in a defined way and can be seen as a global time-dependent reactor control parameter $\rho = \rho(t)$.

The AKR-2 (as most of other reactors, too) has no instruments for direct measurement of the reactivity. Therefore, the reactor operator has to determine the operating condition of the reactor from the behaviour of the reactor power and the reactor period (or doubling time).

The typical behaviour of the reactor power and of the reactor doubling time in dependence on reactivity for start-up and power changes is presented in Fig. 6.

At constant reactor power, the reactor period (or the doubling time as displayed on the control desk) is infinite. Any power increase or power reduction corresponds to a positive reactor period or a negative reactor period, respectively. After any reactivity change, the reactor takes about 1 min in order to show a stable reactor period T_S (time for getting the equilibrium between production of prompt and delayed neutrons).

At very small reactor power levels, the signal for the reactor period may fluctuate slightly because the statistical fluctuations of the neutron density have larger relative amplitudes at low powers than at high powers.

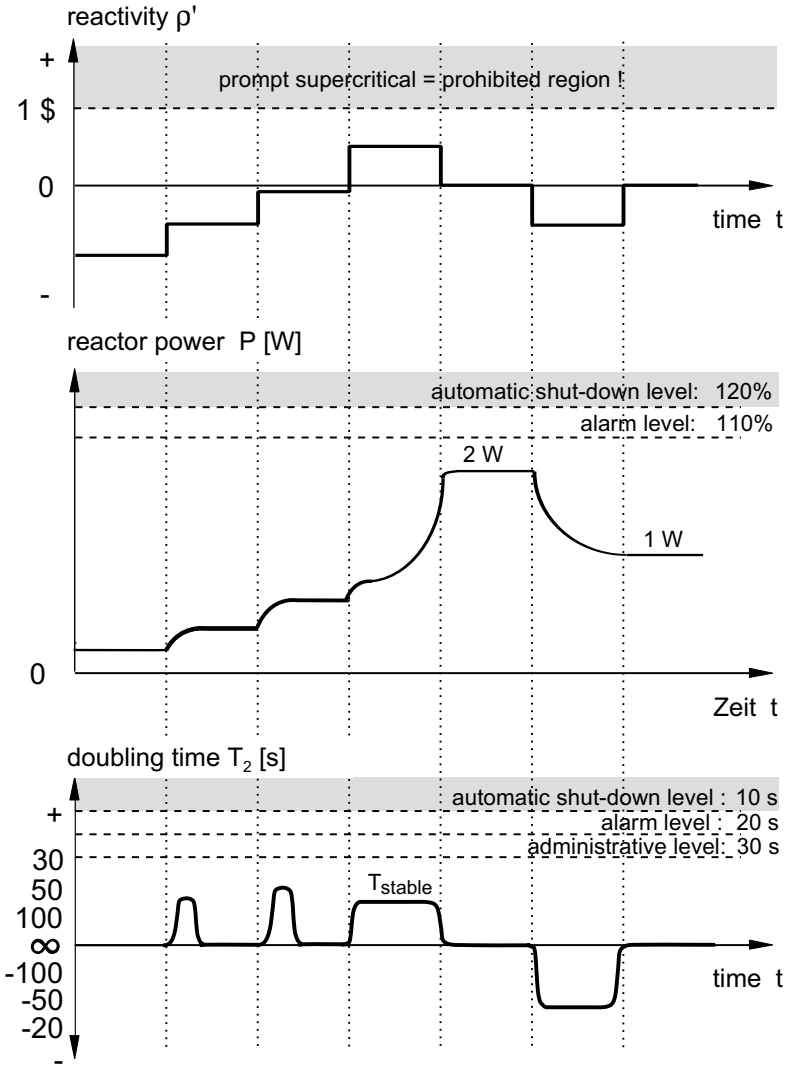


Fig. 6, Behaviour of reactor power and doubling time in dependence on reactivity ρ

6. Instructions Concerning the Protocol

The protocol should contain:
- short description of the experiment
- copy of the record in the operation logbook about the functional check
- log of the start-up procedure
- critical control-rod positions as a function of the power
  - measured values including error ranges
  - analysis and discussion of the control rod position
- measurement of gamma and neutron dose rates in dependence on the reactor power
  - cross section of the reactor hall with the points of the measurements indicated
  - graphical representation of the measured data including error ranges
  - analysis and discussion of the results

7. Index of Relevant Variables

n neutron density (proportional to the neutron flux density Φ, to the number of neutrons N and to the reactor power P)

k multiplication factor

ρ reactivity, $\rho = (k - 1) / k$

ß total fraction of delayed neutrons from fission (for U-235: ß = 0.641 %)

l neutron life time, $l^* = l / k$

λ averaged decay constant for precursors of delayed neutrons

C concentration of the delayed neutron precursors

T reactor period, i.e. time interval for reactor power change for the factor of $e \approx 2.71$

8. Questions to Answer

1. Which are the main components for construction and operation of a thermal reactor and how do they work?

2. What is a zero-power reactor?

3. Which parameter describes the reactor with regard to the criticality and the transient behaviour?

4. What is the purpose of the neutron source for operating a nuclear reactor?

5. What neutron-physical phenomenon enables the safe control of a nuclear reactor? Give a prove using the respective formalism!

6. Which components guarantee the safe operation of a nuclear reactor and how do they work?

7. Why should a reactor have a negative temperature coefficient of the reactivity?

8. What does a reactor operator has to take care for in a reactor start-up and what is the procedure of a start-up?

9. What does the reactor operator have to do in a safety-relevant event (exceptional event)?

References:

/1/ Glasstone, S., Edlund, M.C., The Elements of Nuclear Reactor Theory, New York 1952
/2/ Technical Description and Procedure of Operation for the Reactor Facility AKR-2, TU Dresden, Institute of Power Engineering
/3/ Ackermann, G. (ed.), Betrieb und Instandhaltung von Kernkraftwerken, Leipzig 1982

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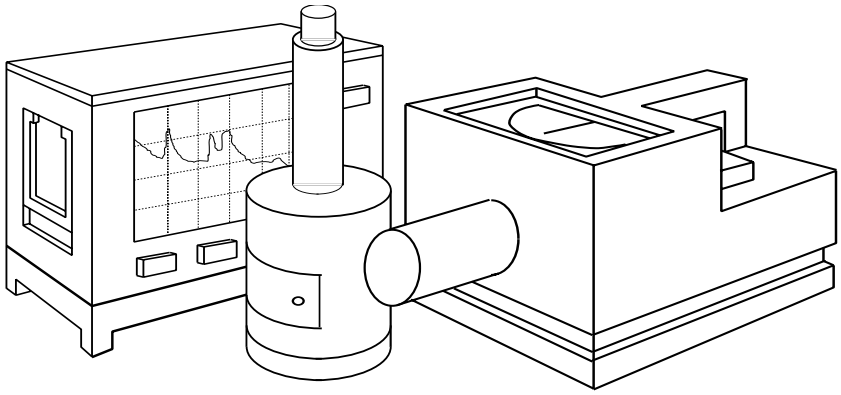
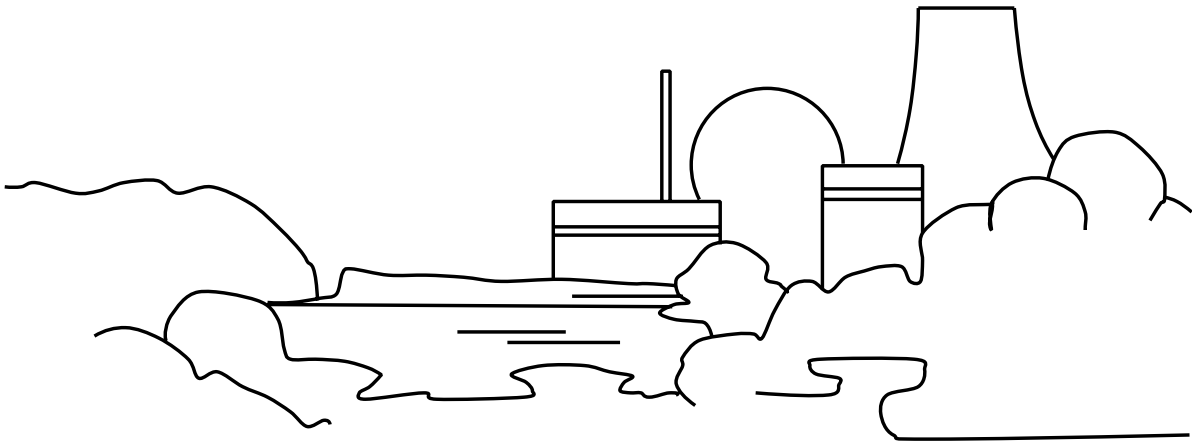


Reactor Training Course

Experiment

"Control Rod Calibration"



Instruction for
Experiment "Control Rod Calibration"

Content:

Figures:

Appendix: Mathematical derivation of the INHOUR equation

(issued: March 2015)

1. Motivation

For the safe operation of a nuclear reactor it is of high importance to know:
- the reactivity characteristics of all control units (mainly the control rods),
- the maximum available positive reactivity reserve (excess reactivity), and
- the shut-down reactivity.

By calibrating a control rod, the influence on the reactivity by inserting a neutron absorber into the reactor core will be investigated quantitatively and the theoretical relation with other quantities of reactor physics will be illustrated.

2. Tasks

1. A control rod of the reactor AKR-2 has to be calibrated in dependence on the rod position by the **method of stable positive reactor period** measurement. Based on measured doubling times of the reactor power the corresponding stable reactor periods and, furthermore, the reactivity equivalents according to the INHOUR equation have to be calculated.
   If the obtained positive reactivity change is compensated by movement of another control rod such that the reactor returns to criticality, it is possible to calibrate a second control rod by the **compensation method** at the same time.

2. Graphical representations of
   - the **differential** control rod characteristics, as well as
   - the **integral** control rod characteristics
   both with respect to the rod position have to be prepared for two control rods of the AKR-2.

3. Including the reactivity characteristic of the third control rod of the AKR-2, which is available at the facility:
   - the **excess reactivity**, and
   - the **shut-down reactivity** (for partial shut-down)
   have to be calculated and discussed.

3. Introduction

The characteristic reactivity values of the control rods (and of all other control units) are essential reactor-physical parameters of a nuclear reactor. They need to be known on account of nuclear safety. Consequently, the reactivity efficiencies of the control rods need to be quantified in the cases of first criticality of a reactor or first criticality after substantial changes in the reactor core (materials and/or geometry).

If the reactivity characteristics of the control rods are known, it is also possible to determine the excess reactivity of a reactor using the positions of its calibrated control rods in the critical condition. Furthermore, the control rod characteristics can be used for easily quantifying the changes in reactivity, which may be induced by various reasons.

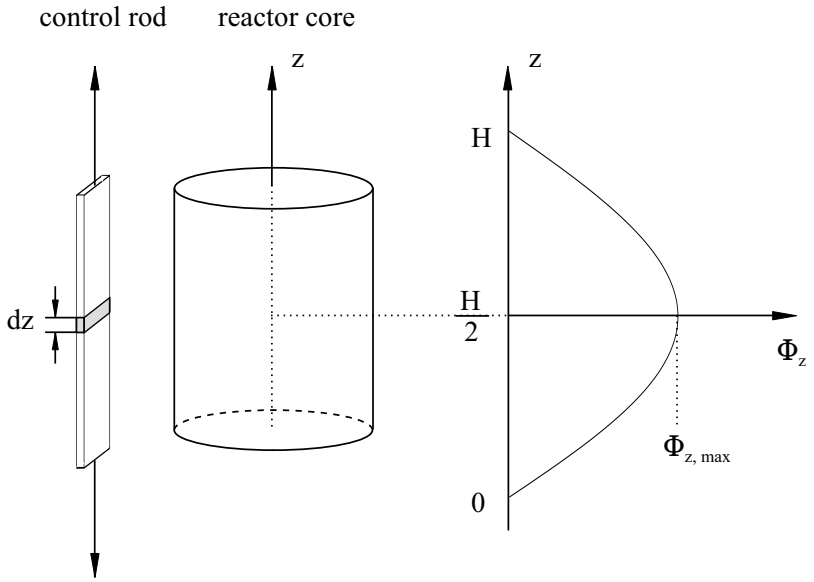


Fig. 1, Cylindrical reactor core with control rod and axial neutron flux density distribution Φ_z

The reactivity characteristic of any control rod can also be determined by theoretical considerations. But the determination of the rod reactivities by calculation is a rather difficult problem, which can be solved with reasonable effort only by applying various approximations. Anyways, calculated reactivity values need to be proven by experiments.

For illustration, consider the arrangement shown in Fig. 1. A neutron absorber moves vertically parallel to the z-axis of the cylindrical core of a thermal reactor. Considering the point of origin to be in the middle plane of the reactor core, the axial component of the neutron flux density can be written as

$$\Phi_z = \Phi_{z,\,max} \cdot \sin\left(\frac{\pi \cdot z}{H}\right) \qquad \textbf{(1)}$$

The parameter H represents the height of the core. By moving a neutron absorber into the core, the number of free neutrons decreases, i.e. the reactivity is lowered. The decrease in reactivity which is caused by a differential rod slice dz being located at the position z is the larger,
- the larger the macroscopic absorption cross section Σ_a,
- the larger the neutron flux density Φ_z at the position z, and
- the longer the slice dz is.

Moreover, the influence of losing free neutrons across the surface of the reactor (leakage) needs to be considered. Those neutrons originating in the center of the reactor core have the maximum probability to remain in the core and to cause fission reactions. On the other hand, the probability for neutron leakage has its maximum in the region of low neutron density near the surface of the core. Consequently, the various neutrons contribute unequally to the neutron balance and hence, they have an unequal influence on the reactivity. Thus, this function of influence, also called **adjoint flux function**, varies in space approximately in same way as the neutron flux density, i.e. both functions are proportional to each other, and the magnitude of reactivity loss

is determined by two effects
- absorption ($\sim \Sigma_a \cdot \Phi_z \cdot dz$)
- space-dependent adjoint flux ($\sim \Phi_z$)

According to probability theory the total probability is the product of the particular contributions. Therefore, the total effect can be written as

$$d\rho \sim \Sigma_a \cdot \Phi_z^2 \cdot dz \qquad \textbf{(2)}$$

By introducing a constant of proportionality C, the equation for the **differential control rod characteristic** results in

$$\frac{d\rho}{dz} = C \cdot \Sigma_a \cdot \phi_z^2 \qquad \textbf{(3)}$$

A control rod which is moved into the reactor core by a finite length $\Delta z = z2 - z1$ changes the reactivity by

$$\Delta\rho = \int_{z_1}^{z_2} d\rho = \int_{z_1}^{z_2} C \cdot \Sigma_a \cdot \phi_z^2 \cdot dz. \qquad \textbf{(4)}$$

Integration over the whole rod length up to the position z gives the **integral control rod characteristic** $\rho(z)$

$$\rho(z) = \int_0^z C \cdot \Sigma_a \cdot \Phi_z^2 \cdot dz. \qquad \textbf{(5)}$$

Applying the neutron flux distribution shown in Fig. 1 into equation (5) gives

$$\rho(z) = C \cdot \Sigma_a \cdot \Phi_{z,\,\max}^2 \cdot \frac{H}{\pi} \left[\frac{\pi \cdot z}{2 \cdot H} - \frac{1}{4} \sin\left(\frac{2 \cdot \pi \cdot z}{H} \right) \right] \qquad \textbf{(6)}$$

and integration up to the full length 0 ... H gives the integral reactivity value of the complete rod

$$\rho_{max} = C \cdot \Sigma_a \cdot \Phi_{z,\,\max}^2 \cdot \frac{H}{2} \qquad \textbf{(7)}$$

After normalising the reactivity to the maximum value ρ_{max}, the integral control rod characteristic gets the shape which is shown in Fig. 2.

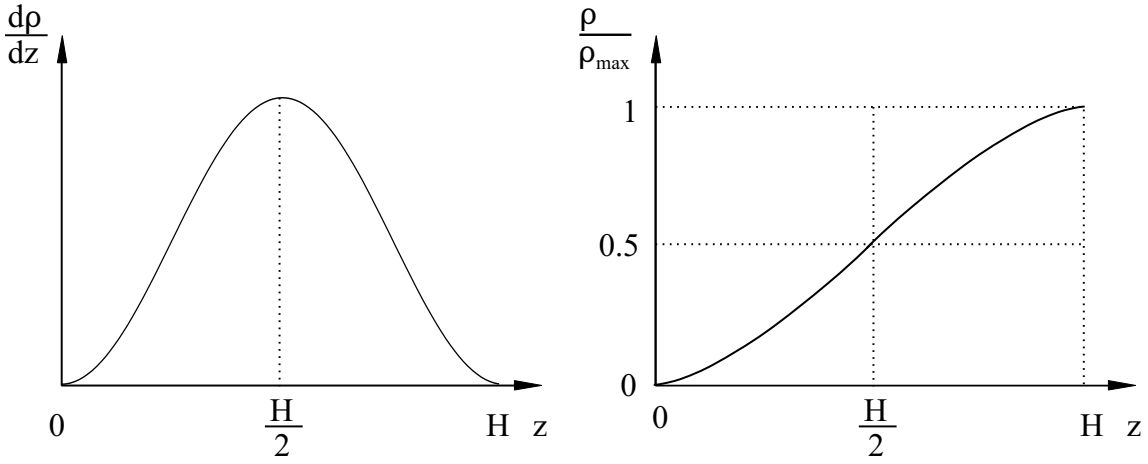


Fig. 2, Control rod characteristics, differential (left) and integral (right, normalised)

The control rods are made of materials having a large macroscopic absorption cross section Σ_a. For practical applications in nuclear reactors, also the mechanical and the chemical properties of these materials have to be considered. In addition, the sensitivity with respect to radiation exposure and economical aspects decide about applicability of a certain material. Most common materials for the control rods of thermal reactors are boron and cadmium.

The control rod calibration is the experimental determination of the function $\frac{d\rho}{dz}(z)$ and, subsequently, to calculate the function $\rho(z)$.

4. Methods for Reactivity Determination

The reactivity is always determined via the time dependence of the reactor power, which follows from the emission rates of prompt and delayed neutrons. The time dependent behaviour of the reactor power after a sudden (positive) change in the reactivity is shown in Fig. 3.

The production rate of prompt neutrons immediately reacts on the reactivity change (prompt jump). In contrast, the production rate of the delayed neutrons initially remains constant. Since by the prompt neutrons alone the reactor is subcritical, the system remains in a quasi-stationary condition until the production rate of the delayed neutrons starts to increase. This is about 100 ms after the reactivity change.

Then, the production rate of delayed neutrons noticeably starts to increase and after about 100 s, an equilibrium condition of the production of prompt and delayed neutrons has been established. The change in reactor power is then determined by the so-called stable reactor period T_s.

Negative reactivity jumps give analogous system answers but a decrease in power.

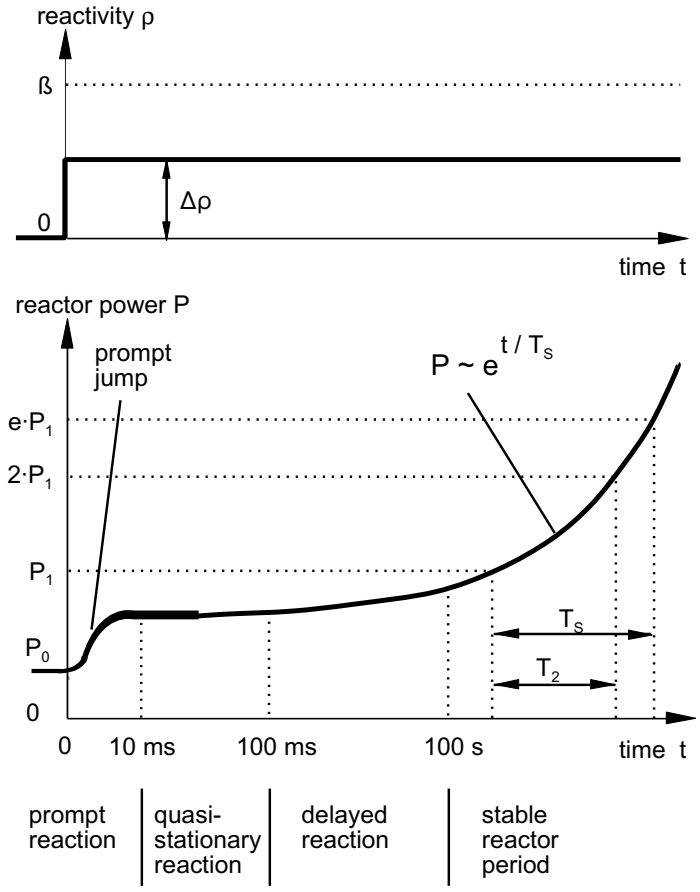


Fig. 3, Time behaviour of the reactor power after a positive reactivity jump ($0 < \rho << \beta$ at time $t = 0$)

The calibration of the control rods can be done applying several methods, which all use various specific effects:

Rod-Drop Method
Before and after dropping a control rod, the time behaviour of the reactor power has to be registered. The resulting change in reactivity due to the rod drop can be determined from the magnitude of the so-called prompt jump, which marks the decrease in power.

Periodic Power Modulation
Periodic changes in the position of the control rods induce periodic power changes. The corresponding reactivity change can be determined from the amplitudes of these power oscillations.

Method of Inverse Kinetics
This method bases on mathematically modelling the time-dependent processes in the reactor. In the model, the reactivity is continuously regulated such that the actual reactor power meets the calculated reactor power in every single instant. This method is used by so-called reactivity meters that display the reactivity permanently. The technical effort for such reactivity meters is rather high.

Positive period method (measurement of stable reactor period T_s)
This is a conventional method for the determination of the reactivity. It will be applied also in the given experiment.
The relation between the measured stable reactor period and the unknown value of reactivity is given by the so-called INHOUR equation (derived from "inverse hour"). This equation follows from the reactor-kinetic equations (for a detailed derivation see the appendix)

$$\frac{dn}{dt} = \frac{\rho - \beta}{l^*} \cdot n + \sum_{i=1}^{6} \lambda_i \cdot C_i + S$$
$$\frac{dC_i}{dt} = \frac{\beta_i}{l^*} \cdot n - \lambda_i \cdot C_i \qquad (i = 1, \ldots, 6) \qquad \textbf{(8)}$$

with:
l^* = average effective life-time of prompt neutrons,
ρ = reactivity,
β_i = absolute fraction of the i-th group of delayed neutrons,
C_i = concentration of the delayed-neutron precursors of the i-th group,
λ_i = decay constant of the delayed-neutron precursors of the i-th group, and
S = source strength of the external neutron source in the reactor.

This system of ordinary differential equations has 7 solutions corresponding to the 7 eigenvalues ω_i. The system can be solved using the ansatz

$$n_i(t) = n_{i,0} \cdot e^{\omega_i \cdot t} \qquad resp. \qquad C_i(t) = C_{i,0} \cdot e^{\omega_i \cdot t} \qquad \textbf{(9)}$$

As a consequence of the linearity of the system the neutron density n(t) is a linear combination of the 7 solutions, which have the form given in equation (9):

$$n(t) = n_0 \cdot e^{\omega_0 \cdot t} + n_1 \cdot e^{\omega_1 \cdot t} + \ldots + n_6 \cdot e^{\omega_6 \cdot t} \qquad \textbf{(10)}$$

Its solutions ω_i are given schematically in Fig. 4.

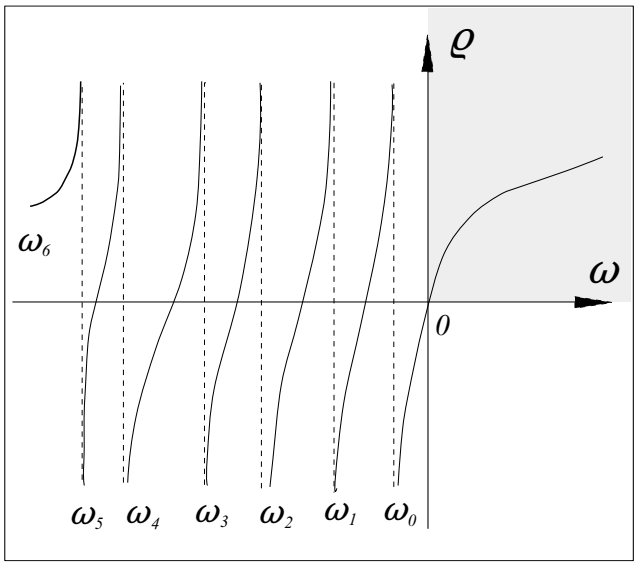


Fig. 4, Schematic presentation of the solutions ω_i of the INHOUR equation

Having a positive amount of reactivity ($\rho > 0$) only one of the 7 values ω_i is positive. Consequently, 6 of the 7 contributions in equation (10) decay within a short period of time and vanish.

If finally the remaining terms

$$n(t) = n_0 \cdot e^{\omega \cdot t} \qquad resp. \qquad C_i(t) = C_{i,\,0} \cdot e^{\omega \cdot t} \tag{11}$$

are inserted into the reactor-kinetic equations (8) and, additionally, respecting the relation $\omega = 1 / T_S$, then the **INHOUR equation** follows as:

$$\rho = \frac{l^*}{T_S} + \sum_{i=1}^{6} \frac{\beta_i}{1 + \lambda_i \cdot T_S} \tag{12}$$

Considering only small positive reactivity values, i.e. for large stable reactor periods, in equation (12), the denominator of the sum can be approximated by $\lambda_i \cdot T_S$ (since $1 << \lambda_i \cdot T_S$) and the first term l^*/T_S can be neglected against the sum. Thus, equation (12) can be approximated by

$$\rho \approx \frac{1}{T_S} \cdot \sum_{i=1}^{6} \frac{\beta_i}{\lambda_i} \tag{13}$$

The reactivity ρ may be normalised to the fraction of delayed neutrons ß. This is reasonable, because the quotient ρ / ß occurs in any investigation concerning the kinetic reactor behaviour. Although this quantity $\rho' = \rho$ / ß carries no physical unit, it gets attached by the artificial unit Dollar ($) with 1 $ = 100 ¢ to distinguish it from the non-normalised quantity ρ. In this representation, the INHOUR equation has the form

$$\rho' = \frac{\rho}{\beta} = \frac{l^* / \beta}{T_S} + \sum_{i=1}^{6} \frac{a_i}{1 + \lambda_i \cdot T_S} \qquad \textbf{(14)}$$

with $a_i = \beta_i / \beta$ being the fraction of the delayed neutrons of group i relative to the total fraction of all delayed neutron groups. Some group specific information is summarised in Tab. 1.

At the AKR, the parameter l^* / β has the value $l^* / \beta = 0.0051$ s.

i	λ_i [s^{-1}]	$a_i = \beta_i / \beta$
1	0.0124	0.033
2	0.0305	0.219
3	0.111	0.196
4	0.301	0.395
5	1.14	0.115
6	3.01	0.042

Tab. 1, Data of delayed neutron groups for the application in the INHOUR equation

5. Procedure of the Experiment

The reactor has to be set to a critical condition at low power (about 0.3 W), which has to be done with leaving one rod completely in the core (position 0), pulling out another rod completely (position 4000), and adjusting the critical condition with the third rod. The neutron source has to be removed from the reactor.

Then, the rod which has to be calibrated (that rod being in position 0) is drawn out by a certain defined length (e.g. position 800). Consequently, the corresponding change in reactivity $\Delta\rho$ causes a supercritical condition with increasing power. Initially, all solutions of the INHOUR equation contribute to the increase. Only after the higher solutions have decayed, i.e. after equilibrium of delayed and prompt neutrons has established (about 60 … 100 s after moving the rod), the stable reactor period can be measured. It is that time interval which the reactor takes for a power increase by a factor of e = 2.71. In practical cases, it is more convenient to measure the doubling time T_2, instead. This quantity is related to the stable reactor period by

$$T_S = \frac{T_2}{\ln 2} \qquad \textbf{(15)}$$

Inserting the resulting value of the stable reactor period T_S into the INHOUR equation (12) or (14), gives the corresponding value of reactivity.

By application of a trick even a second control rod can be calibrated in a single experiment at the same time (compensation method). For this purpose, after measuring the doubling time and calculating the positive reactivity, a second control rod (that is totally pulled out at the beginning in position 4000) is used to re-establish the critical condition. By this way, the same absolute amount of reactivity is fed to the reactor as before, but with negative sign. The corresponding change in the position of the second control rod can be determined from the display on the control panel.

After adjusting the critical reactor condition at about 0.3 W reactor power, again, the first rod can be drawn out of the core by a defined length for the second time. According to the procedure explained above, the doubling time has to be measured and afterwards, the positive amount of reactivity due to the movement of the first rod has to be compensated by moving in the second rod, again.

This procedure has to be repeated until the first rod has reached its upper end position (position 4000). The second rod is supposed to have the position 0 after the compensation assuming the reactivity values of both rods are identical.

The calculation of the reactivity values according to equations (12) or (14) is time consuming, since for each value of the stable reactor period, all 6 groups of delayed neutrons contribute to the sum in the equation. For this reason, preferably a programmable calculator or a computer code should be used to evaluate the measured doubling times. Such a computer program is provided at the working place.

Fig. 5 shows the relation between doubling time (or stable reactor period, respectively) and the reactivity according to the INHOUR equation using the parameter values from Tab. 1.

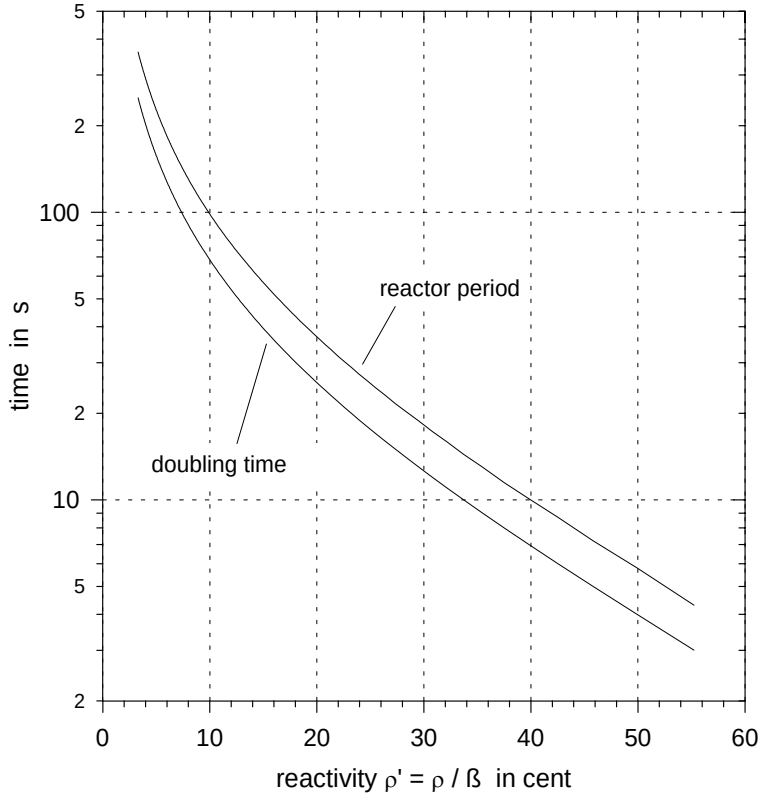


Fig. 5, Relation between doubling time or reactor period, respectively, and reactivity according to the INHOUR equation

The determination of the reactivity by measuring the stable reactor period is easy to perform at least for determination of small reactivity changes, since only a relation between the reactivity and the stable reactor period is needed. Further physical quantities describing the reactor are unnecessary to know at this point. But it must be mentioned, that the application of the IN-HOUR equation is restricted to small reactivity range (0 … 20 ¢) only, because higher reactivity values would cause too large rates of power increase and a too strong influence of drift or noise effects. Moreover, this method is rather time-consuming since for the determination of the characteristic of a complete rod, several subsequent measurements of the reactor period has to be carried out.

6. Calculation of the Total Reactivity of the Control Rods and Determination of the Excess Reactivity and of the Shut-Down Reactivity

The **total reactivity value** of a control rod is given by the the integral control rod characteristic determined over the complete rod length (Fig. 2).

The **excess reactivity** is that amount of positive reactivity which results from drawing out all control rods each to its upper end position, starting from a critical reactor condition. Based on the integral control rod characteristic of each control rod i (at the AKR is i = 1, 2, 3), it is determined which positive reactivity $\rho_{i,\,excess}$ can be still fed into the reactor by complete pulling out of all control rods starting from their positions in critical reactor condition. For the determination of the total excess reactivity, the individual critical positions of the rods relative to each other are irrelevant, since the total excess reactivity is the sum of the excess reactivities of all rods

$$\rho_{excess} = \rho_{1,excess} + \rho_{2,excess} + \rho_{3,excess}$$

Fig. 6 illustrates the idea of calculating the excess reactivity from the measured control rod characteristics.

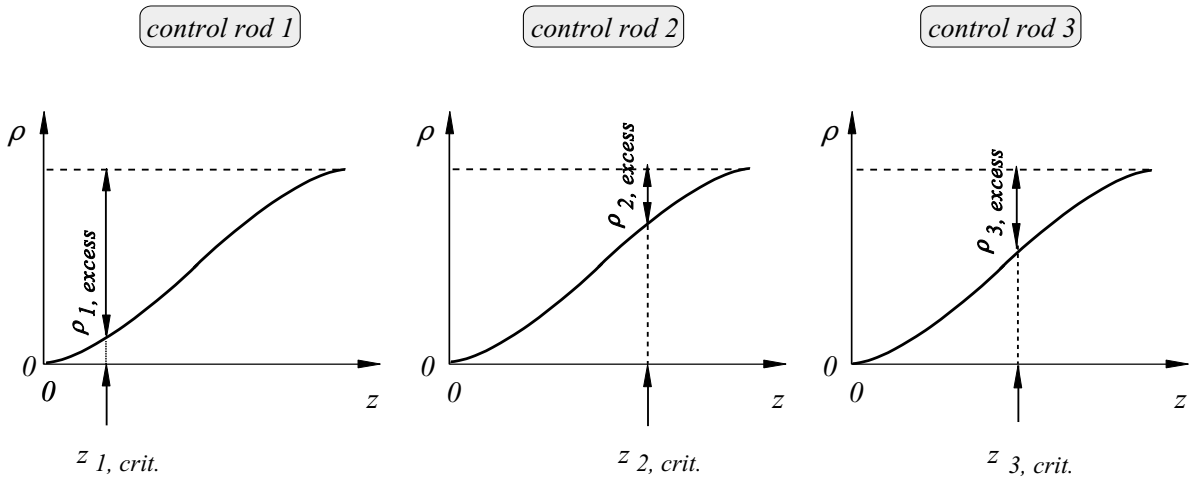


Fig. 6, Determinationof the excess reactivity based on measured control rod characteristics

In the given reactor experiment at AKR-2, usually only two of the three control rods are calibrated. Since the calculation of the total excess reactivity requires the characteristic of the third rod, too, this characteristic will be approximated by the average of the integral rod characteristics of the other two rods. Such an assumption is justified, because all three rods are designed identically and positioned symmetrically in the reactor.

The starting point for the determination of the excess reactivity is always a critical reactor condition - no matter what particular positions of the rods relative to another mark the critical condition. Conveniently, that critical rod configuration should be chosen which marks the critical condition at the start of the experiment. Doing so, the entire reactivity of the first rod contributes to the excess reactivity (full rod length is available) while for the second rod no contribution has to be taken into account (rod is already in its upper end position). Only the excess reactivity of the third rod has to be determined by interpolation of the corresponding characteristic.

The total excess reactivity is an important criterion for the nuclear safety of a nuclear reactor. If the total excess reactivity ρ_{excess} of a reactor does not exceed the fraction of delayed neutrons ß, i.e. for keeping $\rho_{excess} < ß$ (or identically $\rho'_{excess} < 1$ \$), such a reactor can never reach a prompt-supercritical condition - neither due to technical failure nor due to wrong action of the reactor operator. At the AKR, the criterion $\rho_{excess} < ß$ is fulfilled and defined in the operation licence of the facility. An important result of the given experiment is to prove this condition!

In contrast to the previously discussed quantity, the **shut-down reactivity** (of the partially shut-down reactor) is that amount of negative reactivity which results from dropping all control rods to their lower end position 0, starting again from critical condition. The determination of the shut-down reactivity goes along with that of the excess reactivity but with opposite direction of rod moving. In case the excess reactivity is already known, the shut-down reactivity can be calculated as the difference of the total reactivity and the excess reactivity

$$\rho_{shut-down} = \rho_{total} - \rho_{excesss}$$

7. Comments on the Protocol

- short description of the experiment and its procedure
- table containing all measured values including measurement uncertainties
- numerical analysis and discussion of the errors
- graphical representation of the (differential and integral) control rod characteristics
- determination of the excess reactivity and the shut-down reactivity (for partial shut-down)

8. Questions to Answer

1. Explain the derivation of the INHOUR equation and discuss its solutions for the cases:
   a) $\rho < 0$ b) $\rho = 0$ c) $\rho > 0$!

2. What is the time behaviour of the reactor power for a negative reactivity jump?

3. What is the so-called stable reactor period in the context of the time behaviour of a reactor?

4. What is the excess reactivity and how it can be determined from the control rod characteristics? What is the maximum value of the excess reactivity such that the reactor would be safely controllable?

5. What is the idea of the calibration of a control rod simultaneously to the calibration of another control rod (compensation method)? How to determine the end value of the reactivity of the second rod, in case the total amounts of reactivity of the rods are unequal?

6. What is the benefit of the control rod calibration?

7. What are the physical and technical requirements for the material properties of a control rod?

Appendix: Derivation of the INHOUR equation

Fundamentals are the point-kinetic equations:

$$\frac{dn}{dt} = \frac{\varrho - ß}{l^*} \cdot n + \sum_{i=1}^{6} \lambda_i \cdot C_i \qquad \text{(A1)}$$

$$\frac{dC_i}{dt} = \frac{ß_i}{l^*} \cdot n - \lambda_i \cdot C_i \qquad \text{(A2)}$$

Solutions correspond to:

$$n(t) = n_0 \cdot e^{\omega \cdot t} \quad \Rightarrow \quad \frac{dn}{dt} = n_0 \cdot e^{\omega \cdot t} \cdot \omega = n \cdot \omega \qquad \text{(A3)}$$

$$C_i(t) = C_{0,i} \cdot e^{\omega \cdot t} \quad \Rightarrow \quad \frac{dC_i}{dt} = C_{0,i} \cdot e^{\omega \cdot t} \cdot \omega = C_i \cdot \omega \qquad \text{(A4)}$$

from (A2) follows by insertion of (A4):

$$C_i \cdot \omega = \frac{ß_i}{l^*} \cdot n - \lambda_i \cdot C_i$$

$$C_i \, (\omega + \lambda_i) = \frac{ß_i}{l^*} \cdot n$$

$$C_i = \frac{ß_i \cdot n}{l^* \cdot (\omega + \lambda_i)}$$

respectively

$$C_i \cdot \lambda_i = \frac{\lambda_i \cdot ß_i \cdot n}{l^* \cdot (\omega + \lambda_i)} \qquad \text{(A5)}$$

insertion of (A5) and (A3) in (A1) results in:

$$n \cdot \omega = \frac{\varrho - ß}{l^*} \cdot n + \sum_{i=1}^{6} \frac{\lambda_i \cdot ß_i \cdot n}{l^* \cdot (\omega + \lambda_i)}$$

by multiplication of both sides by l*/n follows:

$$l^* \cdot \omega = \varrho - ß + \sum_{i=1}^{6} \frac{\lambda_i \cdot ß_i}{\omega + \lambda_i}$$

and resolved to ρ:

$$\varrho = l^* \cdot \omega + ß - \sum_{i=1}^{6} \frac{\lambda_i \cdot ß_i}{\omega + \lambda_i} \qquad \text{with} \quad ß = \sum_{i=1}^{6} ß_i \quad (!)$$

$$= l^* \cdot \omega + \sum_{i=1}^{6} \left(ß_i - \frac{\lambda_i \cdot ß_i}{\omega + \lambda_i} \right)$$

$$= l^* \cdot \omega + \sum_{i=1}^{6} ß_i \cdot \left(1 - \frac{\lambda_i}{\omega + \lambda_i} \right)$$

$$= l^* \cdot \omega + \sum_{i=1}^{6} ß_i \cdot \left(\frac{\omega + \lambda_i - \lambda_i}{\omega + \lambda_i} \right)$$

$$= l^* \cdot \omega + \sum_{i=1}^{6} \frac{ß_i \cdot \omega}{\omega + \lambda_i}$$

$$= l^* \cdot \omega + \sum_{i=1}^{6} \frac{\omega}{\omega} \cdot \left(\frac{ß_i}{1 + \frac{\lambda_i}{\omega}} \right)$$

$$= l^* \cdot \omega + \sum_{i=1}^{6} \frac{ß_i}{1 + \frac{\lambda_i}{\omega}}$$

If only such positive reactivity addition is taken into account which cause a stable reactor period (i.e. $0 < \rho < ß$), it can be seen from equation (A3) $n(t) = n_0 \cdot e^{\omega \cdot t}$ that in this case ω must be $\omega = 1/T_s$.

$$\varrho = \frac{l^*}{T_s} + \sum_{i=1}^{6} \frac{ß_i}{1 + \lambda_i \cdot T_s}$$

INHOUR equation

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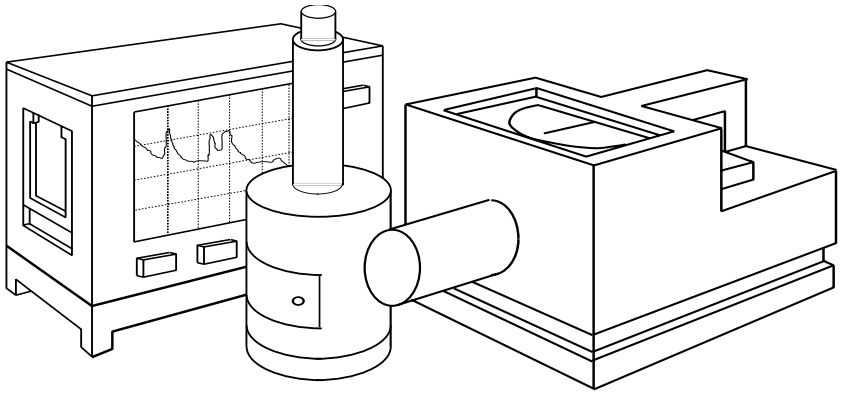
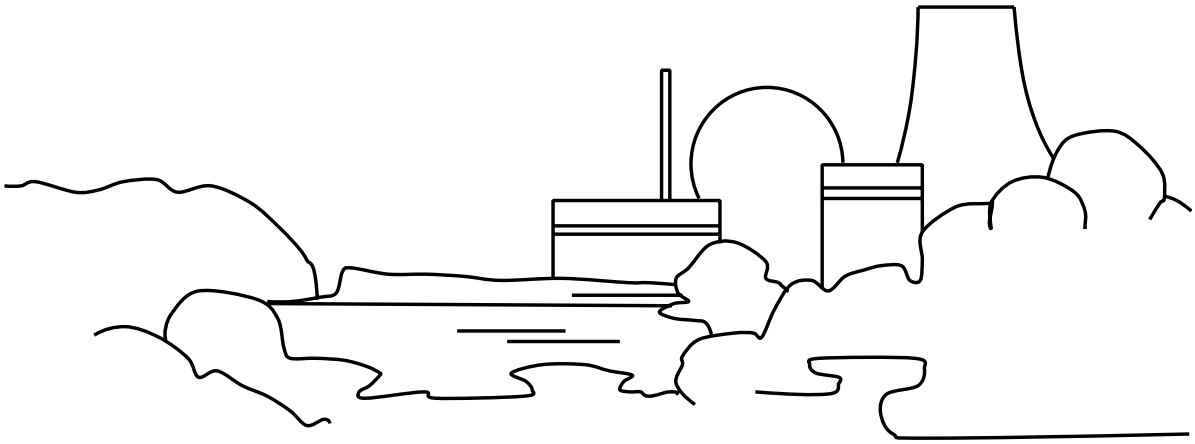


Reactor Training Course

Experiment

"Critical Experiment"



Instruction for
Experiment "Critical Experiment"

Content:

Figures:

Appendix:
- Protocol Sheet of the Critical Experiment
- Template for diagram

(issued: March 2015)

1. Motivation

The critical experiment is a procedure for checking experimentally the correct core loading of a nuclear reactor with nuclear fuel. It is carried out in case of the initial commissioning of a nuclear reactor or in case after modification of the reactor core (geometry and/or material changes) if the critical mass and the critical control rod positions are only known from calculations.

The critical experiment ensures
- that, on the one hand, the reactor is actually loaded with sufficient fuel for obtaining criticality and
- that, on the other hand, the reactor core does not contain too much nuclear fuel to avoid that the permitted excess reactivity would be exceeded or that the reactor could even become prompt supercritical.

The reactor must be controllable by the control rods at any time, i.e. control rod movements must guarantee that both, multiplication factors $k>1$ and (for safety reasons even more important) $k<1$ can be adjusted.

For example, it may never happen that due to a too high fuel loading the AKR becomes critical or even supercritical already while the core sections still approach one to another, because in that case, control by the control rods would be impossible.

The critical experiment requires profound professional knowledge regarding the physical processes in a nuclear reactor and also a high level of responsibility of the operating staff. The critical experiment needs to be performed with special caution and care.

The training experiment aims at conveying the **measurement and analysis methods** for a critical experiment which enable
- an always safe approach to criticality and
- the reliable predetermination of the critical reactor parameters.

At the AKR, a critical experiment can be carried out in two ways:
- enlarging the amount of fuel in the reactor core by stepwise adding fuel-element plates or
- stepwise approach of the core sections to each other with having a fixed fuel loading in each of the sections.

For the initial commissioning of the AKR, the critical mass was adjusted by stepwise adding of fuel plates.

Because the manipulation of nuclear fuel requires a lot of effort and additional comprehensive precautions and prescriptions and is very time consuming, in the training procedure the critical experiment will be carried out by stepwise approach of the lower core section to the upper one.

2. Tasks

1. From the increase of the neutron density n which is caused by the stepwise approach of the lower core section of the AKR to the upper one, the distance x_{crit} that corresponds to criticality has to be determined by extrapolation.

2. From the measured neutron count rates N for the neutron density n, determine and plot the relation between the position x of the lower core section and
   - the multiplication factor $k(x)$,
   - the subcritical amplification $M(x)$, and
   - the reactivity ρ(x).

3. General Remarks

For every new-build reactor facility or after considerable modifications of its components, those parameters with a direct influence on the criticality have to be experimentally determined. In principle, this concerns all those quantities z_i which affect the neutron balance and correspondingly the criticality of the system:

$$k = \mathrm{f}\,(z_1, \ldots, z_n)$$

The nuclear fuel itself as well as the moderator, the reflector, the control rods, the neutron detectors, the experimental facilities, and all other materials being located in the core belong to those quantities.

The level of influence is given by the material characteristics as well as by the installed masses and the geometric arrangement.

For the initial commissioning of a reactor, the starting point usually is an almost completed system with only one free parameter z_i. In the so-called "critical experiment", the system is led to criticality by variation of this single free parameter z_i, i.e. for $z_i \rightarrow z_{krit}$ follows $k \rightarrow 1$ (Fig. 1).

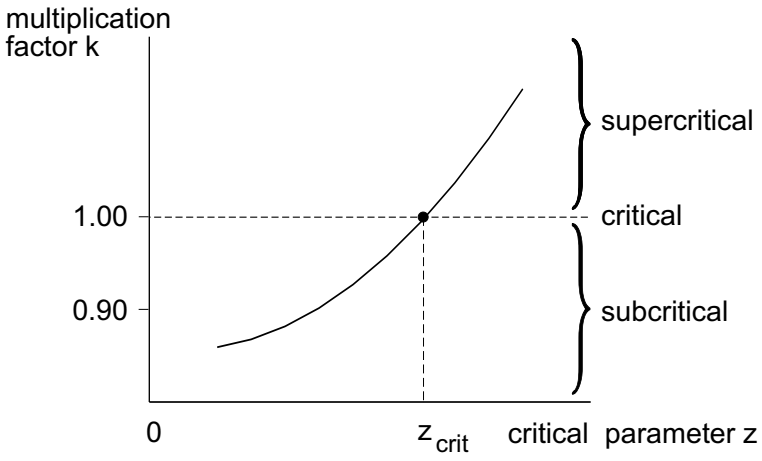


Fig. 1, Dependence of the multiplication factor k on the critical parameter z of the critical experiment

For achieving criticality of a reactor for the first time, mostly the fuel mass ($z = m_{fuel}$) is chosen as the free parameter in the critical experiment. Starting from a subcritical loading, the amount of fuel is enlarged stepwise until the reactor is critical or supercritical.

Another feasible procedure is to approach cautiously two subcritical masses which would give together a critical mass. In this case, the distance between the two subcritical masses represents the critical parameter. This alternative is applied in the given experiment (the critical parameter z_i is the position of the lower core section).

4. Theoretical Background

At the beginning of the experiment, the lower core section of the AKR is in its lowest position, the reactor is reliably subcritical (x = 0). The arrangement is shown schematically in Fig. 2.

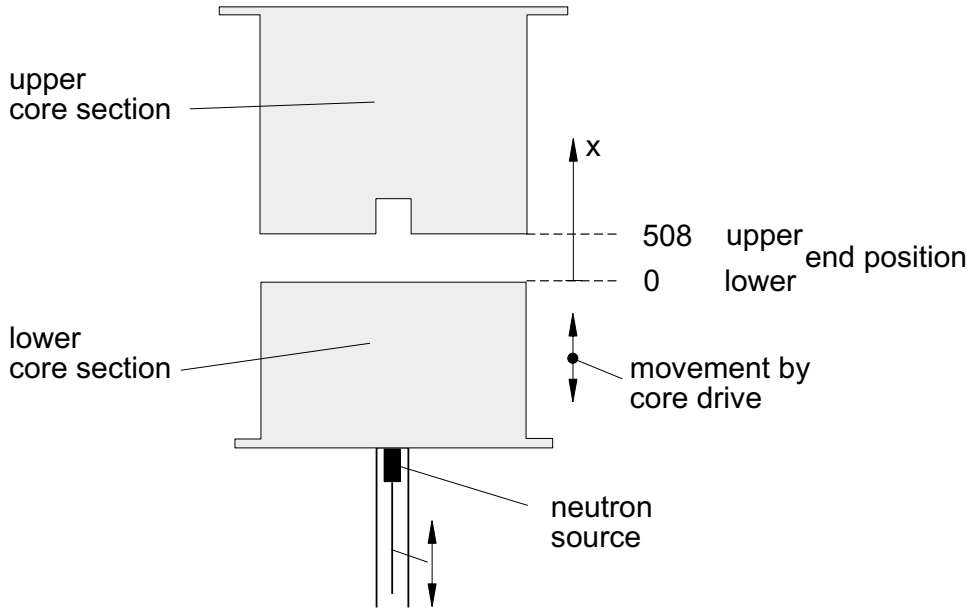


Fig. 2, Schematic arrangement of the experiment

A subcritical reactor contains almost no free neutrons. By inserting the neutron source, which has a source strength *S* and emits neutrons into the reactor core, the neutron density is increased to such level that
- a sufficient count rate at the neutron detectors is available, and
- therefore statistically safe measurements of the reactor state are possible.

For determining the reactor state, the neutron density $n(t\rightarrow\infty)$ is used, which is the result of the **subcritical amplification of the neutron source**. Theoretically, it is reached only after infinitely long time. The neutron density *n* is proportional to the neutron flux density Φ in the reactor,

proportional to the reactor power P as well as proportional to the count rates N measured by the neutron detectors.

The asymptotic neutron density n after infinitely long time is

$$\begin{aligned} n_{(t\to\infty)}(\sim\Phi,\sim P,\sim N) &= S\cdot l + S\cdot l\cdot k + (S\cdot l\cdot k)\cdot k + \ldots \\ &= S\cdot l\ (1 + k + k^2 + k^3 + \ldots\ ) \\ &= S\cdot l\cdot\frac{1}{1-k} \quad \text{(limit value of the geometric series with k<1)} \end{aligned} \tag{1}$$

with S source strength of the neutron source
l neutron generation time
k multiplication factor of the (subcritical) reactor

The time dependence for approaching the asymptotic neutron density is

$$n(t) = \frac{S\cdot l}{1-k}\ (1 - e^{-\frac{(1-k)}{l}\cdot t}) \tag{2}$$

After infinitely long time, equation (2) results in equation (1). As it can be seen from equation (2), the nearer the system is to the critical state ($k \to 1$), the slower the approach to the asymptotic value continues n ($t\to\infty$).

At the beginning of the experiment and for $t < 0$, the distance between the two core sections be x_0 and the entire arrangement has the multiplication factor $k(x_0) = k_0$.

At $t = 0$, the distance between the core sections is reduced by the value Δx. The resulting multiplication factor corresponding to the new position x shall be $k(x)$. The neutron density n increases according to equation (2). After a sufficient time period ($t \to \infty$), the neutron density has virtually reached its asymptotic value n ($t\to\infty$) (equation (3)):

$$n(x, t\to\infty) = S\cdot l\cdot\frac{1}{1-k(x)} \tag{3}$$

The factor

$$M(x) = \frac{1}{1-k(x)} \tag{4}$$

is also known as the **subcritical amplification**.

Fig. 3 shows the relation between the discussed quantities *x, k(x), and n(x,t)*.

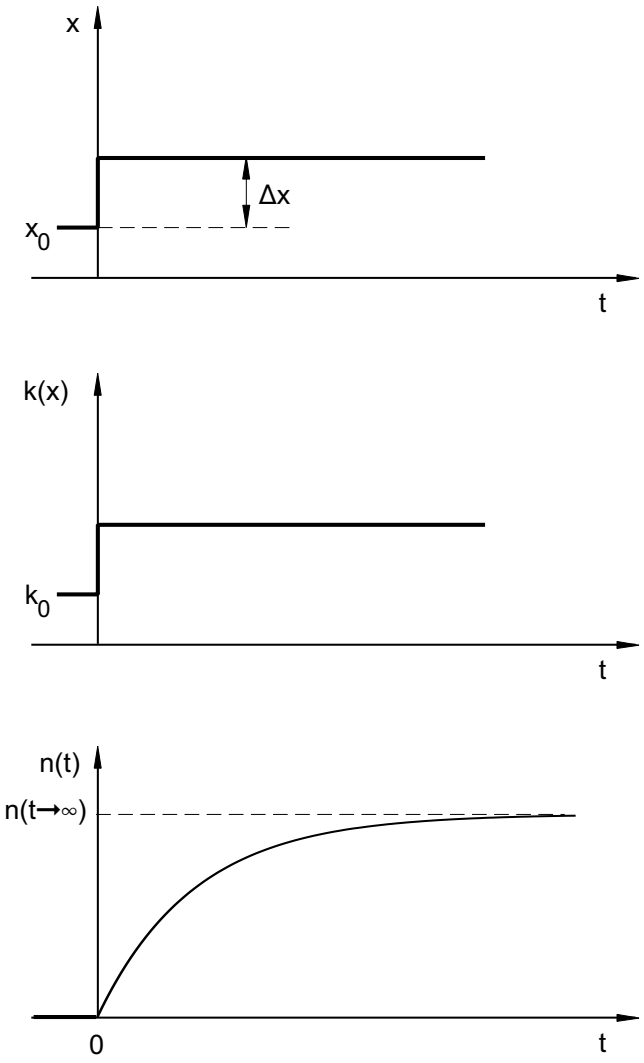


Fig. 3, Neutron density over time with approaching to the asymptotic value for reduction of the distance between the two core sections

In the experiment, the distance between the core sections is reduced stepwise.

If waiting until the asymptotic value is nearly reached, then for ($k \rightarrow 1$) the subcritical amplification and the neutron density n ($t \rightarrow \infty$) as well goes to infinity:

$$\lim_{k \rightarrow 1} n(t \rightarrow \infty) = \infty \qquad \textbf{(5)}$$

Hence, for ($k \rightarrow 1$), the reciprocal value of the neutron density $1/n(t \rightarrow \infty)$ tends to zero:

$$\lim_{k \rightarrow 1} \left(1/n(t \rightarrow \infty)\right) = 0 \qquad \textbf{(6)}$$

Since the pulse rate N of a neutron detector of the reactor instrumentation is directly proportional to the neutron density n, it can also be written:

$$N \sim n \qquad \text{resp.} \qquad N = C \cdot n \qquad \textbf{(7)}$$

The constant C depends on the positioning and on the sensitivity of the neutron detector.

After approaching the asymptotic value $n(t \rightarrow \infty)$, the **inverse pulse rate** of a channel $1/N$ is:

$$1/N(x) = \frac{1}{C \cdot n(t \rightarrow \infty)} = \frac{1 - k(x)}{C \cdot S \cdot l} \qquad \textbf{(8)}$$

If plotting $1/N(x)$ over x, the point of intersection of the resulting curve with the abscissa axis gives the position x_{crit} of the lower core section that marks exact criticality (Fig. 4).

$$1/N(x_{krit}) = 0 \qquad \textbf{(9)}$$

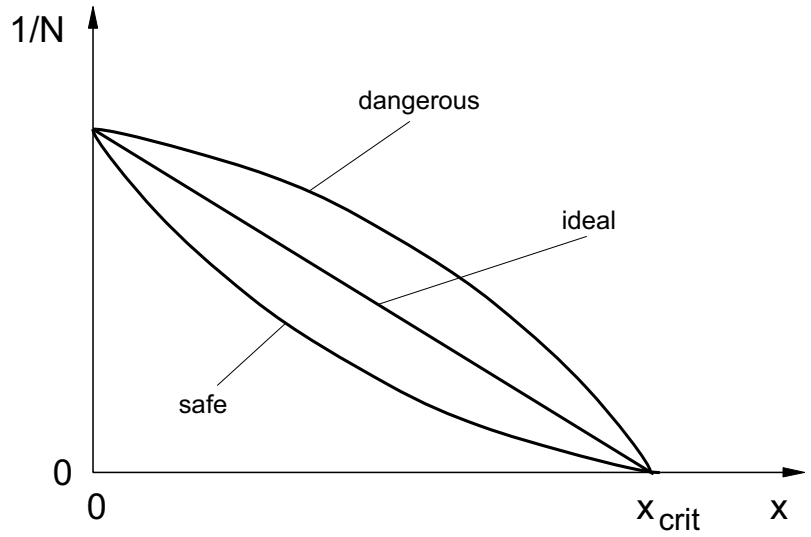


Fig. 4, Reciprocal pulse rate $1/N$ over position x of the lower core section for the determination of x_{crit} by extrapolation

In principle, three shapes of the curve $1/N(x)$ are possible:
- ideal shape (linear),
- dangerous (concave with respect to abscissa), and
- safe (convex with respect to abscissa).

The shape of the measured curve depends essentially on the positions of the neutron source and the neutron detector.

For the stepwise movement of the lower core section by steps of the distance Δx, the neutron pulse rate N increases by an amount ΔN:

$$\Delta N = N(x+\Delta x) - N(x) = C \cdot S \cdot l \frac{k(x+\Delta x) - k(x)}{(1-k(x+\Delta x)) \cdot (1-k(x))} \quad \textbf{(10)}$$

Division by $N(x)$ eliminates the unknown constants and gives

$$k(x+\Delta x) = 1 + \frac{N(x)}{N(x+\Delta x)} (k(x) - 1) \quad \textbf{(11)}$$

Thus, using the known previous value of the multiplication factor the next respective value can be calculated iteratively. The initial value $k_0 = k(x_0)$ must be known for this procedure. At the AKR, the multiplication factor for completely separated core sections is $k_0 = 0.945$.

If $k(x)$ is known then also the subcritical amplification M as a function of x can be calculated:

$$M(x) = \frac{1}{1 - k(x)} \quad \textbf{(12)}$$

The reactivity ρ is defined on basis of the multiplication factor k. From known values $k(x)$, ρ can be calculated as follows

$$\rho(x) = \frac{k(x) - 1}{k(x)} \quad \textbf{(13)}$$

The change in reactivity $\Delta\rho(x)$ is given by

$$\Delta\rho(x) = \left(\frac{d\rho}{dx}\right) \cdot \Delta x \quad \textbf{(14)}$$

with

$$\left(\frac{d\rho}{dx}\right)_x = -\frac{\rho(x)}{N(x)} \cdot \frac{N(x+\Delta x) - N(x)}{\Delta x} \quad \textbf{(15)}$$

Because the function $d\rho/dx$ may occasionally be increasing tremendously in the vicinity of criticality (dangerous shape in Fig. 4) the approach to criticality has to be carried out with continuously decreasing step width Δx in order to avoid the increase in reactivity $\Delta\rho(x)$, i.e. the product $(d\rho/dx)\cdot\Delta x$, to become too high.

5. Procedure of the Experiment

5.1. A prerequisite for carrying out the critical experiment through stepwise movement of the lower core section is an experimentally secured facility status. The excess reactivity of the reactor must not exceed the limit of 0.3 % for completely joined core sections.

5.2. First, the facility is checked in the same way as for a normal restart (i.e., among others, by movement of the lower core section by about 30 digits and by subsequent total shut-down by activating the push-button "Hand-RESA" (SCRAM)).

Additionally, the correct drop-off of the control rods has to be checked:

- For this purpose, the limit switch "Kernhälften zusammen" (core sections joined) has to be bridged by turning the key-switch "Simulation Kernhälften zusammen" (simulation of joined core sections). The key-switch administratively ensures that this operational state can be intentionally set only.
- The control rods are moved from their inner end position consecutively. The movement is interrupted, when the control-panel display shows that the respective control rod left its lower end position.
- A manual SCRAM ("Hand-RESA") signal is released at the control panel.
- The drop-off of the control rods has to be checked on the monitor screens of the control desk (confirmation by indication of the lower end position).

After this check, the facility must not be shut down until the end of the critical experiment.

The correct operation of the partial and the total shut-down systems is a condition for the start of the critical experiment.

5.3. When starting the experiment, the core sections are completely separated. Then, the start neutron source is moved in and the signal "Kernhälften zusammen" (core sections joined) is bridged via the key-switch "Simulation Kernhälften zusammen" (simulation of joined core sections). Subsequently, all control rods have to be withdrawn out of the reactor core up to their end positions.

5.4. Using the wide-range channels i (i=1,2), the pulse rates N_{out} (1,i) are measured for the completely separated core sections. The control rods are drawn in and the pulse rates N_{in} (1,i) are measured. The measured values of the wide-range channels are displayed directly on the monitor screen of the control desk or indirectly from the presented graphical diagram. These values have to be recorded according to the protocol sheet in the appendix of this instruction. For improving the reliability of the output values, the average of three

measurements is used.

5.5. The lower core section is lifted by 10 mm (100 digits). The control rods are pulled out and the pulse rates N_{out} (2,i) are measured. The control rods are drawn in, again, and the pulse rates N_{in} (2,i) are measured. For improving the reliability of the measurement, the average values of two measurements each are used.

For each wide-range channel i the ratios W_{out} (2,i) = N_{out} (1,i) / N_{out} (2,i) and W_{in} (2,i) = N_{in} (1,i) / N_{in} (2,i) are calculated and recorded in the protocol sheet.

The resulting values W_{out} (2,i) and W_{in} (2,i) are plotted versus the position x of the lower core section and connected each via a straight line with the points W_{out} (1,i) and W_{in} (1,i) [W_{out} (1,i) = N_{out} (1,i) / N_{out} (1,i) = 1 and W_{in} (1,i) = N_{in} (1,i) / N_{in} (1,i) = 1]. The extrapolation of both the lines to the abscissa gives the estimated positions $x_{crit, out}$ (i) and $x_{crit, in}$ (i) for the critical reactor with drawn out control rods and drawn in rods (Fig. 5). The difference between $x_{crit, out}$ (i) and $x_{crit, in}$ (i) is a measure for the control range of the rods expressed in terms of lower core section positions for the critical reactor. For a successful end of the critical experiment, the technically possible maximum position x_{max} (position for joined core section) must range between $x_{crit, out}$ (i) und $x_{crit, in}$ (i).

5.6. The smallest of the values $x_{crit, out}$ (i) is set as the estimated critical position. In the next step, the lower core section is lifted by the half of the difference between the current position and the minimal critical position, but not more than 10 mm (100 digits), i.e.

$$\Delta x_{max} = \frac{x_{crit,min} - x}{2} \leq 100 \; digit \qquad \textbf{(16)}$$

Then, new pulse rates N_{out} (3,i) and N_{in} (3,i) are measured with control rods drawn out and in. The ratios W_{out} (3,i) = N_{out} (1,i) / N_{out} (3,i) and W_{in} (3,i) = N_{in} (1,i) / N_{in} (3,i) are calculated and plotted in the diagram. A straight line is drawn between the points W_{out} (2,i) and W_{out} (3,i) as well as between W_{in} (2,i) and W_{in} (3,i). The extrapolation of the resulting straight lines to the abscissa gives new (more precise than before) critical positions $x_{crit, out}$ (i) and $x_{crit, in}$ (i) of the lower core section.

5.7. Item 5.6 is repeated as long as the extrapolation intersections with the abscissa do reliably fulfil the condition $x_{crit, out}$ (i) < x_{max} < $x_{crit, in}$ (i). Fig. 5 corresponds to the expected result of the experiment.

5.8. If $x_{crit, out}$ (i) > x_{max}, the reactor core contains too less fuel, i.e. the reactor will not become critical. If $x_{crit, in}$ (i) < x_{max}, the reactor core contains too much fuel. The reactor has to be shut down immediatly and fuel has to be removed.

5.9. After providing the condition in item 5.7, the distance between the two core sections has to be reduced to zero using smaller steps for finishing the critical experiment (core sections

completely joined). The gain in reactivity has to be compensated by drawing in the control rods.

5.10. By turning back the key-switch "Simulation Kernhälften zusammen" (simulation of joined core sections) the bridge of the limit switch "Kernhälften zusammen" (core sections joined) is interrupted. The critical experiment ends with record of the critical control rod position and the time of achieving the critical state in the operation logbook.

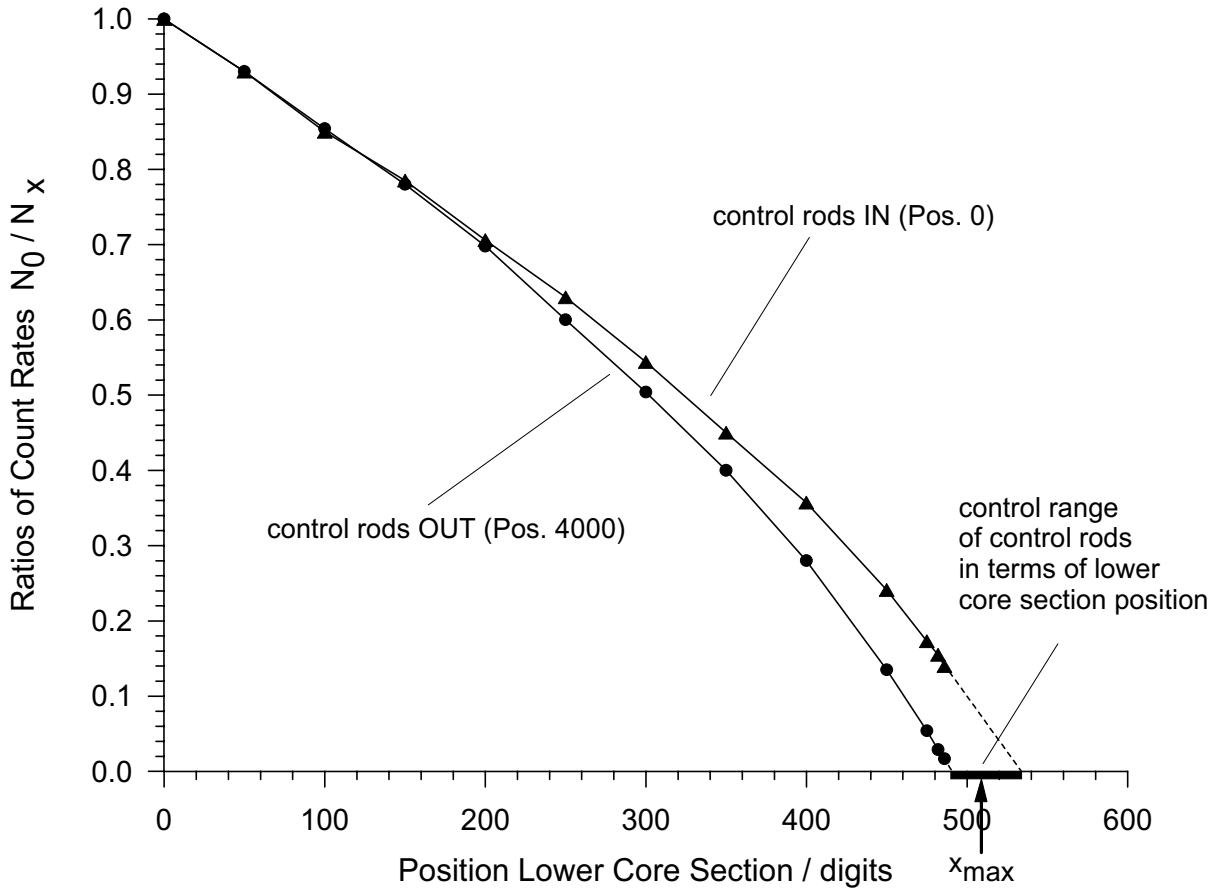


Fig. 5, Expected result of the critical experiment

6. Further Comments on the Experiment

The graphical representations of the measurements for determining the critical positions have to be prepared during the experiment.

For this purpose, the following items are needed:
- millimetre paper (see also appendix)
- rulers
- pocket calculator

The following values have to be recorded in a table or calculated (each for drawn in or drawn out control rods).

Position of lower core section x_i / digits	Control rod position in / out	Count rates N(x,i)	Ratios W(x,i)	Multiplica-tion factor k(x)	Subcritical amplification M(x)	Reactivity ρ(x) / %

An example for the table is provided in the appendix.

7. Questions to Answer

1. Why is it mandatory to carry out a critical experiment before initial commissioning of a nuclear reactor ?

2. Give some quantities that affect the criticality of a nuclear facility!

3. What are the possible operational states of a nuclear reactor ? What operational states are allowed and what operational states must be avoided in any case? Justify the answers!

Appendix: Protocol Sheet of the Critical Experiment

Equations:

$$k_i = 1 + \frac{N_{i-1}}{N_i}\,(k_{i-1} - 1) \qquad \text{with } k_0 = 0.945 \tag{1}$$

$$M_i = \frac{1}{1 - k_i} \tag{2}$$

$$\varrho_i = \frac{k_i - 1}{k_i} \tag{3}$$

Maximum permitted step for the movement of the lower core section:

$$\Delta x_{max} = \frac{x_{crit,\,min} - x}{2} \leq 100 \text{ digit} \tag{4}$$

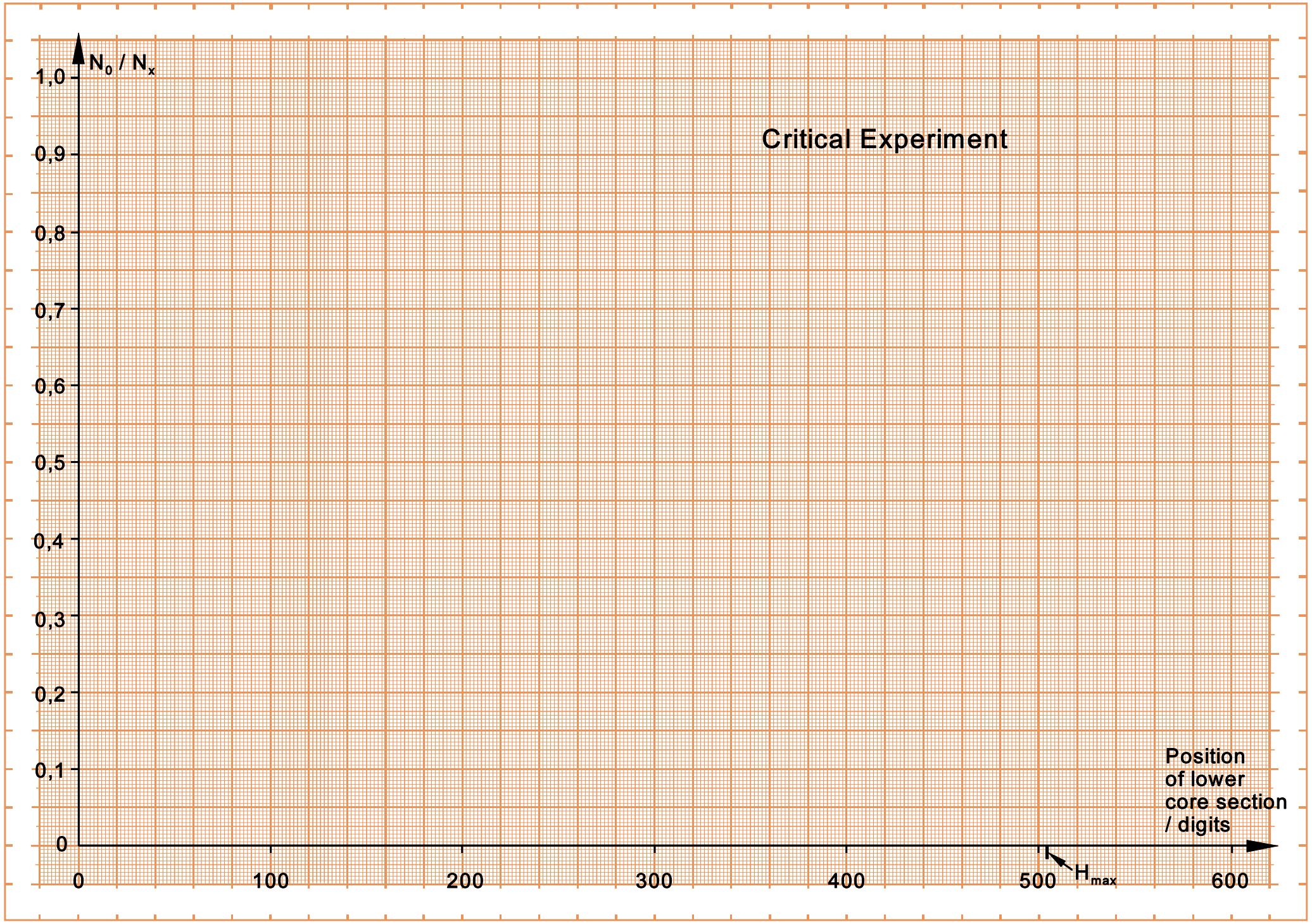
Wide-Range Channel 1 (WB 1)	Wide-Range Channel 2 (WB 2)
Pulse rates of fission chamber	Pulse rates of fission chamber
Name:	Name:

Determination of start values N_0 :

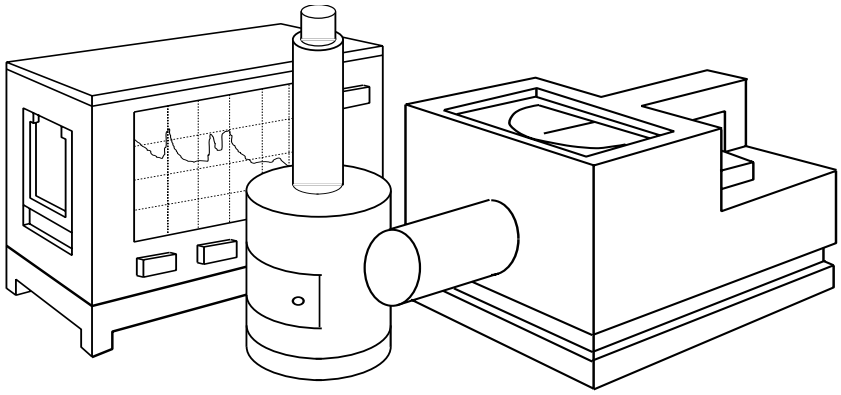
	Control rod position	Measure-ment 1	Measure-ment 2	Measure-ment 3	**Average value N_0**
WB 1	in (0)				
	out (4000)				
WB 2	in (0)				
	out (4000)				

No. of wide-range channel (WB1 / WB2) : .....

Position of lower core sect. x_i / digits	Control rod position	Count rates			Ratios $N_0 / \bar{N}_i$	Multiplication factor k_i	Subcritical amplification M_i	Reactivity ϱ_i
		$N_{i,1}$	$N_{i,2}$	$\bar{N}_i$				
0	in				1.00	0.945	18.18	-0.0582
	out				1.00	0.945	18.18	-0.0582
100	in							
	out							
	in							
	out							
	in							
	out							
	in							
	out							
	in							
	out							
	in							
	out							
	in							
	out							
	in							
	out							
	in							
	out							
	in							
	out							
	in							
	out							


Critical Experiment
N0 / Nx
1,0
0,9
0,8
0,7
0,6
0,5
0,4
0,3
0,2
0,1
0
0
100
200
300
400
500
600
Hmax
Position
of lower
core section
/ digits

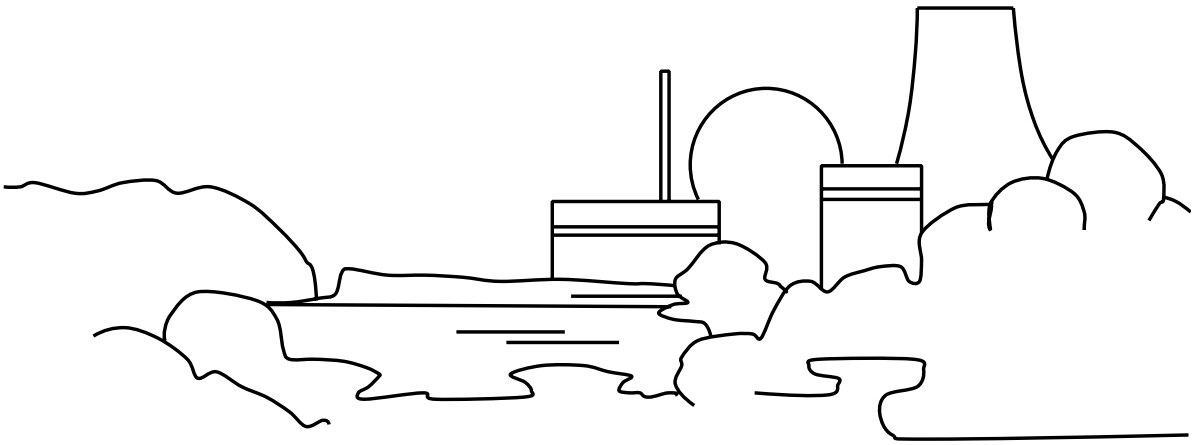
TECHNICAL UNIVERSITY DRESDEN
Institute of Power Engineering
Training Reactor



Reactor Training Course

Experiment

"Adjoint Flux Function"



Instruction
Experiment "Adjoint Flux Function"

Content:

Figures:

(issued: March 2015)

1. Motivation

For the safe operation of a nuclear reactor it is necessary to know all effects that result from changes of both, the material composition and the geometry of the reactor core.

The influence of such changes on the reactor-physical behaviour can be described globally by the resulting reactivity changes.

By measuring reactivity characteristics for samples made from different materials the qualitative and quantitative influence of typical structural materials of nuclear reactors on the reactivity balance can be investigated.

2. Tasks

1. The movement of structural material samples within the central experimental channel of the reactor is realised by a thin-walled aluminum leading tube. Within the tube, the respective sample is fixed at a particular position by distance holders. The tube is labeled in such a way that the position of the sample inside the reactor is determined as function of the distance r from the reactor core center (symmetry axis).

2. Before measuring the samples, the background reactivity of the leading tube is measured after filling it with distance holder elements only (i.e. without any structural material sample) and completely inserting it into the central experimental channel EK 1-2.

3. The reactivity characteristics $\rho(r)$ are measured for samples made from different materials (e.g. absorber, scatterer, fuel, cavity).
   For this reason, the reactivity changes caused by position changes of the sample in the reactor is compensated using calibrated control rods. With help of the control rod characteristics (available at the facility as a result of control rod calibration) the corresponding value of change in reactivity can be determined.

4. The measured reactivity characteristics have to be discussed.

5. The composite and the interaction cross section of an unknown sample can be determined by comparison of the reactivity characteristic of the unknown sample with those of known samples.

3. Introduction

In the experiment "Control Rod Calibration", the influence of an absorber with small spatial dimension on the reactor core has already been discussed. It was shown that the reactivity loss caused by a differential part of the rod dz located at the position z is the larger
- the larger the macroscopic absorption cross section is,
- the larger the neutron flux density Φ at the position z is, and
- the longer the rod part dz is.

This is a result of perturbation theory in one-group approximation.

In continuation to this problem, in the given experiment "Adjoint Flux Function", not only absorbers but also other materials (scattering materials, fuel, cavity) can be investigated. Their position relative to the reactor core will not be changed vertically but horizontally using the experimental channel EK 1-2 (see Fig. 1).

The theoretical background with reference to the application of perturbation theory to radial problems in a two-group approximation is given in the appendix.

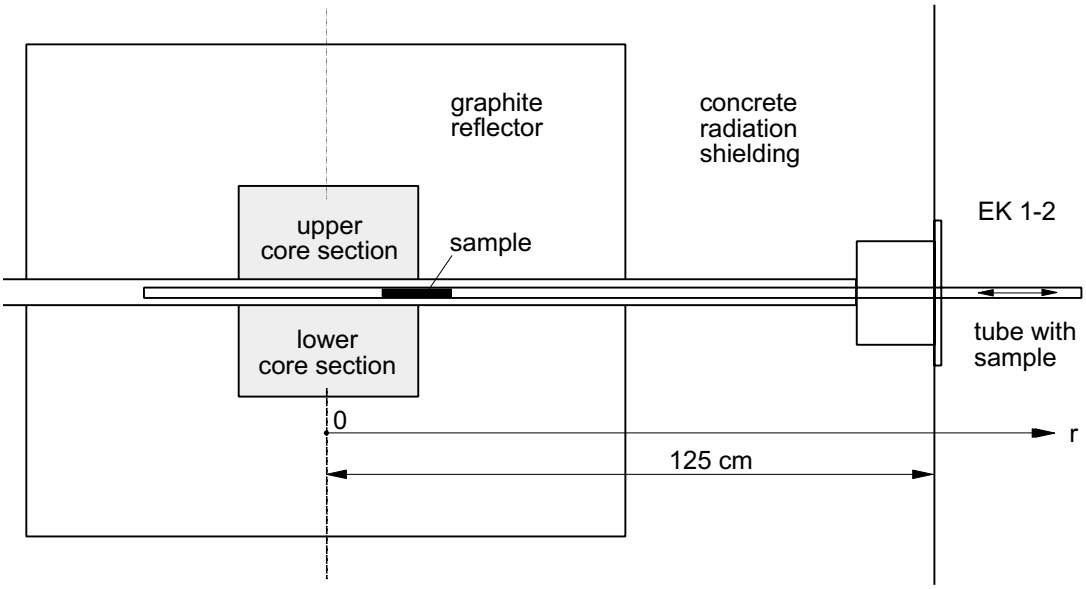


Fig. 1, Experimental setup

4. Instructions Concerning the Experiment

Attention!
Position changes of the sample in the experimental channel EK 1-2 can cause positive reactivity changes! Avoid rapid movements with combined large step lengths! The full withdrawal or insertion of the leading tube is only allowed for a partially shut-down reactor!

1. Investigation of the influence of the empty aluminium leading tube

2. Investigation of the reactivity influence of the tube filled with distance holders only (determine the critical rod position)

3. Insertion of a scattering / moderator sample (cylinder of PE, 10 cm long) into experimental channel EK 1-2 using steps of 5 cm length each within a range from -30 cm ... 0 cm ... +30 cm (0 cm indicates the position of the reactor core center)

4. Plot of the adjoint flux for scattering material

5. Insertion of an absorber sample (cylinder of PVC, 5cm long) into experimental channel EK 1-2 using steps of 5 cm length each within a range from -30 cm ... 0 cm ... +30 cm (0 cm indicates the position of the core center)

6. Plot of the adjoint flux for absorber material

7. Possible repetition of points 5 and 6 e.g. for a graphite or a fuel sample

5. Instructions Concerning the Protocol

- Short description of the objectives of the experiment
- Tables of the measured values
- Analysis of the measurements including intermediate results
- Graphical presentation of the experimentally found characteristics. For the absorber material, a comparison with the theoretically found characteristic should be given. The characteristic of the scatterer has to be discussed qualitatively.
- An error analysis has to be provided

6. Questions to Answer

1. Make sure how to determine the reactivity, which is due to a particular sample and is compensated by a control rod, using a differential or integral rod characteristics!

2. What is the methodical difference between the determination of an integral control rod characteristics and the determination of the characteristics of a sample whose dimensions are small compared to those of the reactor core?

3. Consider a graphite sample (assuming: only $D \neq 0$) being as a scatterer in the center of the reactor core ($r = 0$). Why not $\rho = 0$ has to be expected?

4. Make sure the procedure for the determination of the Uranium content in a sample using the reactivity characteristics!

Appendix 1: Physical Background

As an approximation for the description of neutron transport we use the diffusion equation in the two-group approximation.

$$\begin{aligned} \operatorname{div}(D_1 \cdot \operatorname{grad}\Phi_1) - \Sigma_{rem}\Phi_1 + \varepsilon\cdot\nu\cdot\Sigma_{f,2}\Phi_2 &= 0 \\ \operatorname{div}(D_2 \cdot \operatorname{grad}\Phi_2) - \Sigma_{a,2}\Phi_2 + p\cdot\Sigma_{s,1\to2}\Phi_1 &= 0 \end{aligned} \qquad (1)$$

with $\Sigma_{rem} = \Sigma_{a,1} + \Sigma_{s,1\to2}$, $\Sigma_{s,1\to2} = \dfrac{\Sigma_{s,1}}{\frac{1}{\zeta}\ln\frac{E_o}{E_g}}$

and $D = \dfrac{1}{3\cdot\Sigma_{tr}} = \dfrac{1}{3(\Sigma_T - \Sigma_s\cdot\bar{\mu})}$, $\bar{\mu} = \dfrac{2}{3\cdot A}$

Small perturbations in the context of perturbation theory are deviations of the coefficients D1, D2, Σ_{rem}, $\Sigma_{a,2}$, $\Sigma_{f,2}$, $\Sigma_{s,1\to2}$, while it is assumed that the neutron flux distribution in the reactor is not changed by the sample.

Equation (1) has the following matrix representation:

$$\underline{M}\ \underline{\Phi} = 0 \qquad (2)$$

If the sample that causes the perturbation S is introduced into the reactor, the reactor becomes subcritical or supercritical, i.e.

$$\varrho = \frac{k'_{eff} - 1}{k'_{eff}} \neq 0 \qquad (3)$$

with $k'_{eff} = k^{o}_{eff} + \Delta k_{eff} \neq 1$ ... perturbed state

and $k^{o}_{eff} = 1$ ... unperturbed state

The perturbation caused by the change in the coefficients is compensated, e.g., by re-adjustment of the control rods. It results:

$$k_{eff} = k'_{eff} - \Delta k_{eff} = 1$$

Recriticality corresponds to

$$(\underline{M} + \underline{S}) \cdot \underline{\Phi} = 0 \tag{4}$$

$$\text{with} \quad \underline{S} = \begin{vmatrix} \operatorname{div} \delta D_1 \cdot \operatorname{grad} \Phi_1 - \delta\Sigma_{rem} & \varepsilon \cdot \nu \cdot \delta\Sigma_{f,2} - \Delta k_{eff} \cdot \varepsilon \cdot \nu \cdot \Sigma_{f,2} \\ p \cdot \delta\Sigma_{s,1\to 2} & \operatorname{div} \delta D_2 \cdot \operatorname{grad} \Phi_2 - \delta\Sigma_{a,2} \end{vmatrix} \tag{5}$$

The expression $-\Delta k_{eff} \cdot \varepsilon \cdot \nu \cdot \Sigma_{f,2}$ represents the compensation of the perturbation.

After setting up the adjoint equation

$$(\underline{M}^+ + \underline{S}^+) \cdot \underline{\Phi} = 0$$

and by using eigenvalue theory and carrying out several steps of algebra, for $\rho = \Delta k_{eff}$ follows

$$\begin{aligned} \varrho = \frac{1}{K} \int_{Reactor} [& -\delta D_1 \operatorname{grad} \Phi_1^+ \operatorname{grad} \Phi_1 - \delta\Sigma_{rem} \Phi_1^+ \Phi_1 + \varepsilon \cdot \nu \cdot \delta\Sigma_{f,2} \Phi_1^+ \Phi_2 \\ & -\delta D_2 \operatorname{grad} \Phi_2^+ \operatorname{grad} \Phi_2 - \delta\Sigma_{a,2} \Phi_2^+ \Phi_2 + p \cdot \Sigma_{s,1\to 2} \Phi_2^+ \Phi_1] dV \end{aligned} \tag{6}$$

$$\text{with} \quad K = \int_{Reactor} \varepsilon \cdot \nu \cdot \Sigma_{f,2} \Phi_1^+ \Phi_2 \, dV$$

In the given experiment, especially the influence of an absorber (only $\delta\Sigma_{a,2} \neq 0$) and the influence of a scatterer ($\delta\Sigma_s \neq 0$) are investigated, i.e.

$$\varrho_a = -\frac{1}{K} \int_{Reactor} \delta\Sigma_{a,2} \Phi_2^+ \Phi_2 \, dV \tag{7}$$

for the absorber and

$$\varrho_s = \frac{1}{K} \int_{Reactor} [-\delta D_2 \operatorname{grad} \Phi_2^+ \operatorname{grad} \Phi_2 + p \cdot \delta\Sigma_{s,1\to 2} \Phi_2^+ \Phi_1] dV \tag{8}$$

for the scatterer, because a change in Σ_s causes changes in the diffusion coefficient and also in the slowing-down cross section (frequently called moderation fraction). For cavities, the same equations are applicable with reverse algebraic plus/minus signs.

Equation (6) shows that an increase in the cross sections $\delta\Sigma_a$ und $\delta\Sigma_{s,\,1\to2}$ is weighted with $\Phi^+\Phi$, but a change of D is weighted with $\mathrm{grad}\Phi^+\mathrm{grad}\Phi$. An increase in the absorption cross sections results in a negative reactivity contribution. In the same way, an increase in the diffusion coefficient causes a negative reactivity contribution, since this corresponds to an increase in the leakage term.

Assuming that the perturbation occupies only a small part ΔV of the total volume of the reactor with negligible influence of the neutrons, the integration in equations (7) and (8) can be avoided with the result:

$$\varrho_a = -\frac{1}{K}\cdot\delta\Sigma_{a,2}\Phi_2^+\Phi_2\,\Delta V$$

$$\varrho_s = \frac{1}{K}\left(-\delta D_2 grad\,\Phi_2^+ grad\,\Phi_2 + p\cdot\delta\Sigma_{s,1\to2}\Phi_2^+\Phi_1\right)\Delta V \qquad (9)$$

Appendix 2: Application of Perturbation Theory to Experiments at the AKR (One-Group Theory)

In one-group theory, it is $\Phi_1 = \Phi_2$ und $\Phi_1 = \Phi_1^+$.

Having a cylindrical geometry like that of the AKR, the radial shape of Φ and $\frac{\partial}{\partial r}\,\Phi$ can be expressed using BESSEL functions:

$$\Phi = \Phi_{max} \cdot I_0 \cdot \left(\frac{2.405}{R + \delta} \cdot r \right)$$

$$\frac{\partial}{\partial r}\,\Phi = \Phi_{max} \cdot \frac{2.405}{R + \delta} \cdot I_1 \cdot \left(\frac{2.405}{R + \delta} \cdot r \right)$$

with R = 12.5 cm and δ = 6.35 cm.

The approximation using one-group theory is well applicable for absorbers with small reactivities.

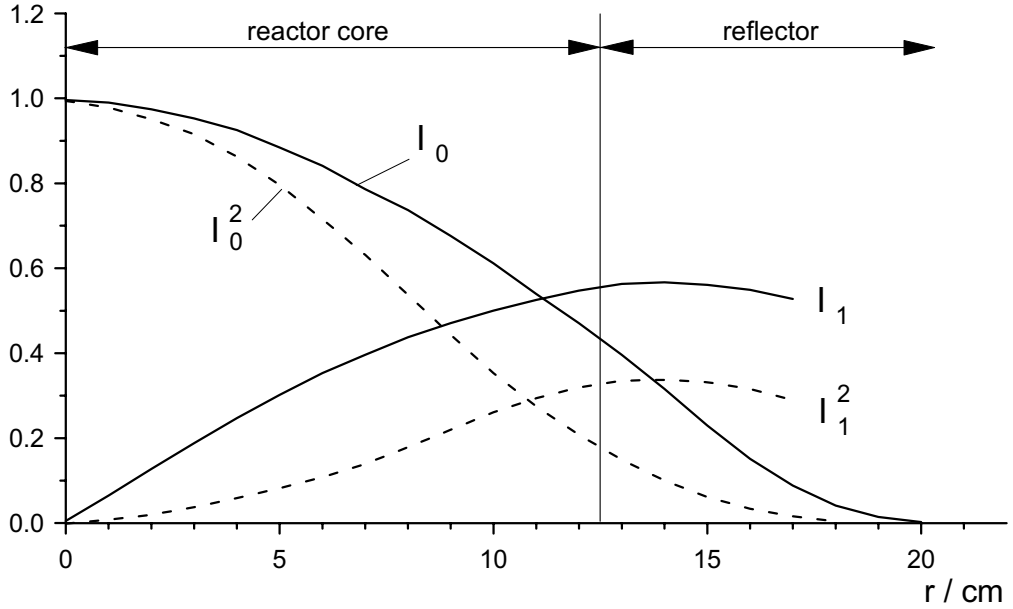


Fig. 2, Normalised BESSEL functions I_0 und I_1 and its squares for the radial dimensions of the AKR

Appendix 3: Application of Perturbation Theory to Experiments at the AKR (Two-Group Theory)

Figure 3 shows the radial distributions of Φ_1 , Φ_1^+ , Φ_2 , Φ_2^+ , $\frac{\partial}{\partial r}\Phi_2$ und $\frac{\partial}{\partial r}\Phi_2^+$, that were calculated for the AKR in the two-group approximation using the computer code RHEIN with the (one dimensional) transport method P1.

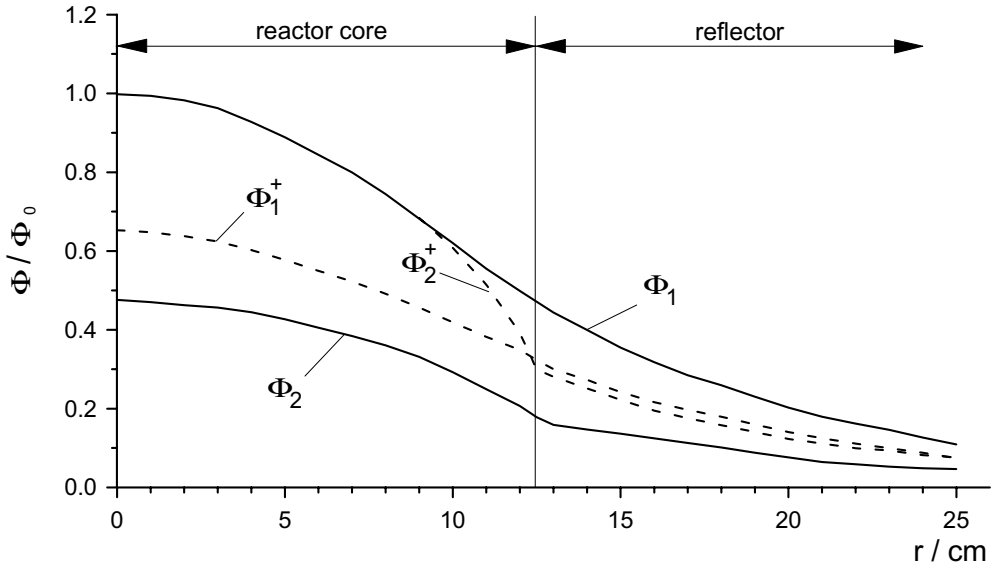


Fig. 3a, Radial distribution of the neutron fluxes and of the adjoint fluxes at the AKR

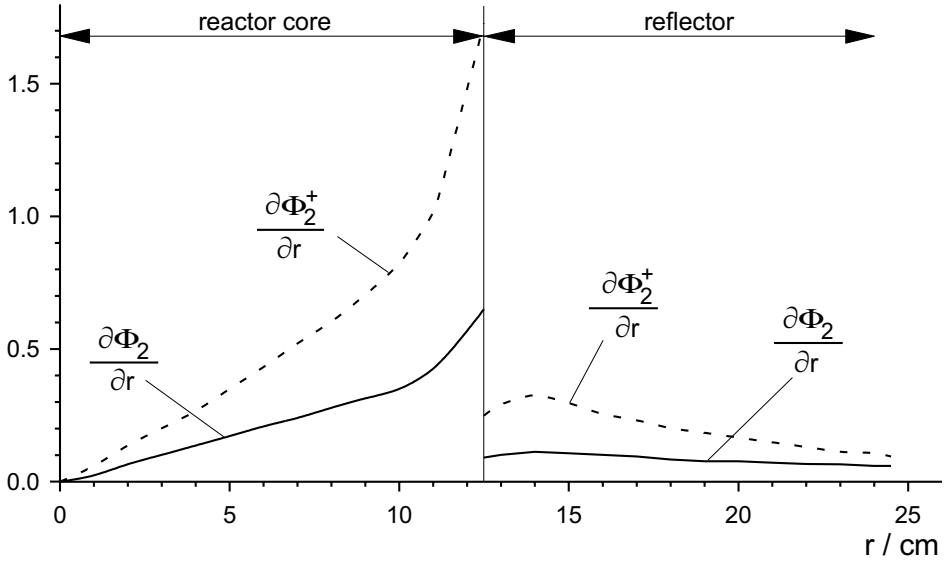


Fig. 3b, Radial distribution of $\frac{\partial}{\partial r}\Phi_2$ and $\frac{\partial}{\partial r}\Phi_2^+$ at the AKR

On the basis of their shapes, the experimentally determined reactivity characteristics of the AKR can be explained (see Figs. 4 and 5).

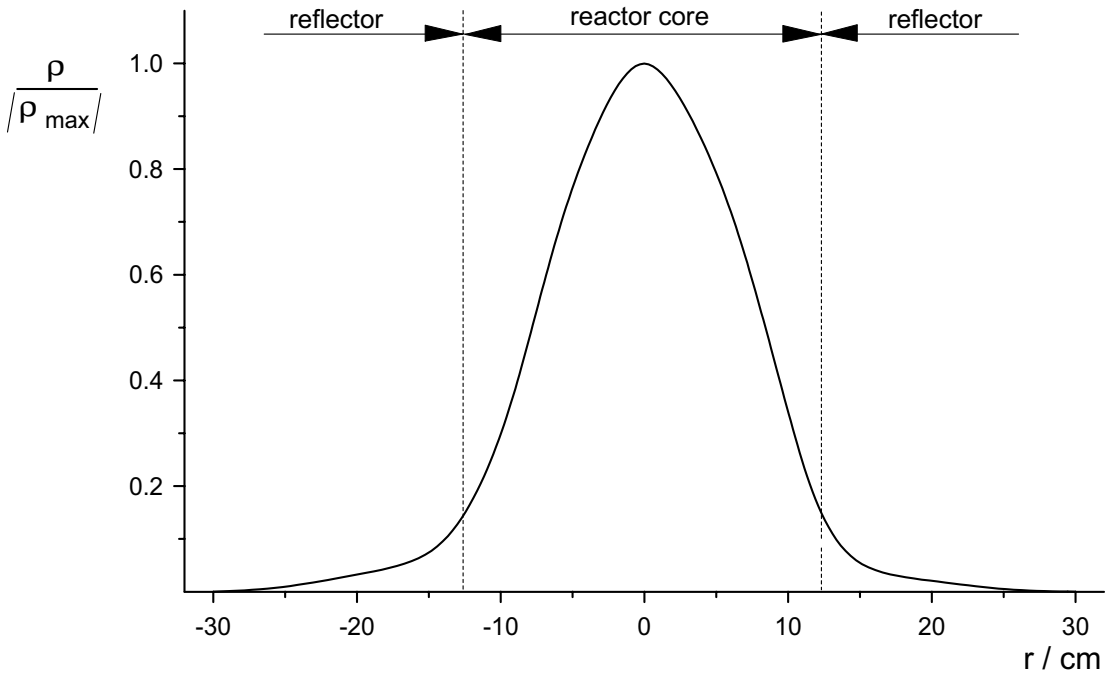


Fig. 4, Typical shape of an absorber characteristic in the AKR

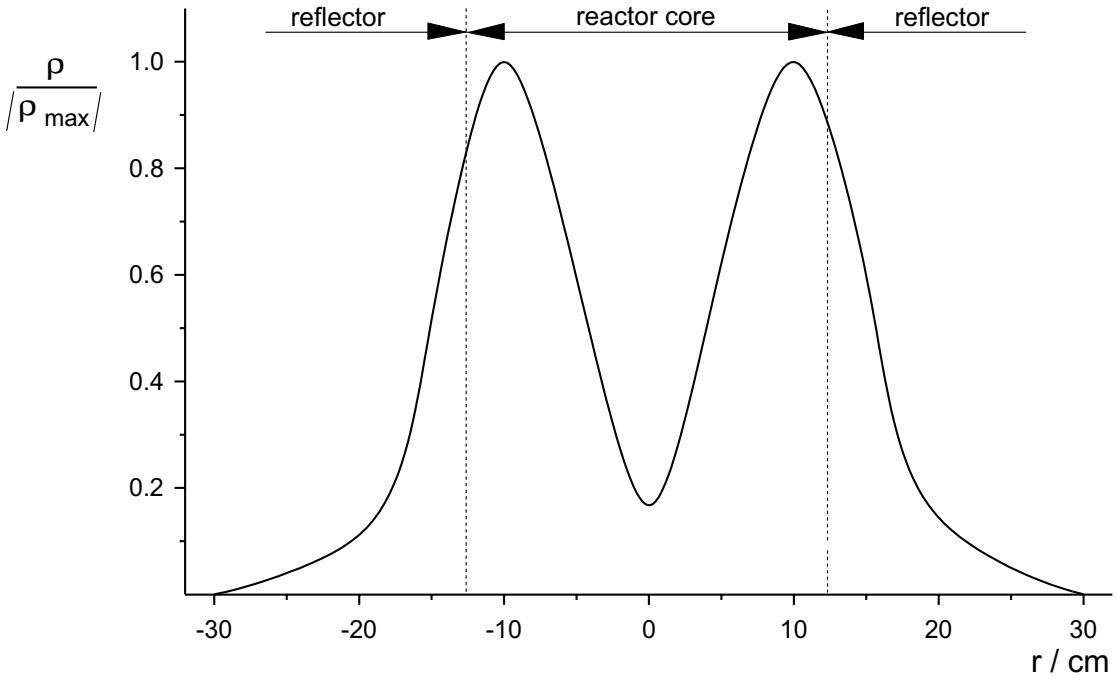


Fig. 5,Typical shape of a scatterer characteristic in the AKR

Appendix 4: Protocol forms

1. Empty sample tube with distance holders only (background measurement)

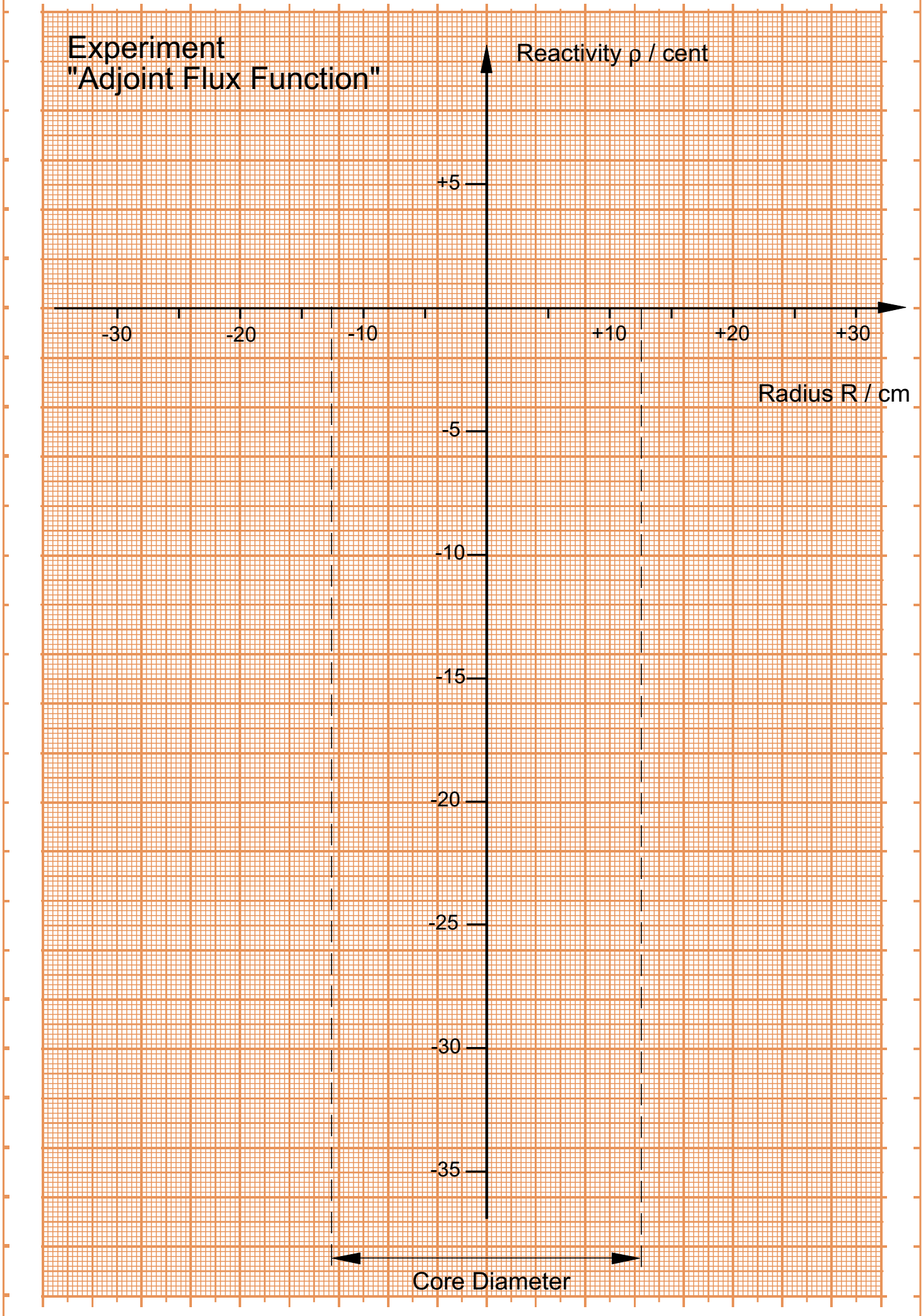
R / cm	**S1**	**S2**	**S3**	ρ **/ cent**
-30				
-25				
-20				
-15				
-10				
-5				
0				
+5				
+10				
+15				
+20				
+25				
+30				

2. Neutron scattering material (cylinder 10 cm polyethylen):

R / cm	**S1**	**S2**	**S3**	ρ **/ cent**
-30				
-25				
-20				
-15				
-10				
-5				
0				
+5				
+10				
+15				
+20				
+25				
+30				

3. Neutron absorbing material (cylinder 5 cm PVC):

R / cm	**S1**	**S2**	**S3**	ρ **/ cent**
-30				
-25				
-20				
-15				
-10				
-5				
0				
+5				
+10				
+15				
+20				
+25				
+30				


Experiment
"Adjoint Flux Function"
Reactivity ρ / cent
+5
-5
-10
-15
-20
-25
-30
-35
-30
-20
-10
+10
+20
+30
Radius R / cm
Core Diameter

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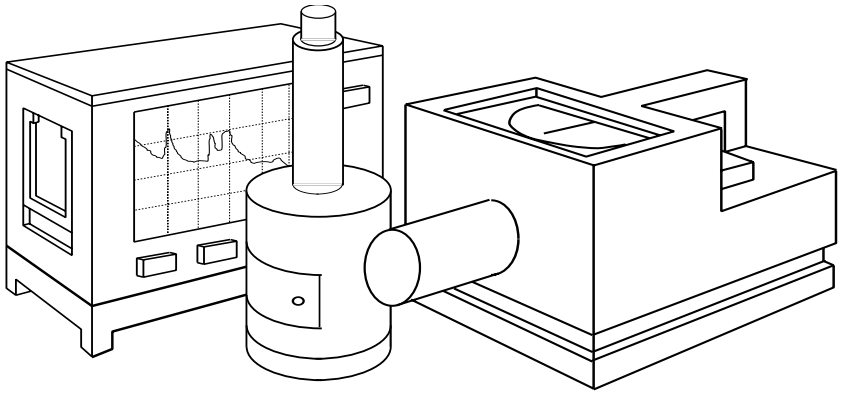
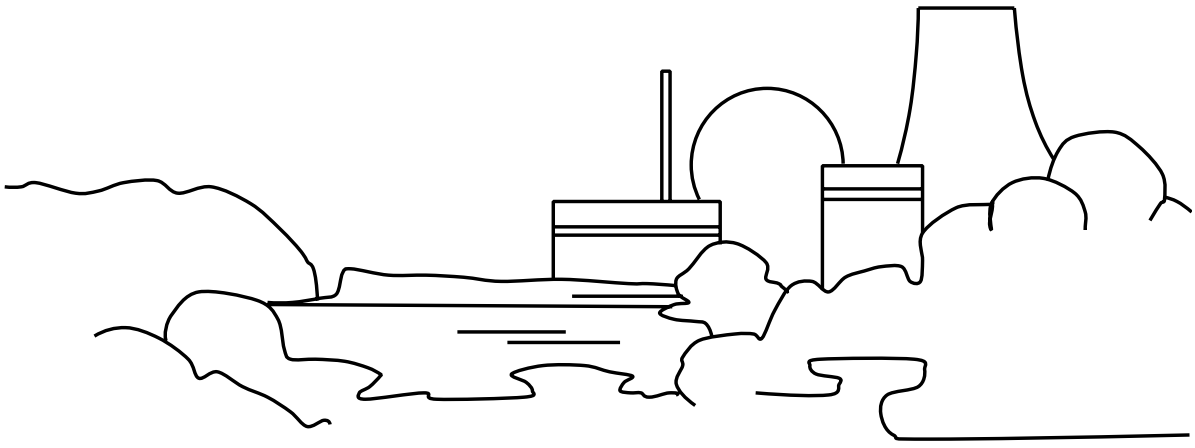


Reactor Training Course

Experiment

"Neutron Flux Distribution"



Instruction for
Experiment “Neutron Flux Distribution”

Content:

Figures:

(issued: March 2015)

1. Motivation

For any nuclear reactor, knowledge on the spatial neutron flux density distribution is of major interest: in research reactors, it is an input variable for many experiments (e.g. as the source strength for irradiation experiments), in power reactors, it is important for determining the distribution of heat sources. In the given experiment, the radial flux density distribution of the thermal neutrons is measured. The used method, i.e. the activation of suitable detector foils, is one of the most important methods in nuclear technology.

2. Introduction

Figure 1 shows the radial distribution of the thermal neutron flux density. It was calculated by multi-group diffusion calculations (considering both, thermal and fast neutrons). In the fuel region, the distribution approximately fits the cylindric BESSEL function $J_0(2.405 \cdot r/(R_K+\delta))$ that has its first root at $R = R_K+\delta$. The comparison of its shape with the shape of $J_0(2.405 \cdot r/R_K)$ for the bare reactor shows that due to the reflector, the reactor behaves like a bare reactor with a radius that was increased by the distance δ.

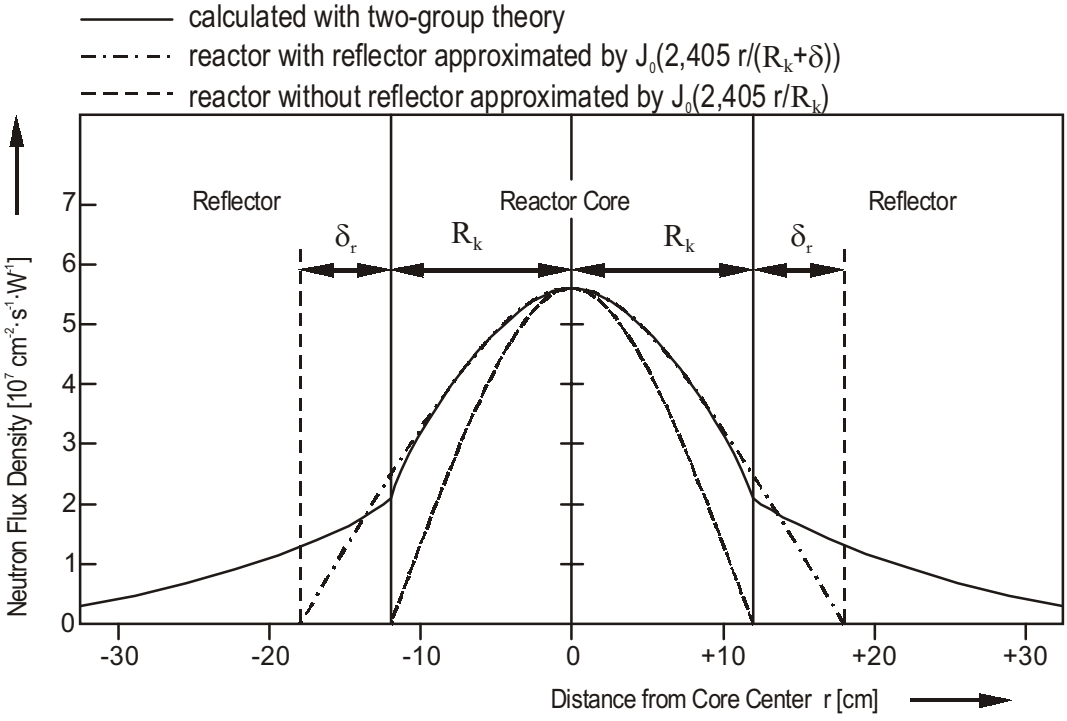


Fig. 1, Radial flux density in the AKR-2

For a reactor with a fixed (material) composition, the critical condition gives the reactor minimum size, which is determined such that the neutron losses by leakage are exactly compensated by the multiplication of the neutron content inside the core.

If adding a reflector, this minimum size, which is in the bare reactor $R = R_K+\delta$, is achieved with a much smaller core volume with the radius R_K. Thus, less of the valuable fuel needs to be utilised for a reactor with reflector than for a bare reactor. The parameter δ is called reflector saving.

Considering the axial neutron flux density, analogous results are found but with the cylinder function replaced by the trigonometric function $\cos(\pi\cdot(z/2)/(Z_K+\delta))$. The parameter Z_K stands for the half of the core height.

The parameters $B_r = 2.405/(R_K+\delta)$ and $B_Z = \pi\cdot(z/2)/(Z_K+\delta)$ of the distributions play a major role for determining the geometry. The quantity $B^2 = B_r^2+B_z^2$ is called geometric buckling because this quantity determines the second spatial derivative in the reactor fundamental equation

$$\Delta\Phi(r,z) + B^2\cdot\Phi(r,z) = 0$$

and, hence, the curvature of the flux.

3. Measurement of the Neutron Flux Density Distribution

Neutron activation samples are used as detectors. These are insensitive to γ-radiation and they can be kept small enough that they are spatially well-defined and do not disturb the reactor neutron flux density.

Several samples can be irradiated simultaneously at different positions inside the reactor such that the measurement of the relative flux density is not influenced by reactor power fluctuations.

For measurements of thermal neutrons, manganese suits well due to the reaction:

$$^{55}Mn(n,\gamma)^{56}Mn \xrightarrow{T_{1/2} = 2{,}58\ h} {}^{56}Fe + \beta^- + \gamma$$

In the thermal energy region, the capture cross section of manganese (σ_0 = 13.3 b) is well proportional to the inverse neutron velocity (i.e. a so-called 1/v-absorber). Resonances exist only for energies higher than 0.337 keV and contribute to the integral cross section less than 5 %. Of course, this value can vary at different positions in the reactor because of spatial dependence of the neutron energy spectrum. But using manganese this effect can be neglected, since it ranges within the achievable error thresholds of the experiment. The half-life of 2.58 h is from the experimental point of view very convenient: On the one hand, it is large enough that time corrections are not essential and, on the other hand, it is small enough that also low neutron flux values give well-measurable activities due to approaching saturation activity. Fig. 2 shows the decay scheme and the typical gamma energy spectrum of manganese.

For the measurement of the activity of ^{56}Mn, the emitted γ-radiation is advantageously used, since the self-absorption within the sample is very low. Thus, perturbations due to irregularities due to the thickness of the sample is avoided.

Due to the design of the AKR-2, the activation samples cannot be placed into the reactor core itself. But the samples can be arranged in the central experiment channel EK1-2, which leads horizontally through the reactor core centre. If aluminium is used as a sample holder, which behaves from reactor-physics point of view like air, the activation detectors are surrounded by materials whose scattering and moderation properties are similar to those of the core and reflector materials. Thus, the measured neutron flux density distribution $\Phi(r)$ is identical to that of the reactor core which results from the two-group diffusion theory (Fig. 3).

The neutron activation detectors used in the experiment are disks of manganese having a diameter of about 1 cm. The aluminium sample holder has cuts after every 2 cm for inserting the disks. The weights of all manganese samples range between (0.048 ... 0.051) g and can be considered to be identical with regard to the overall accuracy of the experiment.

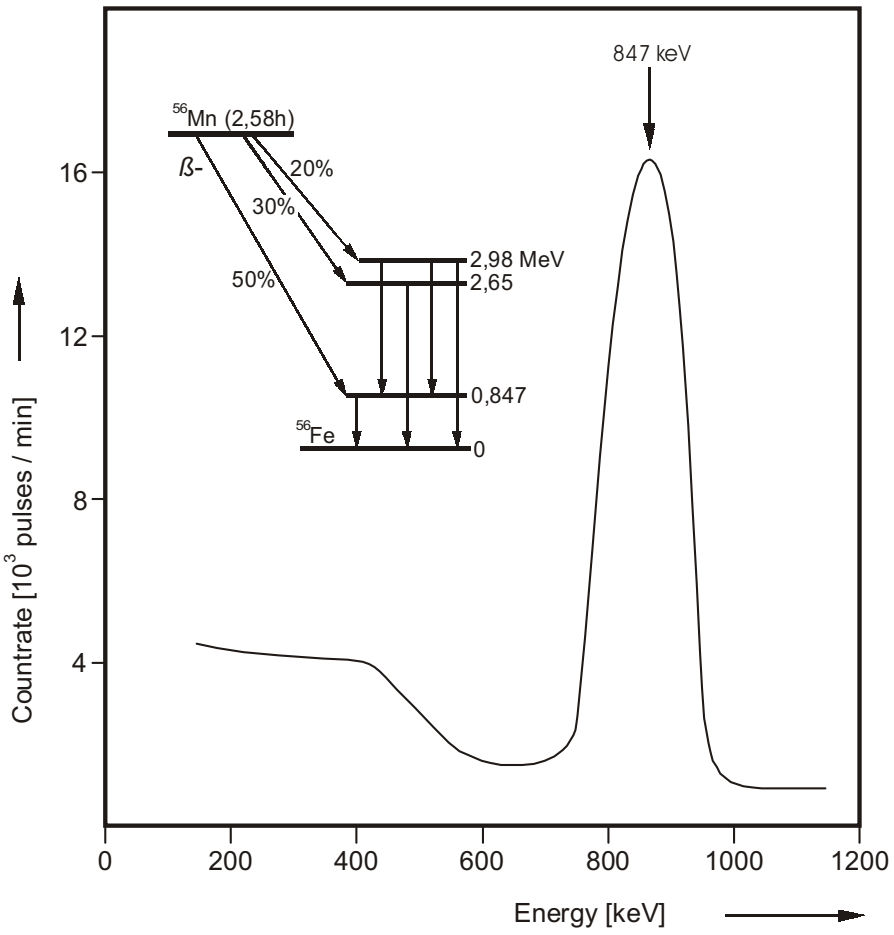


Fig. 2, Decay scheme and gamma energy spectrum of ^{56}Mn

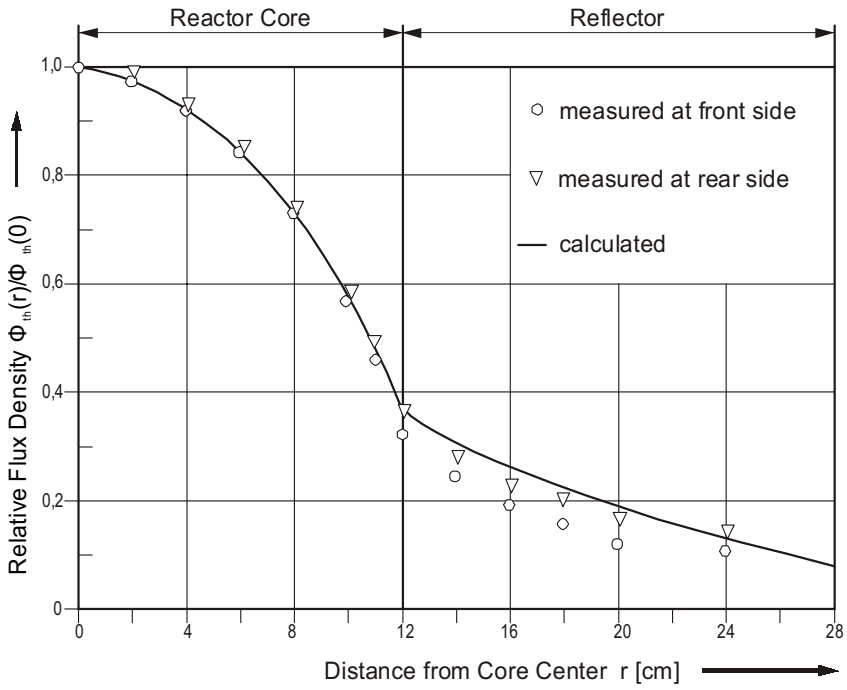


Fig. 3, Typical relative radial flux density distribution in the AKR-2

4. Instructions for the Experiment

4.1. Equipment of the Measurement

The measurement setup is presented in Fig. 4. It consists of the scintillator-photomultiplier-unit, a preamplifier and a computer for the analysis. The activation samples and the aluminium sample holder are visible in the foreground of the picture as well. The scintillator is coupled to a multichannel analyser via a preamplifier and a spectroscopic amplifier. In the preamplifier, the primary charge pulses are converted into voltage pulses (output-voltage is proportional to the input charge). In the spectroscopic amplifier, the pulses are amplified up to a range of 1 ... 10V.

The multichannel analyser recognises the incoming pulses, evaluates them with respect to their height (voltage), which contains the information about the energy, and sorts them into consecutive pulse-height channels. The result is a pulse-height spectrum that contains the energy information of all recognised gamma quants. Each measured gamma quant is a count in the corresponding energy channel of the multichannel analyser.

It should be realised that even discrete gamma energies do not cause sharp lines in the spectrum, because of statistical fluctuations of the elementary processes in the detector and in the post-processing electronics. This phenomenon creates from theoretically sharp lines GAUSSIAN bell-curves (peaks). The mean pulse heights correspond to the energy of the gamma quants. Using a PC programme, the gamma energy spectrum can either be observed on the screen, directly evaluated or saved.

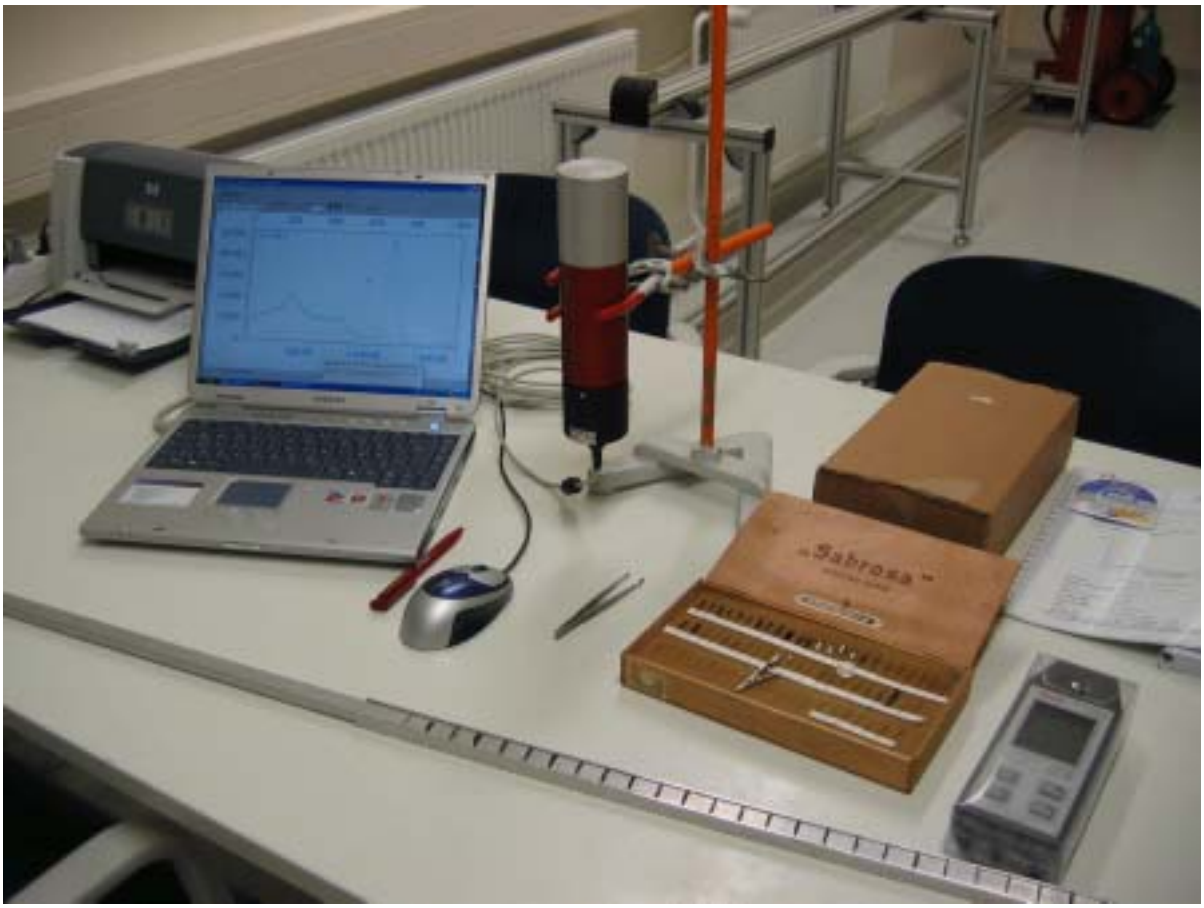


Fig. 4, Measurement equipment

Detectors for gamma spectroscopy are usually scintillation detectors (e.g. NaI) or semiconductor detectors made of Ge(Li) or increasingly high purity germanium (HPGE).

In the given experiment, NaI is used as scintillation detector for the measurements. Scintillation detectors are relatively easy to handle and have a high efficiency. A disadvantage of scintillators is the limited energy resolution, i.e. the ability to resolve two gamma peaks that are located very close to each other.

Scintillation detectors consist of a combination of a luminescent material (scintillator) that is stimulated by ionising radiation to emit flashes (scintillations) and a photomultiplier that converts the flashes into electrical pulses.

Within the scintillator, the absorbed energy of a gamma quant is converted into light by excitation of the scintillator material and its subsequent return to the ground state. For spectrometric purposes, the absorbed gamma energy and the number of emitted light quants and the subsequent pulse height at the output of the photomultiplier must be proportional to each other.

Inorganic mono-crystals made from sodium iodide (NaI) doped with Thallium for activating the light emission has proved a suitable scintillator material in spectroscopy of gamma radiation (NaI(Tl)-scintillators). Because of their high physical density $\rho = 3.67$ g/cm³, their high content

of iodine of 85 wt%, and due to the ordinal number of Z = 53, they have a high absorption capability for gamma radiation. Clear crystals that are homogeneously doped and transmissible for their own fluorescent light can be produced in dimensions up to a thickness of e.g. 300 mm or diameter of 400 mm. Also high-energy gamma quants are sufficiently absorbed by these crystals. Commercially available crystals are hermetically sealed (because NaI is hygroscopic) e.g. within aluminium cases, having a glass or plastic window for the light emission. In order to avoid light losses all surfaces of the crystal except for the emission window are not polished but surrounded with a reflecting material, e.g. MgO.

The energy resolution of NaI(Tl)-scintillators is usually given with regard to the gamma line of the radionuclide Cs-137 (E_γ = 662 keV). Best energy resolution obtained with selected NaI-scintillation crystals and photomultiplier combinations are in the order of about 6 %, common values for commercial scintillator crystals and photomultipliers range between (8...12) % FWHM.

The scintillations from the crystal need to be transmitted to the photocathode of the photomultiplier with only as small as possible losses. Immersion layers (e.g. silicone oil) between scintillator surface and photomultiplier reduce total reflection.

By the photoelectric effect, a few free electrons are emitted on the photocathode. Between the photocathode and the anode of the photomultiplier, a graduated high voltage is applied via a voltage divider and a series of intermediate electrodes (dynodes). In the electric field between each two of the dynodes, the electrons accelerate and liberate further electrons by collision ionisation at the next dynode. This effect leads to a low-noise amplification of the initial number of electrons (factor 10^5...10^9). Thus, scintillation flashes are transformed into energy-proportional electric current and voltage signals and are amplified in the photomultiplier. Since ambient light would completely overlap the scintillation effect of the detector and also stimulate the photocathode of the photomultiplier to emit electrons, the entire scintillator-photomultiplier system needs to be light-tightly sealed.

4.2. Irradiation of the Activation Samples

The sample holder is filled with the manganese sample disks. For the experiment, 30 samples are arranged with a spacing of 2 cm each. This allows a full measurement of the neutron flux in the core and reflector region of the AKR-2.

Afterwards, the experimental channel EK1-2 of the AKR-2 has to be opened, inserted radiation protection plugs have to be unloaded and wipe test for checking possible contamination inside the channel have to be carried out. Then, the reactor is started and made critical at 2 W power. For the further experimental procedure, it is reasonable to use the automatic power control. At a steady-state power of 2 W, the sample holder is inserted into the experimental channel.

A label at the sample holder simplifies the positioning. The start time of the irradiation has to be recorded. The irradiation should last about 45 min. During this time, presence in front of experimental channel should be avoided due to reasons of radiation protection.

4.3. Energy Calibration of the Spectrometer

During the time of irradiation, the energy calibration of the spectrometer can be carried out.

Analysing spectra of pulse heights, only the channel numbers of peaks and the respective number of pulses per channel are available. In order to link the channel number of the multi-channel analyser to the corresponding gamma energy an energy calibration of the spectrometer is necessary.

For the calibration, pulse height spectra of several well known gamma sources (calibration sources) are measured with the multichannel analyser and the channel numbers of the measured photopeaks are related to the well known photopeak gamma energies. A set of such calibration sources is available at the AKR-2 consisting e.g. of nuclides given in Tab. 1.

Nuclide	Co-60	Cs-137	Kr-85	Ho-166m	Pb-210	Am-241
Energy [keV]	1173 1332	662	514	184, 280, 411, 712, 810	47	60

Tab. 1, Calibration sources and their gamma energies

After booting the PC and connecting the scintillator (type: ScintiSpec) via USB-cable, the program winTMCA32 has to be started. Via the pull-down menu HARDWARE the high voltage can be adjusted and set to an appropriate value. In the given experiment, a high voltage of 681 V is used.

The calibration sources, one after the other, have to be placed onto the scintillator. In the pull-down menu SPEKTRUM the item ENERGIEKALIBRIERUNG has to be activated. The cursor is moved to the first photopeak whose energy shall be calibrated and is set by a double-click. The corresponding energy has to be recorded before continuing to the next line in the calibration table. This procedure is repeated for all further peaks of the calibration spectrum. Afterwards, the button FIT has to be pressed and finally the calibration is fixed with the button SETZEN. Thus, the energy calibration is ready to use for subsequent measurements.

4.4. Evaluation and Protocol

After irradiation of the samples for a time of about 45 min in the experimental channel EK1-2, the reactor has to be shut down (after having deactivated the automatic power control). The time of reactor shut-down has to be recorded. Since the half-life of the aluminium sample holder is 137 s, a high activation of the holder itself can be expected (saturation activity!). Therefore, the sample holder should remain in the experimental channel for about further 10 min after reactor shut-down for decay of its radioactivity. Only after measuring the dose rate, transportation to the evaluation desk is allowed.

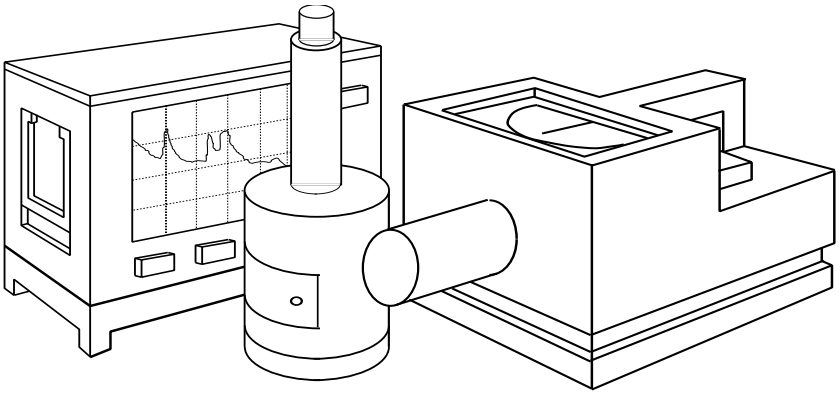
For the evaluation of the activation samples, the photopeak of ^{56}Mn of 847 keV energy is used. In the given experiment, the small mass tolerances of the manganese samples are irrelevant. Though in an absolute measurement, the tolerances had to be taken into account.

For the measurement of the particular samples, the live-time of the spectrometer has to be set to 60 s. The particular samples are measured one after the other and the start time of each measurement has to be recorded. This enables a correction of the count rate that has been already decreased since the end of the irradiation. It is reasonable to elaborate a suitable routine to analyse the data.

The determination of the relative activity is done by measuring the peak areas, i.e. said more precisely the net peak areas. After the measurement of each sample, the net areas can be determined by the following functions of the program winTMCA32:

- Open the pull-down menu BERECHNEN
- Mark the peak by using the button PEAKSUCHE
- By repeated use of the pull-down menu BERECHNEN, the spectrum inside the peak limits is integrated (SPEKTRUM INTEGRIERT) and an additional window appears.
- By activation of the flag ZÄHLRATE the net pulse rate is displayed.
- After having recorded the net pulse rate, the window has to be closed and the spectrum to be deleted. Afterwards, the spectrometer is ready for the measurement of the next Mn sample.

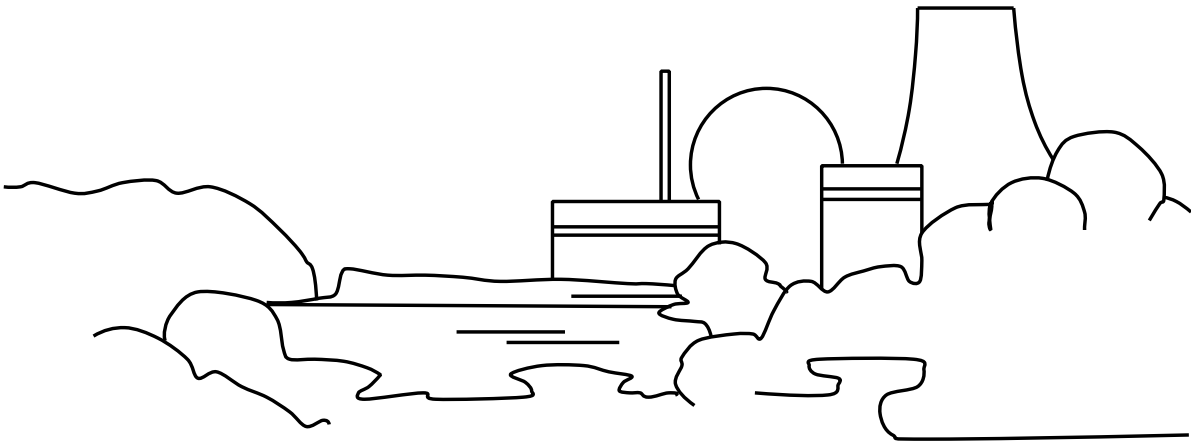
TECHNICAL UNIVERSITY DRESDEN
Institute of Power Engineering
Training Reactor



Reactor Training Course

Experiment

"Gamma Dosimetry and Dose Rate Determination"



Instruction for
Experiment "Gamma Dosimetry and Dose Rate Determination"

Content:

Figures:

(issued: January 2019)

1. Motivation

The experiment aims on familiarising with the methods of calibrating different detectors for determination of the dose and the dose rate. Furthermore, the dose rate and the activity in the vicinity of an enclosed source of ionising radiation (Cs-137) will be determined taking into account the background radiation and the measurement accuracy. Additionally, the experiment focuses on the determination of the dose rate of a shielded source as well as on the calculation of the required thickness of the shielding protection layer for meeting the permissible maximum dose rate. The dose rate at the open reactor channel of the AKR-2 will be measured.

2. Theoretical Background

2.1. Properties of Ionising Radiation and Interactions of Gamma Radiation

The radiation emitted along with the transformation of radionuclides is the outward characteristic of radioactivity. The radiation is categorised as α-radiation (He-4 nuclei), ß$^+$- and ß$^-$-radiation (positrons and electrons, respectively) and γ-radiation. As a by-product of such transformations or as a result of them conversion electrons and X-rays can be observed. In some cases of nuclear reactions, also neutrons are released that will interact with matter, too.

Gamma radiation is high-energy electromagnetic radiation that are emitted in form of photons with a discrete energy distribution along with a transition of an atomic nucleus from an excited state into a lower or the ground state. Simply speaking, the origin of gamma radiation corresponds to the origin of visible light, which, in contrast, is the result of transitions between states of the atomic shell.

The interaction of gamma radiation with matter differs significantly from that of charged particles with matter. Whereas α- and ß-radiation lose energy by ionisation or radiation slowing-down processes, γ-rays lose energy mainly by the photo effect, Compton scattering, and pair production.

The attenuation of gamma radiation, i.e. the intensity I after penetration of a layer with the thickness x consisting of absorber material relative to the intensity I_0 without any absorber, can be described via the attenuation equation $I = I_0 \cdot \exp(-\mu x)$ with μ being the linear attenuation coefficient. This coefficient is the arithmetic sum of those coefficients which quantify the three independently appearing kinds of absorption interactions of gamma radiation, i.e. $\mu = \mu_{co} + \mu_{Ph} + \mu_{Pa}$ (Fig. 1).

The quantity μ_{co} represents the Compton scattering (also called: Compton effect), which is the transfer of energy from a photon to an electron of the atomic shell by collision. The gamma quant is redirected, i.e. scattered. This process goes along with a defined loss in energy dependent on the angle (being between 0 ... 180°). The emitted electron is called a Compton electron. Compton electrons have a continuous energy spectrum.

The quantity μ_{Ph} represents the photo effect, which is the transfer of the entire energy of a photon to an electron of the atomic shell and hence, the photon vanishes. The electron (photo electron) is emitted out of the atomic shell with the total energy of the extinguished photon minus the ionisation energy.

The quantity μ_{Pa} represents pair production, which is the transformation of a photon to an electron-positron pair. This effect appears only at photon energies higher than 1.02 MeV (two times the rest mass of an electron = 2 · 0.51 MeV).

The percentage of each of these effects depends essentially on the energy of the photon and the atomic number of the absorber (Fig. 1).

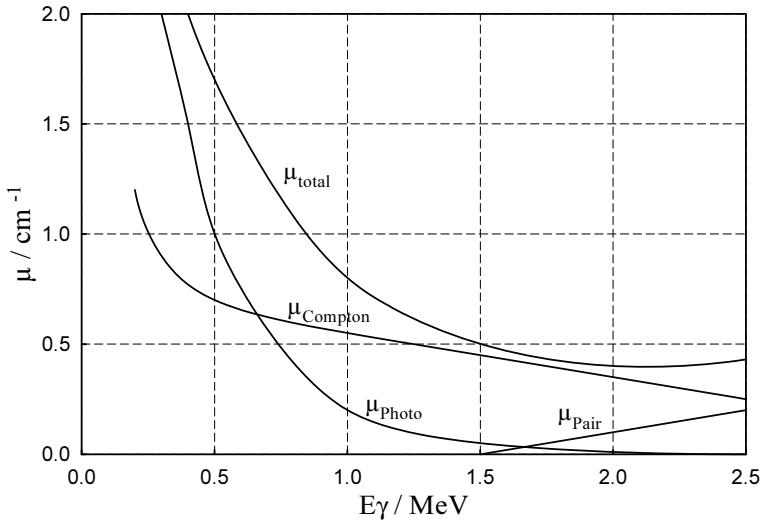


Fig. 1, Composition of the attenuation coefficient μ of γ-radiation in lead

The dependence on the atomic number is:

for the photo effect	proportional to Z^4 ... Z^5
for Compton scattering	proportional to Z
for pair production	proportional to Z^2

From these relations, the suitability of a special absorber material for certain gamma energies can be assessed. The choice for a particular material is determined not only by radiation protection reasons but also by economic, technical and low-weight criteria (e.g. in case of protective clothing).

The electrons which are released by each of the three processes carrying different amounts of energy cause ionisation of the matter which they penetrate. This phenomenon is the basis of devices for the determination of the γ-energy. It is also the reason for the biological effectiveness of γ-radiation.

2.2. Detection of Ionising Radiation

Radiation measuring devices recognise the existence of ionising radiation and determine the kind, the energy, the intensity, and the direction of the radiation. For this purpose, a variety of measurement devices have been developed. The training experiment at AKR-2 focuses on dose and dose rate measurement devices. The ionisation chamber is an important detector for this purpose. It has a simple concept, can be applied in various geometries and dimensions, and can be adjusted according to a particular radiation in a wide range.

Design and method of operation of an ionisation chamber:
In the simplest case, an ionisation chamber is a cylinder filled with air or an inert gas. It is equipped with two insulated plate electrodes which are connected to a power supply over a resistance R (Fig. 2).

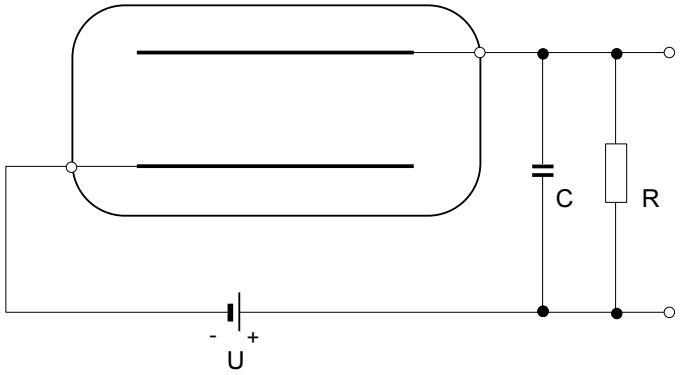


Fig. 2, Design of an ionisation chamber

Inside the chamber, a charged particle generates pairs of ions, whose number per unit length depends on the energy and the kind of the particle, on the type of the gas, and on the pressure of the gas. Due to the applied electric field these charge carriers migrate to the electrodes and cause a chamber current i, which increases for increasing chamber voltage. For large numbers of particles, an average chamber current appears whose magnitude is a measure for the number of incoming particles. Such an ionisation chamber that works on the current is used especially for detection of beta and gamma radiation.

2.3. Quantities and Units of Dosimetry

Although the unit of the **energy dose D** (unit: 1 Gray = 1Gy) was introduced respecting the biological effectiveness, exposures of a certain energy dose by various kinds of radiation can have widely differing biological effects. This property of the radiation can be quantified by the **radiation weighting factor w_R** (Tab. 1). The values of the radiation weighting factor w_R depend on the type and quality of the external radiation field or on the type and quality of the radiation emitted from an incorporated radionuclide. In this context, the quantity of the **dose equivalent H** is introduced, whose unit is Sievert [Sv].

Type and energy range	Radiation weighting factor w_R
X-, ß- and γ-radiation	1
Protons	2
Neutrons (energy dependent)	ca. 2.5 - 20
α-radiation, fission products	20

Tab. 1, Radiation weighting factor w_R (StrlSchV Annex 18, Part C)

The dose equivalent H and the energy dose D are related as follows:

$$H = D \cdot w_R$$

Because for X-, γ- and ß-radiation the radiation weighting factor is $w_R = 1$, for these (and only these!) kinds of radiation is:

$$1\ Gy = 1\ Sv$$

All of these relations are independent on the time of irradiation.

The **effective dose E** is defined as the sum over all organs T

$$E = \Sigma\ (w_T \cdot H_T)$$

with H_T - organ absorbed equivalent dose
w_T - tissue weighting factor for the respective organ or tissue T

The values of the tissue weighting factors for particular organs or tissues are given in Tab. 2.

tissues or organs	tissue weighting factors w_T
red bone marrow, colon, lung, stomach, chest	each 0.12
gonads	0.08
bladder, oesophagus, liver, thyroid gland	each 0.04
skin, bone surface, brain, salivary gland	each 0.01
other organs or tissues (together)	0.12

Tab. 2, Tissue weighting factors w_T (StrlSchV Annex 18, Part C)

The most important quantities in radiation protection are summarised in Tab. 3.

The application of scientifically reasonable measures in radiation protection requires profound knowledge about the interaction between radiation and biological tissue. For the handling of sources of ionising radiation, the Radiation Protection Act (German: Strahlenschutzgesetz, StrlSchG) and the Radiation Protection Ordinance (German: Strahlenschutzverordnung, StrlSchV) define maximum doses for certain time periods. The values of these dose limits vary for various organs, parts of the body, and for whole body, as well as for occupationally radiation exposed people, particular groups of the population, and the whole population. Moreover, as a general principle the radiation exposure has to be kept **a**s **l**ow **a**s **r**easonably **a**chievable (ALARA principle). This principle has to be kept also for radiation exposures below the legal limits.

According to the German Radiation Protection Act the individual personal exposure to radiation per one calendar year must not exceed the following limits (in mSv):

Values in mSv	Occupationally exposed person		Individual members of the general public (due to the work with radioactive substances)
	Category A	Category B	
Effective dose (whole body)	20 [1)]	6	1 [2)] or 0.3 [3)], resp.
Eye lens	20	15	15
Skin, hands, feet, forearms	500	150	50

[1)] exceptionally an effective dose of 50 mSv for a single year can be permitted, where 100 mSv for five consecutive years may not be exceeded, in maximum 400 mSv during the whole professional life

[2)] direct irradiation **and** discharge of radioactive substances (air and water)

[3)] only discharge of radioactive substances (air and water)

Moreover:

- people under 18 years	1 mSv/a	
but: trainees and students	6 mSv/a	insofar as this is required for the purpose of their training
- women of child-bearing age	2 mSv/month	at the uterus
- during pregnancy altogether	1 mSv	due to the occupation of the mother

Through the Radiation Protection Ordinance, the legislative body requires that the rooms being used for handling sources of ionising radiation have to be classified into radiation protection areas. These are

a) supervised areas,
b) controlled areas,
c) exclusion areas

Fig. 3 shows the permissible thresholds of the various radiation protection areas.

Value	Definition	Legal unit	Former unit	Conversion
Activity	number of radioactive transformations per time unit	Bequerel 1 Bq = 1 s^{-1}	Curie	1 Ci = 3.7 $\cdot 10^{10}$ Bq
Energy dose	total amount of absorbed energy per mass element divided by its mass	Gray 1 Gy = 1 J / kg	rad	1 rad = 0.01 Gy
Dose equivalent	energy dose multiplied by the radiation weighting factor of the respective kind of radiation	Sievert 1 Sv	rem	1 rem = 0.01 Sv
Energy dose rate	energy dose per time unit	Gy / s	rad / s	1 rad / s = 0.01 Gy / s
Dose equivalent rate	dose equivalent per time unit	Sv / s	rem / s	1 rem / s = 0.01 Sv / s

Tab. 3, Dosimetric quantities and units

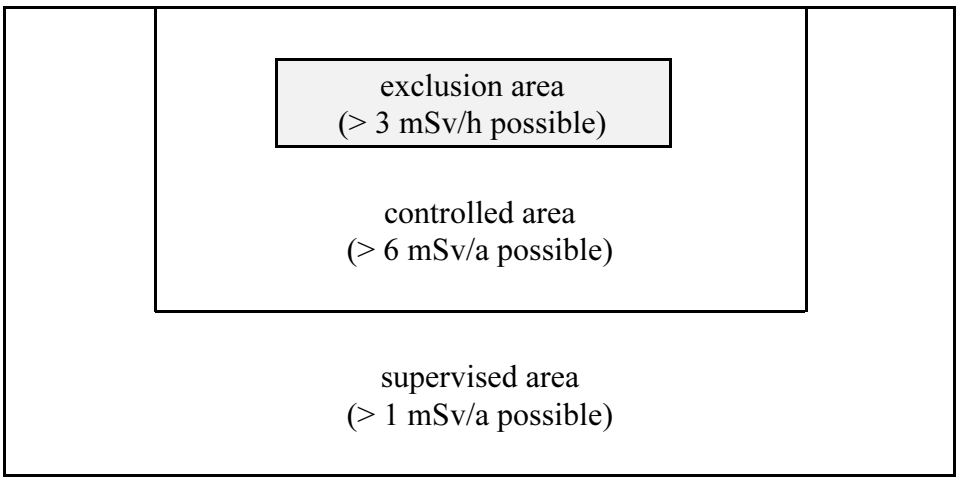


Fig. 3, Classification and legal limit values of radiation protection areas

The definitions of the limits of controlled areas and exclusion areas base on a 40-hours work-week with 50 workweeks per calendar year, as far as no other causal details about the duration of stay are given.

3. Procedure of the Experiment

For handling of radioactive substances or of sources of ionising radiation, the following principles of radiation protection need to be kept with care!

1. No one is allowed to be subject of uncontrolled radiation exposure (personal dosimetry control).

2. The group of people being exposed to radioactivity needs to be kept as small as possible.

3. The duration of exposure needs be kept as short as possible (work organisation).

4. The distance from the radiation source needs to be kept as large as possible (manipulators, pincers).

5. Use shieldings as far as possible.

6. Tidiness and cleanness

7. Exceptional events have to be reported to the radiation protection officer.

3.1. Commissioning and Calibration of the Dosimeter Thermo FH40G

1. Switch-on the device.
2. Wait until the end of the self-test.
3. The device is ready in case of no error messages.

3.2. Commissioning and Calibration of the Dosimeter Berthold LB 133-1

Screw the counter tube into the face area of the device until the end stop is reached. Press the large red push-button. At normal natural background radiation, the device indicates a dose rate of (0.1 ... 0.2) µSv/h.
Then, the battery status has to be checked. For this purpose, press the respective button below the handle bar, the pointer should stand in the area with black background. Fully charged batteries allow service for about 8 hours.
For changing the measuring range use the respective buttons below the handle bar. By pressing the button "Zeitkonstante" the standard time constant can be increased by a factor of 10 from 2 s to 20 s. For locating radiation fields and for overview measurements, the smaller time constant is recommended, whereas the large time constant is used for more precise measurements and for the measuring in the most sensitive range "x1".
The device has an adjustable alarm threshold. When exceeding the previously set threshold, the device gives an acoustic alert over a time of about 4 s. The alert is reactivated only after the display value has fallen below the threshold. Though, the indicator lamp blinks as long as the threshold is exceeded. Note that the value for stopping the alert is about 20 % lower than the threshold's scale value (hysteresis). The acoustic alert can neither be deactivated nor be changed in its volume.

3.3. Commissioning and Calibration of the Dosimeter STEP RGD 27091

1. Switch-on the device.
2. A check of the battery status is not necessary, because the device performs a self-test and automatically reports a low battery status (optically).
3. Put the switch to zero adjustment ("Nullabgleich").
4. Put the switch "Messbereich" to 20 and set the zero-balance to a display value <0.5 using a potentiometer.

3.4. Setup of the Experiment

The setup of the experiment is given in Fig. 4.

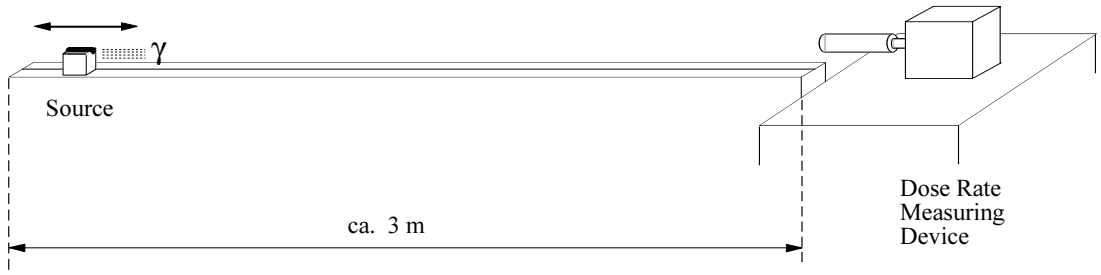


Fig. 4, Setup of the experiment

After calibrating the devices, the three detectors are placed at the intended spot on the measurement table. Only shortly before the measurement, the source is taken out of the shielded lead container (use pincers!) and mounted onto the slide (the distance between slide and measurement table is about 3 m).

For the shielding investigations the shielding material has to be placed directly in front of the source.

3.4.1. Measurement of Dose Rate in Various Distances from the Radiation Source

The measurements are carried out with all three measuring devices. The radiation source has to be placed in those distances from the detectors that are given in the example protocol (Tab. 4). The opening of the source has to point into the direction of the detectors.
The measurements start with the largest distance. The whole series of measurements will be repeated afterwards. For positioning of the measurement devices high accuracy is required.

Getting measured values:

Thermo FH40G:	direct reading on the display
LB 133-1:	choose the larger time constant and the largest measuring range, wait until the setting time is over (about 30 s)
STEP RGD 27091:	direct reading on the display

3.4.2. Measurement of Dose Rate behind Radiation Shielding

The radiation attenuation coefficient μ has to be determined for the shielding materials heavy concrete, light concrete, and lead. Its impact has to be discussed with relation to radiation protection. For this purpose, the shielding of the respective material has to be placed between the source and the detector. From the measured values, the attenuation coefficient can be calculated (Tab. 5).

3.5. Measurement of Dose Rate at the open Reactor Channel

1. Mount the spherical chamber on the device
2. Switch-on the dosimeter
3. Mount the spherical chamber in such a way that the middle of the chamber is positioned according to the distance tick marks
4. Insert the measurement stick as far as possible into the reactor channel (attention: Do not stand directly in front of the channel opening!)
5. Pull-out the stick stepwise according to the respective tick marks
6. The distances which have to be set for measurement are given in the example protocol (Tab. 6)

4. Evaluation of Measuring Results

1. Plot the function P_x in dependence on distance r using the values measured with the dosimeters (use log-log scales!)

2. Calculate the function $P_x = K \cdot \frac{A}{r^2}$ from the given activity of the source Cs-137 and compare the result with the measured values

   with P_x - dose rate [mSv/h] due to a point-like gamma emitter in air

   A - activity [Bq]
   A (Febr. 1993) = $0.26 \cdot 10^9$ Bq ($T_{1/2}$ = 30 a)

   K - dose constant [$mSv \cdot m^2 / (h \cdot GBq)$]
   for Cs-137 is K = 0.0925 $mSv \cdot m^2 / (h \cdot GBq)$

   r - distance to the source [m]

3. Discuss the appropriateness of each of the detectors and also the reasons for the differences in the results

   Some specifications of the measurement devices:
   - **STEP RGD 27091:**
     Indication error < 5 %, the detector is an ionisation chamber
   - **Thermo FH 40G** and **Berthold LB 133-1:**
     Indication error < 15 %, very low energy dependence, the detector is an ionisation chamber

4. Calculate the linear attenuation coefficient from the values of the dosimeters for all three shielding layer thicknesses using $P_x = B_D \cdot P_{x0} \cdot e^{-\mu x}$ with P_{x0} being the dose rate at the place of measurement without any shielding

5. Calculate the half-value thickness $x_{1/2}$ [cm] with $x_{1/2} = \frac{\ln 2}{\mu}$

6. Calculate the mass-attenuation coefficient μ' [cm^2/g] using $\mu' = \frac{\mu}{\rho}$

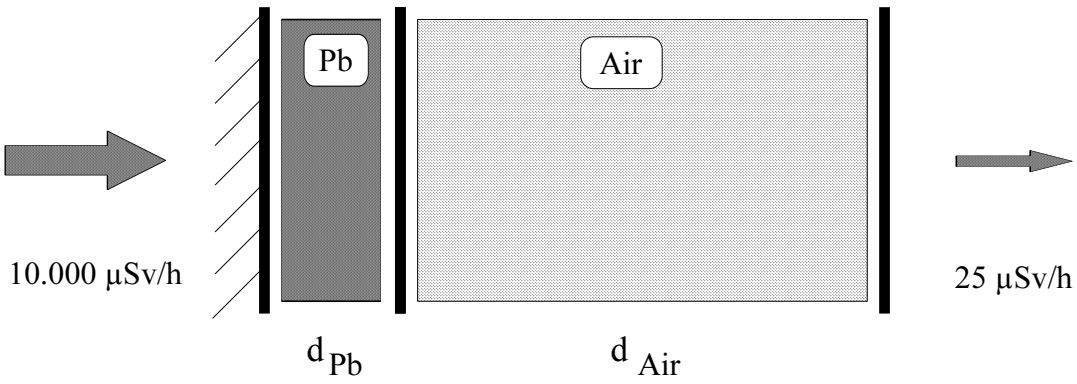
   with $\rho = 0.6$ g/cm^3 (light concrete)
   $\rho = 2.5$ g/cm^3 (heavy concrete)
   $\rho = 11.7$ g/cm^3 (lead)

Remarks regarding the dose build-up factor B_D:
The dose build-up factor is defined as the ratio between the sum of the dose rate of all photons (scattered and non-scattered) and the non-scattered photons.
It depends on the energy of the photons, the shielding layer thickness, the radiation protection material, and the measurement geometry. By means of the build-up factor, calculations become possible also for wide spreaded beams as well as for extended or thick absorbers via the attenuation factor.
For the shielding measurements in the experiment, the dose build-up factor B_D can be neglected (i.e. $B_D \approx 1$), because the Compton quants are scattered away (from the detector) and do not contribute to the dose rate behind the shielding in the used setup.

7. Calculate the required shielding layer thickness to meet a gamma dose rate of 50 mSv/a (or more precisely, of about 25 mSv/h) in a distance of half a meter from the source (use the mean value of μ!). Use lead for the shielding material!

$$D = D_0 \cdot e^{-\mu_{air} \cdot d_{air}} \cdot e^{-\mu_{Pb} \cdot d_{Pb}}$$

$$d_{total} = d_{Pb} + d_{air} = 0{,}5 \text{ m}$$



8. Discuss the impact of the shielding material on the design of the radiation protection facilities

9. Plot the dose rate over the length of the AKR-2 tangential channel

Distance [m]	RGD27091 1. meas. [µSv/h]	RGD27091 2. meas. [µSv/h]	FH40G 1. meas. [µSv/h]	FH40G 2.meas. [µSv/h]	LB 133 1.meas. [µSv/h]	LB 133 2.meas. [µSv/h]
0.05						
0.1						
0.2						
0.3						
0.4						
0.5						
0.6						
0.8						
1.0						
1.2						
1.5						
2.0						

Tab. 4, Investigations regarding the distance law for various detectors

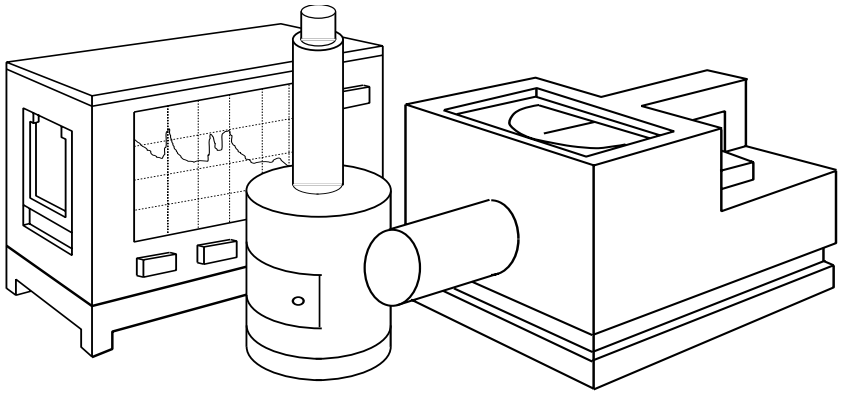
	RGD27091 1. meas. [µSv/h]	RGD27091 2. meas. [µSv/h]	FH40G 1. meas. [µSv/h]	FH40G 2. meas. [µSv/h]	LB 133 1. meas. [µSv/h]	LB 133 2. meas. [µSv/h]
light concrete (20 cm)						
heavy concrete (20 cm)						
heavy concrete (40 cm)						
lead (5 cm)						

Tab. 5, Protocol for the determination of the mass attenuation factor (example)

Distance [m]	Gamma dose rate [µSv/h]
1.3	
1.25	
1.2	
1.1	
1.0	
0.9	
0.8	
0.7	
0.6	
0.5	
0.4	
0.3	
0.2	
0.1	
0.0	

Tab. 6, Protocol for the determination of the gamma dose rate at the open reactor channel (example)

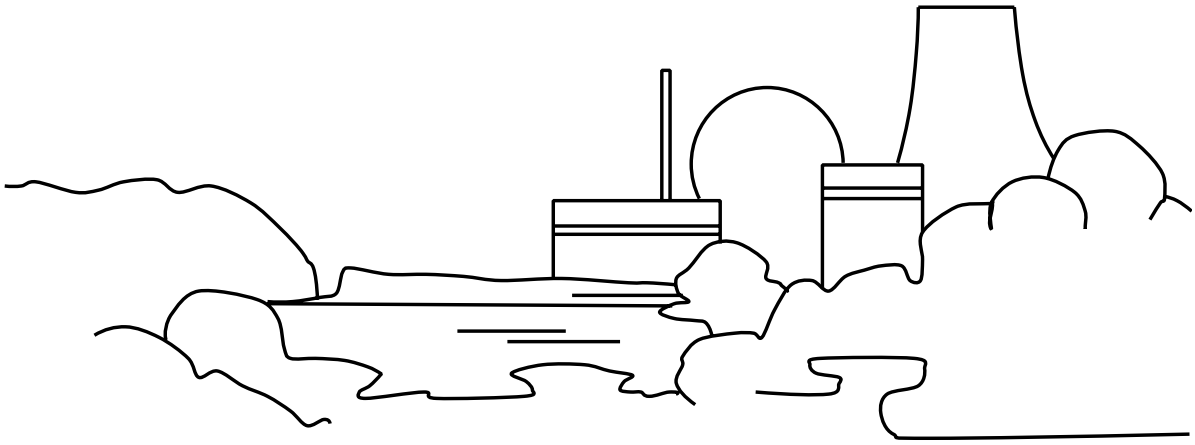
TECHNICAL UNIVERSITY DRESDEN
Institute of Power Engineering
Training Reactor



Reactor Training Course

Experiment

"Activation and Decay of Radioactive Nuclides"



Instruction for
Experiment "Activation and Decay of Radioactive Nuclides"

Content:

Figures:

(issued: March 2015)

1. Introduction

In the year 1896, the French physicist Henry Becquerel noticed an unknown radiation emitted by uranium salts that resulted in the blackening of a photo plate. Further work by Becquerel, along with Pierre and Marie Curie found the radioactivity of thorium and eventually lead to the discovery of the chemical elements polonium and radium. However, only the first artificially induced nuclear reactions successfully pioneered by Ernest Rutherford by bombardment of nitrogen nucleons with α-particles, made possible the fabrication of artificial radionuclides with various properties. Since then, particularly in the second half of the 20th century, the significance of applied radioactivity methods has risen enormously.

Currently, artificial radioactive isotopes are being produced in any desired quality and quantity, mostly in nuclear reactors by the irradiation with neutrons originating from nuclear fission. Furthermore, advanced radiation measurement technologies are now an inherent part of science, technology and medical applications. Typical examples are the radiation-based measurements of thickness and of fluid level, as well as radiographic investigations, medical irradiations and radioactive tracer applications. However, these methods exhibit some characteristic drawbacks as well. As such, specific safety precautions are necessary and need to be thoroughly applied and controlled in order to prevent severe radiation damages.

The purpose of the experiment to be conducted in this session is to demonstrate the fundamentals of the production of radioactive isotopes in nuclear reactors and to show some of their basic properties as well as their appropriate handling. Therefore, selected materials will be activated in the reactor and subsequently, the decay of these materials will be measured. In one more part of the experiment, an unknown material will be irradiated and identified subsequently by regarding its decay curve and determination of its half-life. The latter experiment exemplifies one of the many possibilities offered by applied radioactivity.

2.2. Fundamentals of the Experiment

2.1. Important Terms regarding Radioactivity

The chemical elements of the periodic table are characterised by a specific atomic number (number of protons) and the associated number of electrons in the atomic shell. Additionally to protons, the atomic nucleus includes neutrons. Both, neutrons and protons are termed nucleons. For any chemical element with its characteristic numbers of protons, atomic nuclei may exist that include a varying number of neutrons. These nuclei are called isotopes of this element. They appear at the same place in the periodic table and exhibit the same chemical properties, but different atomic masses. For most elements there are several isotopes that can or cannot be radioactive, respectively.
The term nuclide is used for any arbitrary atomic nucleus, characterised by both, its atomic and its mass number. If a nuclide is radioactive, it is termed a radionuclide. A radioactive isotope is termed radioisotope.

To summarise, radionuclides are radioactive nuclei of any chemical element of the periodic table. On the contrary, radioactive isotopes are invariably associated with a specific chemical element.

Thus, $^{28}_{13}Al$ and $^{66}_{29}Cu$ are two radionuclides, whereas $^{28}_{13}Al$ is a radioisotope of aluminum and $^{66}_{29}Cu$ is a radioisotope of copper.

Radioactivity is the feature of particular nuclei which can change their nucleus composition without any exterior influence. Whether or not this change occurs spontaneously depends on the stability of the nucleus, which is, among others, a function of the ratio between its number of protons and its number of neutrons. This is why for different isotopes of one chemical element some may be radioactive, while others are not. Radioactive nuclei exhibit an energy abundance, which they may release by emission of radiation and accompanying nucleus transformation (decay). These nuclei are unstable. Radiation emission cannot be controlled or manipulated by exterior influences.

The rate of nucleus transformations can be measured and is used as a means to quantify its radioactivity. According to the SI standard (Système International d'Unités), use of the unit Becquerel is mandatory:

$$1 \text{ decay / s} = 1 \text{ Bq}$$

In some cases, the historical unit 1 Ci (Curie) can still be found that was established based on the decay of 1 g of Radium-226 ($1 \text{ Ci} = 3.7 \cdot 10^{10} \text{ Bq}$), but is not used anymore.

The most important radiophysical characteristics of radionuclides, crucial for allocating their potential application, are the following:

a) type of the emitted radiation:
- α-emitter (emission of $^{4}_{2}He$ nuclei)
- $ß^{+}$-emitter (emission of positrons)
- $ß^{-}$-emitter (emission of electrons)
- γ-emitter (emission of electromagnetic radiation)

b) energy of the emitted radiation:
In nuclear physics the measuring unit electron volt (eV) has been established particularly for radiation energy. 1 eV is the energy gained by an electron traversing an acceleration voltage of 1V, thus:

$$1 \text{ eV} = 1.602 \cdot 10^{-19} \text{ As} \cdot 1 \text{ V} = 1.602 \cdot 10^{-19} \text{ Ws}$$

c) half-life:
The decay of radioactive nuclides results in the production of new nuclides, which can be either stable or radioactive. The half-life $T_{1/2}$ denotes the time after which exactly half of the original amount of the radioactive material still exists, while the other half has decayed. The half-life is constant and characteristic for every radionuclide.

2.2. Production of Radioactive Nuclides

Radioactive nuclides originate from stable nuclei due to nuclear reactions if the resulting nucleus exhibit abundant energy, which is released either by radiation emission or emission of a particle. The material to be irradiated is called the target. Protons, neutrons, deuterons, alpha particles or gamma quants are commonly used as bombarding species.

Mostly, the following nuclear reactions are used for the production of artificial radioactive nuclides:

a) Exchange reaction: the bombarding particle x penetrates the target nucleus, leading to the emission of a different particle y. If x and y are differently charged, a transformation of the chemical element occurs.

b) Capture reaction: The bombarding particle x is absorbed in the target nucleus. The abundant energy is being released as electromagnetic γ-irradiation.

c) Fission reaction: By bombardment with neutrons or high-energy charged particles, atomic nuclei, particularly heavy ones, are being split in two different pieces of similar masses. This continuously occurs as a chain reaction in nuclear reactors. In this case, the production of radionuclides (radioactive waste) is not desired.

The mechanism of nuclear reactions can be assessed in two steps: In the first step, the bombarding particle is being absorbed in the nucleus. The binding energy of the bombarding particle is released and, additionally, its kinetic energy is transferred to the target nucleus. Thereby, a highly instable, highly excited temporary nucleus is being generated (compound nucleus). In the second step, this compound nucleus relaxes within a very short time ($<10^{-17}$ s) by emission of either a particle or a γ-quant or by splitting.

Nuclear reactions are represented by the following reaction equation

$$x + A \rightarrow C^* \rightarrow B + y \quad \text{or shorter} \quad A(x,y)B$$

with the bombarding particle x, the target nucleus A, the temporary compound nucleus C*, the nucleus B remaining after the reaction and the particle y emitted due to the decay.

An example of such a nuclear reaction is the following.

$$^{14}_{7}N + ^{4}_{2}\alpha \rightarrow ^{18}_{9}F^* \rightarrow ^{17}_{8}O + ^{1}_{1}p \quad \text{resp.} \quad ^{14}N(\alpha,p)^{17}O$$

In a nuclear reactor radioactive nuclides are essentially produced by capture of thermal neutrons. Thermal neutrons have a mean energy of 0.025 eV, being equivalent to a speed of 2200 m/s at room temperature.

The capture reactions with thermal neutrons occur as follows:

$$n_{th} + {}^{M}_{Z}A \rightarrow {}^{M+1}_{Z}A^{*} \rightarrow {}^{M+1}_{Z}A + \gamma \qquad \text{resp.} \qquad {}^{M}_{Z}A(n_{th},\gamma){}^{M+1}_{Z}A$$

with the mass number M and the atomic number Z. The same element A thus remains, but a different isotope is created which mass number M is by one higher than that of the original isotope. Since the proton/neutron ratio changes, the created isotopes are usually radioactive.

For the materials used in the experiment, aluminum and copper, the following representations are applicable:

$$^{27}_{13}Al(n_{th},\gamma)^{28}_{13}Al$$

Natural copper consists of 2 different isotopes, $^{65}_{29}Cu$ (30.9%) and $^{63}_{29}Cu$ (69,1%). Both isotopes are being activated as follows:

$$^{65}_{29}Cu(n_{th},\gamma)^{66}_{29}Cu \qquad \text{resp.} \qquad ^{63}_{29}Cu(n_{th},\gamma)^{64}_{29}Cu$$

2.3. Decay of Radioactive Nuclides

Stability and instability of an atomic nucleus depend on its respective composition, determining the type and rate of its decay. The most important properties of a radionuclide can be found in tables for any isotope. These are (as already mentioned): type of emitted radiation, energy of emitted radiation and half-life.

A graphical decay scheme provides a good overview (Fig. 1). It illustrates the type of radiation, the respective loss of energy, the change of the atomic number as well as the relative probabilities of possible decay paths.

In the decay scheme, vertical distances represent changes in energy. Horizontal shifts represent a change of charge, which are positive in case of a shift to the right (i.e. transformation to a chemical element with a higher atomic number e.g. by nuclear electron emission), and negative in case of a shift to the left (i.e. transformation to a chemical element with lower atomic number e.g. by nuclear emission of a $ß^{+}$-particle). The energy levels of the nuclei are represented by horizontal lines. The amount of energy change is indicated at the right of the energy levels either in units of keV or MeV.

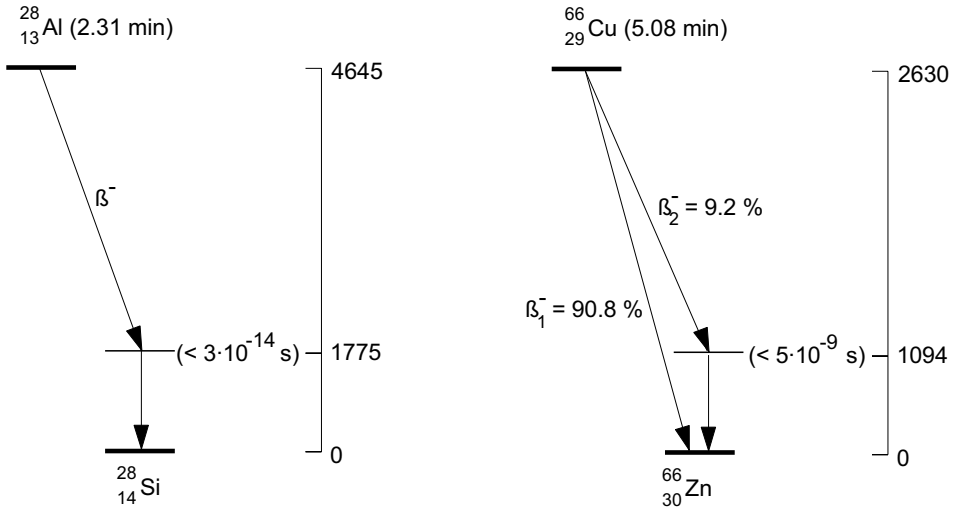


Fig. 1, Examples of decay schemes (for $^{28}_{13}Al$ and $^{66}_{29}Cu$)

2.4. Mathematical Description of the Production and Decay of Radionuclides

Initially, the number of capture reactions per time and volume units occurring in the activation material need to be quantified. The number of activations per second occurring in a cubic centimeter of sample material is called the activation rate **R** and can be determined according to the following equation

$$R = N \cdot \sigma \cdot \Phi \qquad \textbf{(1)}$$

with **N** being the number of atomic nuclei per cm³ in the activation material. N can be determined according to:

$$N = N_L \frac{\rho}{AG} \qquad \textbf{(2)}$$

with the LOSCHMIDT constant $\mathbf{N_L} = 6.025 \cdot 10^{23}$ mol $^{-1}$, the density ρ, and the atomic mass **AG** of the activation sample.

The microscopic cross section σ is an expression for the probability of the occurrence of a nuclear capture reaction of a neutron. σ may be understood as the interaction area of a nucleus facing the incident neutron. However, this area is not equivalent to the geometric cross section of the nucleus, it can be orders of magnitude bigger or smaller. If a neutron hits this area, a nuclear reaction occurs. Since σ is related to the facing area of a nucleus, it is measured in units of 10^{-24} cm². This unit is called barn and abbreviated with b (1b = 10^{-24} cm²). The exact value of σ is determined through evaluation of the probability of the nuclear reaction which is, among others, a function of the velocity (i.e. the energy) of the incident neutron.

A cross section exists for any type of nuclear reaction. Thus, in addition to cross sections for activation (i.e. neutron capture) reactions, cross sections can be defined e.g. also for elastic or inelastic scattering or fission. Finally, the total cross section is the sum of all partial cross sections associated with all possible nuclear reactions.

The product $N \cdot \sigma$ is called macroscopic cross section Σ. In contrary to σ its unit is a reciprocal length and the reciprocal value of Σ is the mean free path **l** of a neutron in the sample material.

$$\Sigma = N \cdot \sigma \qquad\qquad l = \frac{1}{\Sigma} \qquad\qquad \textbf{(3)}$$

Φ is the neutron flux density that accounts for the number of neutrons crossing an assumed area of 1 cm² per second at the place of irradiation, i.e. the measuring unit for the neutron flux density is thus $1 / (cm^2 \cdot s)$.

In the core of a thermal nuclear research reactor, the neutron flux density is on the order of 10^7 $1/(cm^2 \cdot s)$ for zero-power reactors and about 10^{13} ... 10^{15} $1/(cm^2 \cdot s)$ in more powerful research reactors.

In order to calculate the number of radioactive nuclei that exist after the irradiation time t_{irr} it would be wrong simply to multiply the activation rate R with t_{irr}, because in the same time when new activations occur already previously activated nuclei can decay. Thus, it is necessary to take into account both, creation and decay of radioactive nuclei simultaneously.

First, the pure decay behaviour of an existing activated nuclide will be discussed.

K shall be the number of existing radioactive nuclei in a sample. The activity **A** is defined as the number of decays occurring in a time unit, which is thus equivalent to the decrease of existing radioactive nuclei K per time unit dt. Atomic decay reactions are distributed statistically, and consequently, the activity is proportional to the number of existing nuclei K:

$$A = -\frac{dK}{dt} = \lambda \cdot K \qquad \textbf{(4)}$$

This is the differential equation of the radioactive decay with the decay constant λ, which is associated with the half-life $\mathbf{T_{1/2}}$ as:

$$\lambda = \frac{\ln 2}{T_{1/2}} \qquad \textbf{(5)}$$

The solution of the differential equation (4) by separation of variables yields the well known decay law

$$K(t) = K_0 \cdot \exp(-\lambda \cdot t) \qquad \textbf{(6)}$$

with K_0 being the number of radioactive nuclei at the beginning (here: for $t_{decay} = 0$ at the end of irradiation).

In the next step, it shall be calculated how the production of radioactive nuclides during irradiation occurs. For that purpose, the decay law needs to be coupled with the activation reaction rate. Thus, in the decay differential equation one term has to be added that accounts for the constant additional supply of radioactive nuclei, which is the activation rate R:

$$\frac{dK}{dt} = -\lambda \cdot K + \Sigma \cdot \Phi \qquad \textbf{(7)}$$

In this case, the initial condition is K(t=0) = 0, which signifies that no radioactive nuclei exist before irradiation. The solution of this differential equation is:

$$K(t) = \frac{\Sigma \cdot \Phi}{\lambda} [1 - \exp(-\lambda \cdot t)] \qquad \textbf{(8)}$$

According to equation (4), $A(t) = \lambda \cdot K(t)$, which can be introduced in equation (8), it can also be written in terms of the activity:

$$A(t) = \Sigma \cdot \Phi \cdot [1 - \exp(-\lambda \cdot t)] \qquad \textbf{(9)}$$

As obvious from this equation, the term in brackets approaches to 1 for continuing irradiation. Thus, a limiting maximum value for the activity exists, which is called the saturation activity $\mathbf{A_S}$:

$$A(t \rightarrow \infty) = A_s = \Sigma \cdot \Phi \qquad \textbf{(10)}$$

That means, when the saturation activity A_S is attained, the rate of nucleus production is equal to the rate of nucleus decay. It can easily be shown that after an irradiation time equivalent to 5 half-times of the activation material, the obtained activity already accounts for 97 % of the saturation activity. In most applications, it is thus not useful to extend irradiation beyond this duration.

Now, time periods are introduced to label the time regimes during irradiation and activity decay. $\mathbf{t_{irr}}$ should be the irradiation time and $\mathbf{t_{decay}}$ the time of activity decay after end of irradiation. The number of radioactive nuclei at the end of the irradiation period can be calculated according to equation (8):

$$K(t_{irr}) = \frac{\Sigma \cdot \Phi}{\lambda} [1 - \exp(-\lambda \cdot t_{irr})]$$

To determine the number of active nuclei during the activity decay period, this expression has to be inserted as K_0 in equation (6). The number of active nuclei after irradiation is thus

$$K(t_{decay}) = \frac{\Sigma \cdot \Phi}{\lambda} [1 - \exp(-\lambda \cdot t_{irr})] \cdot \exp(-\lambda \cdot t_{decay})$$

and, finally, in terms of the activity

$$A(t_{decay}) = \Sigma \cdot \Phi \cdot [1 - \exp(-\lambda \cdot t_{irr})] \cdot \exp(-\lambda \cdot t_{decay}) \quad \textbf{(11)}$$

The time behaviour of the activation process and the subsequent radioactive decay after the end of irradiation is presented in Fig. 2.

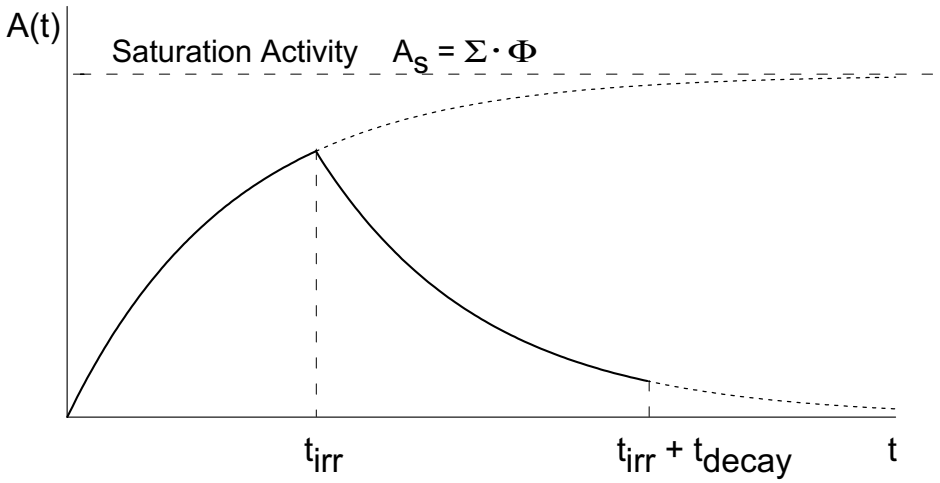


Fig. 2, Time dependence of the activation and the decay of a radionuclide

2.5. Neutron Self-absorption

For calculation of the reaction rate in the material to be activated, equation (1) can only be applied accurately, if the neutron flux density does not decrease within the sample. However, this is the case in most applications as shown schematically in Fig. 3. The figure has to be understood in a way that the incident neutrons impinge perpendicularly on the plate from the left. If solely absorption and no scattering reactions take place the following relation for the decrease of neutron flux density is applicable:

$$\Phi(x) = \Phi_0 \cdot \exp(-\Sigma \cdot x) \quad \textbf{(12)}$$

with Σ being the macroscopic cross section for absorption and **x** the distance from the plate surface.

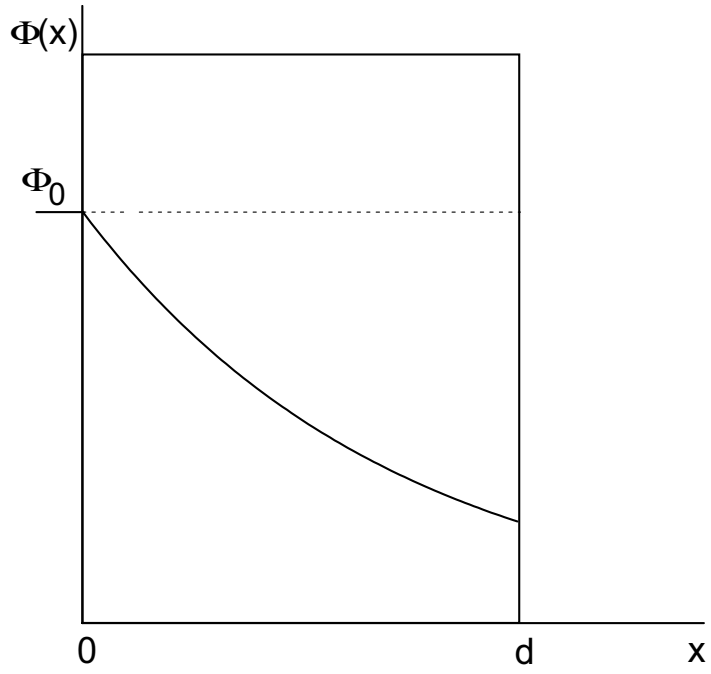


Fig. 3, Decrease of neutron flux density inside of a neutron absorbing material of thickness d (neutron self-absorption)

Such absorptions always occur during irradiation experiments since the extent of the irradiated samples is never infinitesimal small in any direction. Furthermore, calculation of the shielding effect for irradiation applications usually proves to be even more complicated than equation (12) suggests since the incident neutron flux exhibits a non-homogeneous angular distribution. Also, the sample shape may differ from easily describable geometries such as flat plates.

In general, it can be concluded that neutron self-absorption effects are significant if

a) the macroscopic cross section is high and/or
b) the samples are very thick.

3. Experimental Procedure

3.1. Choice of Materials to be activated

As mentioned above, three radiological properties are decisive for the choice of materials for irradiation: the type of emitted radiation, the radiation energy and the half-life of the radionuclide. Additionally, other practical issues can influence the choice such as easy handling, temperature resistance, availability, solubility, and low toxicity.

In the experiment carried out here disc-shaped samples of 1 mm thickness and approximately 30 mm diameter of two different materials will be activated: aluminum and copper. Its densities are 2.70 g/cm³ and 8.92 g/cm³, respectively. The masses of the samples are engraved on the discs in the unit of grams [g]. A third sample of unknown material is to activate, too, and to identify on base of the measured half-life (simple kind of **n**eutron **a**ctivation **a**nalysis (NAA)).

Necessary data for activation and analysis of aluminium and copper are summarised in Tab. 1. As obvious from Tab. 1, two isotopes with different half-lives are being activated in the copper sample. This has to be considered while evaluating its decay curve.

Chemical element	Nuclide to be activated	Fraction in element	$\sigma(n_{th},\gamma)$	Activated nuclide	$T_{1/2}$	Emitted radiation of the nuclide
Al	$^{27}_{13}Al$	100 %	0.215 b	$^{28}_{13}Al$	2.3 min	2.87 MeV β^- 1.78 MeV γ
Cu	$^{63}_{29}Cu$	69.1 %	4.3 b	$^{64}_{29}Cu$	12.8 h	0.57 MeV β^- 0.66 MeV $\beta+$ 1.35 MeV γ
	$^{65}_{29}Cu$	30.9 %	2.1 b	$^{66}_{29}Cu$	5.1 min	2.63 MeV β^- 1.5 MeV β^- 1.04 MeV γ

Tab. 1, Activation and decay data for aluminium and copper

3.2. Start of the Reactor and Activation of the Samples

The safety check of the reactor has to be made and the reactor needs to be started. It should reach criticality at a power of 2W.

Irradiation will be carried out in the horizontal experimental channel no. 3, which goes immediately through the graphite reflector tangentially to the reactor core. The cover on the irradiation channel and the inserted radiation shielding plugs have to be removed. At the open channel, radiation dose rates for both, neutrons and γ-rays have to be determined and discussed with the supervisor with respect to radiation protection measures.

Then the first sample can be inserted by using a manipulator. The depth of insertion is 1.25 m. At this depth, the sample is placed immediately next to the core of the reactor.

The half-lives of the activated isotopes of aluminum and copper account for only several minutes, which results in a steep decline of radioactivity after the end of irradiation. Therefore, the samples have to be irradiated one after another and immediately measured after the end of irradiation. The irradiation time for both samples is 10 min, respectively.

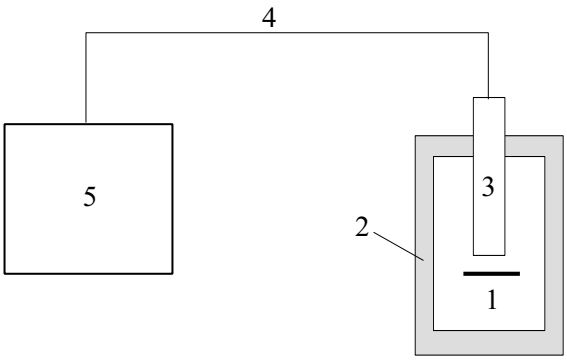
3.3. Evaluation of the Experiment

3.3.1. Description and Configuration of the Measurement Equipment

The γ-radiation of the activated Al and Cu nuclei has to be measured in dependence on decay time. For radiation detection a scintillator is available. Upon incident irradiation, flashes (scintillations) occur in the scintillator due to atomic interactions. These scintillations are converted into photoelectrons in a secondary multiplier phototube (so-called photomultiplier)

by the photoelectric effect.

Between the cathode and the anode of the photomultiplier, several intermediate electrodes (so-called dynodes) are installed. Between each two dynodes, a part of the high-voltage of the photomultiplier is applied. Therefore, the incident electrons are accelerated in such a way that they sputter more electrons in the next dynode (collision ionisation). Thus, the photomultiplier serves to convert scintillations into electric current and voltage signals amplifying them with very low noise. The voltage pulses are further amplified in a subsequent electronic amplifier and counted by a pulse counter. The measurement setup is shown schematically in Fig. 4.



1 Radioactive sample
2 Radiation shielding (lead)
3 Measuring head VA-S-50 with fotomultiplier and scintillator
4 Cable connection
5 Radiation measuring device 20046

Fig. 4, Experimental setup

The following parameters are set at the radiation measurement device:

Threshold (Pegel):	5.7 V
High-Voltage (Hochspannung):	-1140 V (5.70 at the potentiometer)
Amplification (Verstärkung):	22 dB
Measuring time (Messzeit):	6 s = $1 \cdot 10^{-1}$ min
Channel width (Kanalbreite):	DIS

The main purpose of the lead shielding around sample and detector is not to shield radiation from inside the apparatus, but rather to absorb background noise from the exterior surroundings due to reactor operation. Without shielding, this background would result in a high number of noise pulses.

3.3.2. Measurement of the Radioactive Decay of Activated Samples

Before measuring the activated samples, the background radiation has to be measured 5 times during 6 s each, which of the average is calculated. Background radiation impinging on the shielded scintillator results in a continuous noise in the measurement signal that needs to be subtracted for evaluation of the underlying radiation of the activated samples.

The samples irradiated in the reactor are removed from the reactor channel after exactly 10 min of irradiation time and immediately put on the sample holder beneath the scintillator for radiation measurement. The time of the end of sample irradiation has to be exactly recorded and will be taken as the point of origin of the time of decay.

The number of pulses has to be measured for exactly 6s (preset at measuring device), respectively, after time increments of 0.5 minutes for 10 to 15 min after the end of irradiation. The measured time dependence of the count rate has to be illustrated in a graph, after background correction, both, on linear and logarithmic ordinate scale.

3.3.3. Determination of Half-lives

The half-lives of the copper isotope ^{66}Cu and for ^{28}Al have to be determined based on the obtained decay curve. Subsequently, the decay curve and the half-life determined for the unknown activated material under investigation will help to identify the material and its composition. This will show the possibility of identifying unknown materials by neutron activation analysis (NAA).

For every half-life calculation the error bars have to be estimated. Every measured count rate is subject of a statistical error. Statistical analysis based on the POISSON probability distribution show that the count rate of Z can be measured with a statistical error of $\Delta Z = \pm\sqrt{Z}$. The thereby determined error bars should be included in the graph of measured values to illustrate the range of possible decay curves and of the half-lives that can be derived based on the measurements.

3.3.4. Calculation of the Neutron Flux Density at the Point of Irradiation

The relation between neutron flux density Φ at the point of irradiation in the reactor and the sample activity per volume at the end of irradiation is given through equation (9) if $t = t_{irr}$. For evaluation, the decay curve needs to be extrapolated to the point of origin $t_{decay} = 0$, equivalent to the activity at t_{irr}. The volumic activity obtained by equation (9) has to be still multiplied by the sample volume to obtain the total activity.

However, this value is not yet equivalent to the activity of the sample, but only proportionally related. The proportionality factor includes a geometry factor and the counting efficiency of the scintillator: Part of the emitted radiation from the sample is not counted in the detector because it is directed to the bottom or the sides without penetrating the scintillator (geometry factor), and not all of the emitted species that impinge on the scintillator cause a pulse (energy-dependent counting efficiency). Previously, this proportionality factor has been experimentally determined as $C \approx 0.01$.

Considering this information, equation (9) can be rewritten as

$$Z(t_{irr}) - n_0 = C \cdot V \cdot A(t_{irr}) = C \cdot V \cdot \Sigma \cdot \Phi \cdot [1 - \exp(-\lambda \cdot t_{irr})] \quad \textbf{(13)}$$

$Z(t_{irr})$ measured count rate extrapolated for $t_{decay} = 0$
n_0 background noise that has to be substracted from the measured count rate

Additionally, it has to be considered that in natural copper, the ^{65}Cu nuclei account for only 30.9 % of the total. For calculation of the neutron flux density, equation (13) can thus be rewritten as:

$$\Phi = \frac{(Z(t_b) - n_0) \cdot AG}{C \cdot m \cdot 0.309 \cdot N_L \cdot \sigma \cdot [1 - \exp(-\lambda \cdot t_b)]} \quad \textbf{(14)}$$

<u>Remark</u>: Of cause, also the activated isotope ^{64}Cu contributes to the measured count rate as well. However, as calculated in question to ask no. 9, the contribution accounts for less than 10 % of the total count rate and shall thus be neglected in the calculations performed in this task.

4. Radiation Protection

Handling of radionuclides requires particular protection and safety measures in order to exclude any danger to the health of the involved personnel and to prevent uncontrolled distribution of radioactive materials. To ensure strict compliance with radiation protection measures, every involved person has to get accustomed to the particular working technologies required when handling radioactive sources. This involves an extensive learning process to which any person needs to get accustomed to, who is frequently exposed to radionuclides.

For this experiment, the following rules need to be respected in particular:

1. Perform a wipe test in order to confirm that the irradiation channel is free of contamination.

2. Never stand right in front of the open irradiation channel. Approach the channel and insert the sample from the side.

3. When walking around the reactor, avoid passing the open irradiation channel.

4. Before performing the experiment, thoroughly plan the working steps to minimise the time of handling the activated samples.

5. Do not touch the activated sample bare-handed, but use a pincer or forceps. Wear gloves while handling the sample.

5. Questions to Answer

1. What does radioactivity mean ?

2. What do the terms element, isotope, nuclide mean ?

3. What is the difference between natural and artificial radioactivity ?

4. What information is available from a decay scheme ?

5. Which properties are decisive to select a radionuclide for a specific purpose ?
   Discuss by using the example of a liquid level measurement of a storage container or the intake of a radioactive medicament as a tracer.

6. Determine the percentage of the saturation activity reached by a sample irradiated 2, 4, or 7 times its half-life. Derive for how long a nuclide should be irradiated for its activation.

7. Determine the probability for a thermal neutron to reach the center of a 1 mm thick copper or gold sample if it impinges perpendicular the probe surface.
   (σ (n_{th},γ) for Au-197 = 98.7 b, ρ = 19.3 g/cm³)

8. Which peculiarity of natural copper has to be considered when drawing the decay curve (elemental composition) ?

9. Calculate the activity of both irradiated samples (copper and aluminum) after 10 minutes of irradiation with an estimated neutron flux density of $\Phi_{th} = 10^7$ n/(cm² · s). Why is the count rate measured with the measurement equipment lower than the calculated value ?

10. Describe qualitatively the radiation detection process in a scintillator and a photomultiplier.

Protocol for Experiment
"Activation and Decay of Radioactive Nuclides"

Date:

1. Measurement of background radiation (5 measurements for 6 s each)

Number of measurement	Measured value

Average value (=N_0):

2. Determination of half-life

Decay time /min	Al: N_i	Al: N_i - N_0	Cu: N_i	Cu: N_i - N_0	X: N_i	X: N_i - N_0
0						
0.5						
1.0						
1.5						
2.0						
2.5						
3.0						
3.5						
4.0						
4.5						
5.0						
5.5						
6.0						

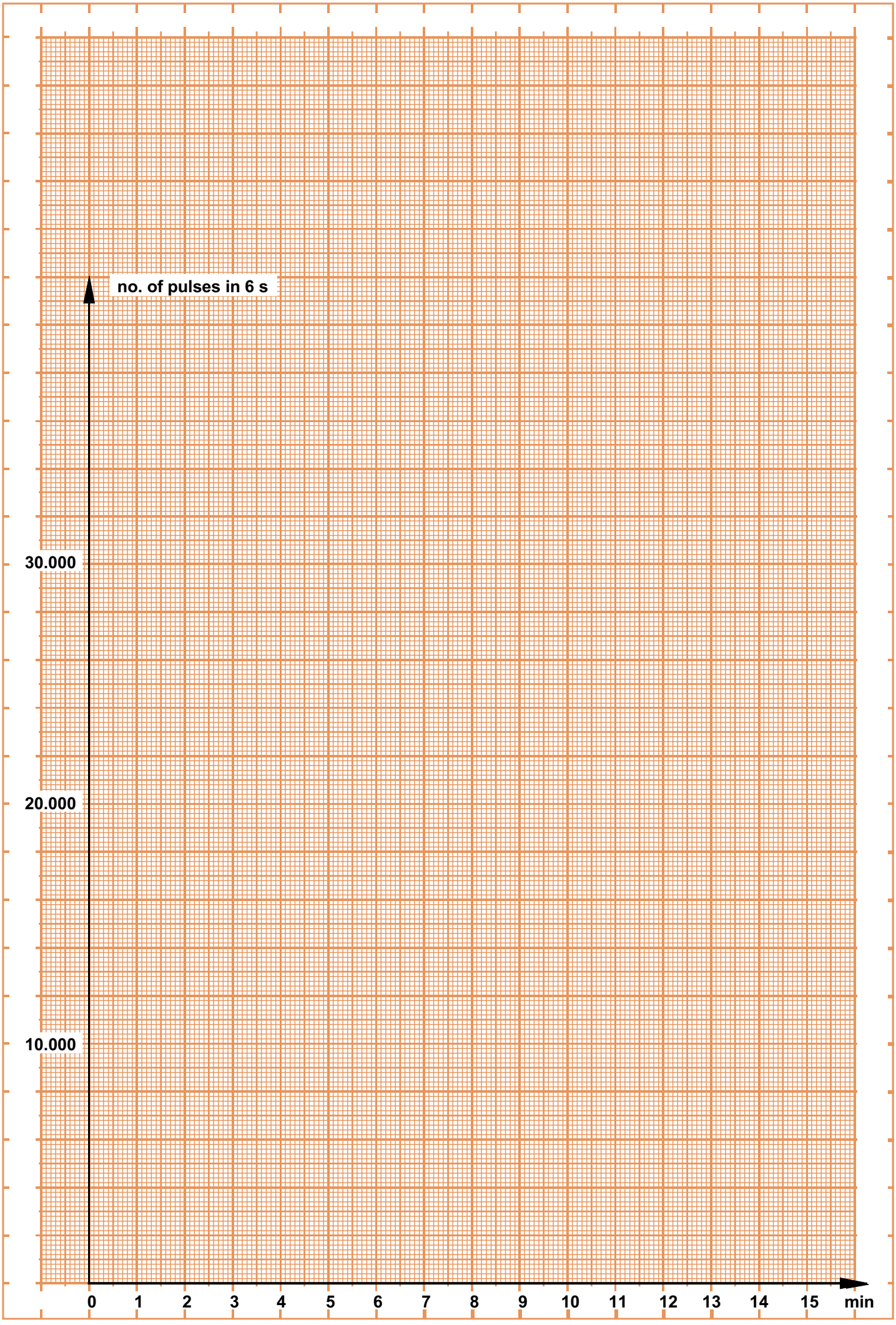
Decay time /min	Al: N_i	Al: $N_i - N_0$	Cu: N_i	Cu: $N_i - N_0$	X: N_i	X: $N_i - N_0$
6.5						
7.0						
7.5						
8.0						
8.5						
9.0						
9.5						
10.0						
10.5						
11.0						
11.5						
12.0						
12.5						
13.0						
13.5						
14.0						
14.5						

Results:

Half-life (Al):

Half-life (Cu):

Half-life (X):


no. of pulses in 6 s
30.000
20.000
10.000
0
1
2
3
4
5
6
7
8
9
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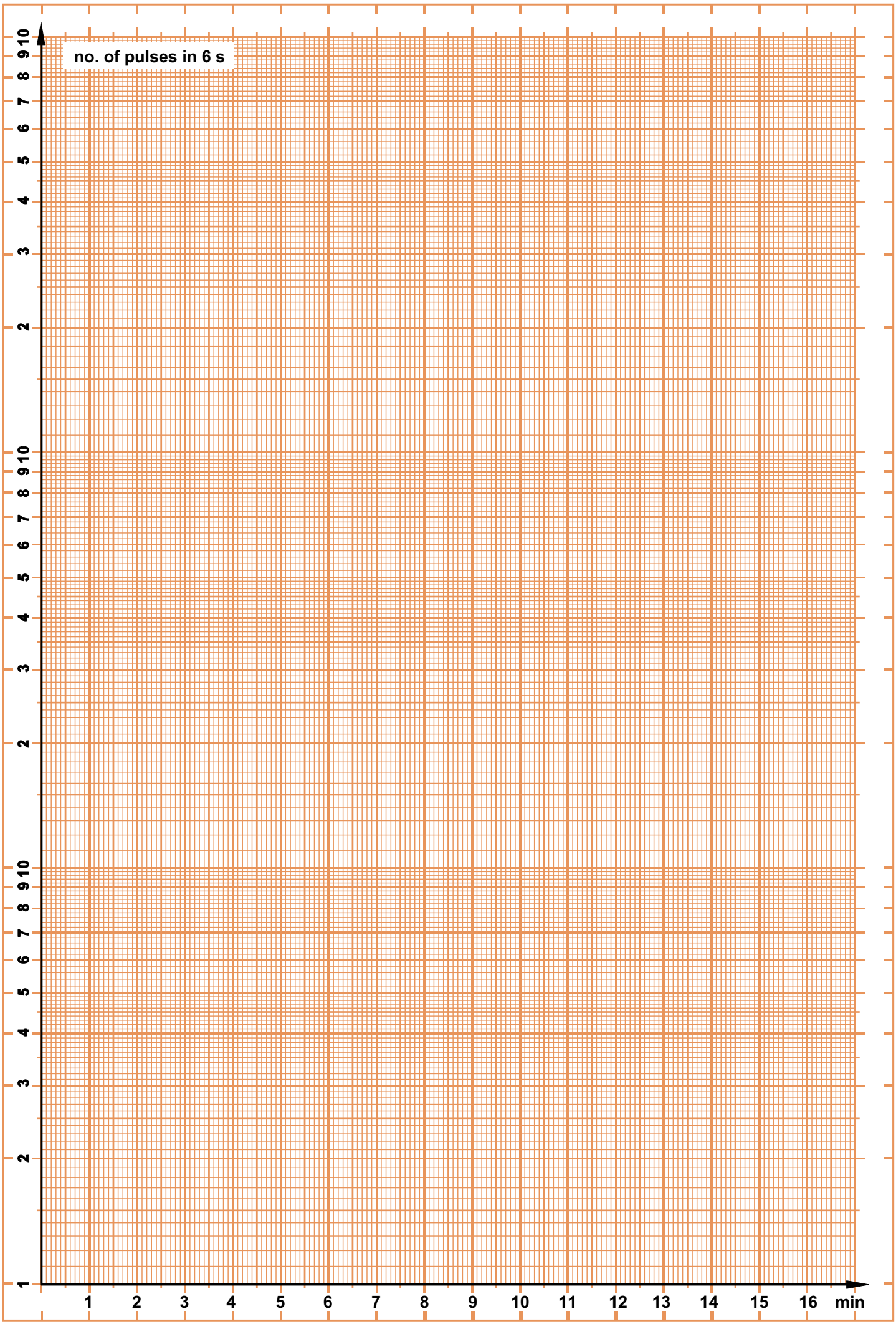
11
12
13
14
15
min


no. of pulses in 6 s
1
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2
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4
5
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7
8
9
10
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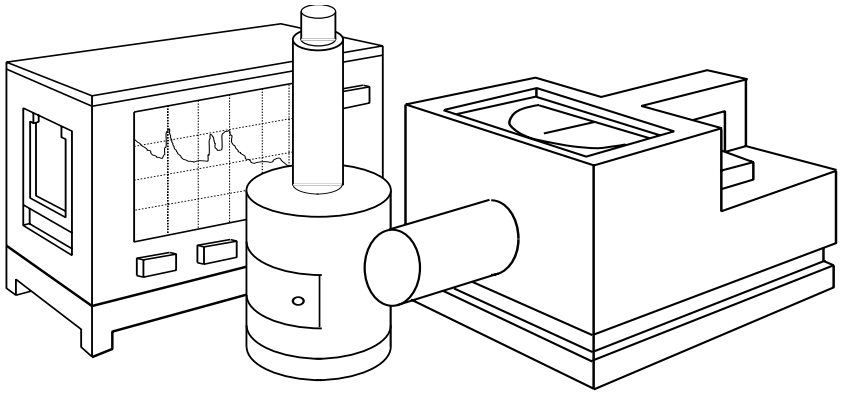
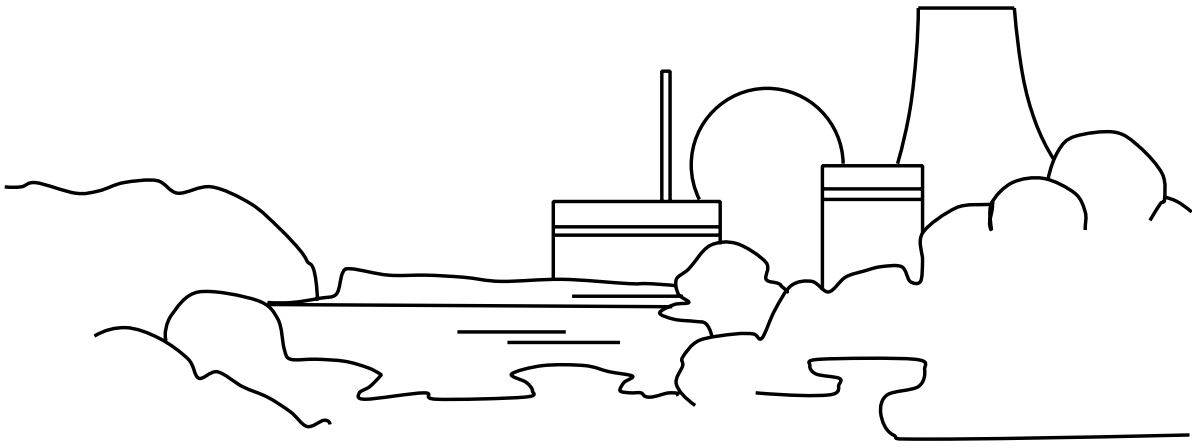
TECHNICAL UNIVERSITY DRESDEN
Institute of Power Engineering
Training Reactor



Reactor Training Course

Experiment

"Identification of Unknown Radionuclides" (γ - Spectrometry)



Instruction for
Experiment "Identification of Unknown Radionuclides (γ-Spectrometry)"

Content:

Figures:

(issued: March 2015)

MOTIVATION:

Gamma spectrometry is one of the fundamental measuring techniques in nuclear technology for unambiguous, clear identification of radionuclides, even in mixtures of various isotopes.
The experiment is intended to show the basics of the methodology of gamma spectrometry, the properties of most important gamma detectors and how to identify clearly radionuclides. Moreover, the absolute activity of a radionuclide could be determined by means of a gamma spectrometer if it is calibrated absolutely with regards to its efficiency.

1. Theoretical Background

1.1. Introduction

In nuclear technology, an important task is to identify unknown radioactive nuclides being alone or in a mixture in a sample and to quantify them with regard to their activity or percentage in the mixture. Nowadays, about 270 stable and more than 2000 unstable (radioactive) nuclides are known.

Radioactivity is the transformation of unstable nuclei into stable ones accompanied by a release of energy. Due to this process, the configuration of the atomic nucleus changes. The various transformation processes are accompanied by the emission of ionising radiation.

With respect to radioactivity, a radionuclide is characterised by three properties, which can be used for identification:
- the kind of emitted radiation,
- the energy of emitted radiation, and
- the half-life.

The **kinds of emitted radiation** are:
- alpha radiation (i.e. nuclei of He-4),
- positive beta radiation (i.e. positrons),
- negative beta radiation (i.e. electrons) and/or
- gamma radiation.

Because of the large number of different radionuclides, the unambiguous identification of a nuclide by only four possible kinds of radiation is very limited. In many radioactive decays, not only one kind of radiation is emitted but sometimes several of them in combination. Especially gamma radiation accompanies alpha and beta radiation in almost all cases.

The measurement of the **half-life** is another possibility to identify radionuclides (see also experiment "Activation and Decay of Radioactive Nuclides").

Although the values of half-lives range between parts of a second and 10^9 years and beyond (i.e. a wide time scale), some nuclides have very similar half-lives and cannot be distinguished within the error tolerances of the experiment. Even more complicated is the case of mixtures of radionuclides with overlaying half-lives. A separation of the particular components would be possible only in a few cases (if the half-lives are very different).

The measurement of the **energy of the emitted radiation**, however, allows an unambiguous identification. In the radioactive decay, the unstable nucleus with its excess of energy gets to the lower energy state of the daughter nucleus by the emission of particles or gamma quants or by resting at an excited immediate state of the daughter nucleus that is left for the ground state only after emission of further radiation. All possible energy levels in an atomic nucleus are predetermined, i.e. through a transformation always the same certain amount of energy is released in form of radiation. As an example Fig. 1 shows the decay scheme of Co-60. The horizontal lines illustrate the energy states of the nucleus. The arrow lines in between mark the kind of decay and its energy. Horizontally shifted lines refer to changes in the charge of the nucleus, i.e. the emission of charged particles. Non-shifted lines mark the emission of gamma quants (no charge). Since energy levels of all existing radionuclides have different values and, consequently, all decay schemes have individual properties, it is possible to identify any nuclide by the measurement of the energy of the emitted radiation. Since almost any radioactive decay is accompanied by gamma emission, spectrometry of this kind of radiation is possible for almost all nuclides. Therefore, gamma spectrometry is the most important and universal method for the identification of radionuclides.

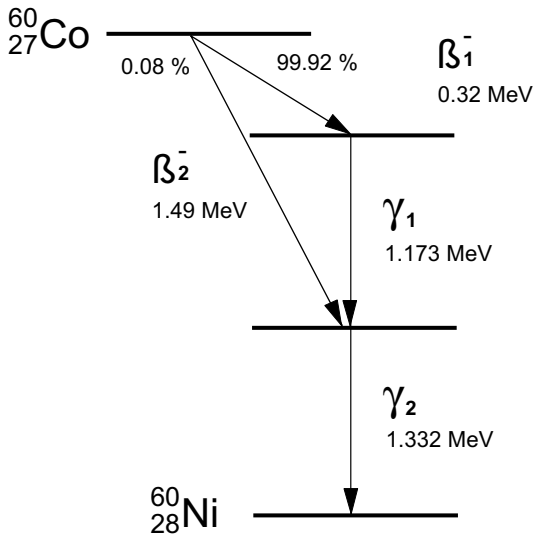


Fig. 1, Radioactive decay of Co-60

1.2. Interaction of Gamma Radiation with Matter

Gamma radiation is an electromagnetic radiation consisting of gamma quants (= photons). Gamma quants do not carry an electric charge. Therefore, the interaction of gamma radiation with matter differs from that of charged particles with matter. The energy absorption is possible by the **photo effect**, the **Compton effect**, and **pair production**. All these effects cause the production of secondary electrons that are detected (as charged particles) in appropriate measurement devices.

The gamma spectrum is a discrete spectrum. Though, gamma detectors do not only detect distinct sharp lines. This is due to the different possible interactions mentioned above of the gamma radiation with the detector material. Fig. 2 shows a typical measured pulse-height spectrum for a discrete gamma energy (that of the radionuclide Cs-137). At the full photon energy E_γ, the **photopeak** appears. The gamma quant is completely absorbed by a shell electron of the detector material. In this case, the photon vanishes and a free electron that carries the same energy as the former photon (subtracted by the small ionisation energy of the electron) is released.

If the photon does not lose its entire energy, only an elastic collision with a shell electron of the detector material occurs (**Compton effect**). In this case, the photon transfers only fractions of

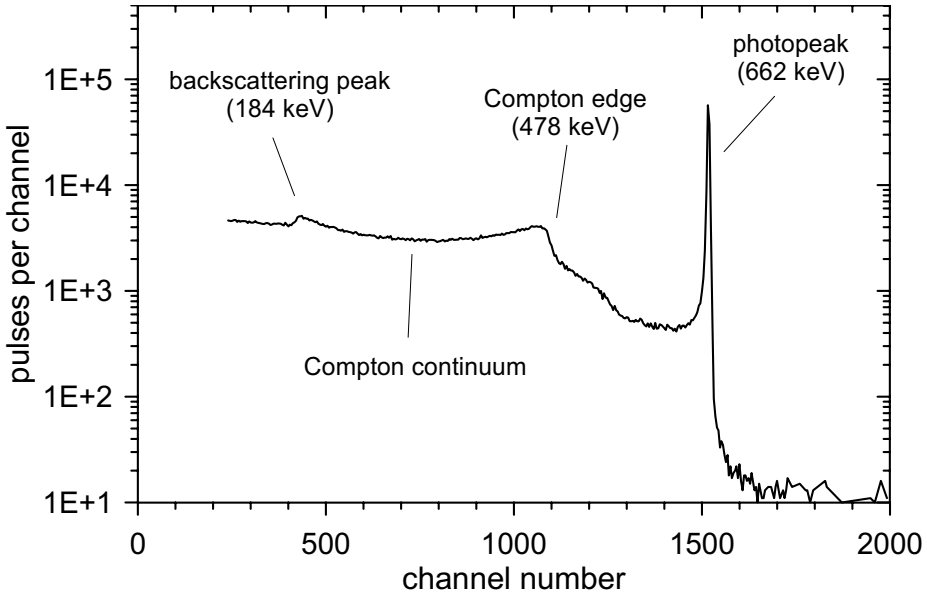


Fig. 2, Measured pulse height spectrum of the radionuclide Cs-137

its energy to the shell electron, is scattered carrying still a reduced energy and leaves the detector or can even have a second interaction. The measured pulse height corresponds to the energy of the produced free Compton electron. According to conservation of momentum and energy the maximum energy that can be transferred to an electron is (at an angle of 180°):

$$E_c = \frac{E_\gamma}{\left(1 + \frac{m_0 \cdot c^2}{2 \cdot E_\gamma}\right)} \qquad \textbf{(1)}$$

(with $m_0 \cdot c^2 = 511$ keV being the energy equivalent of the electron mass (electron rest energy)). This maximum energy of the Compton effect corresponds to the **Compton edge** in the pulse height spectrum. At lower energies, the spectrum continues with a plateau, the so-called **Compton continuum** that results from energy transmission at angles less than 180°.

At photon energies of some MeV, the full-energy peak is caused by **pair production**. If the energy of the photon exceeds the amount of twice the rest energy of the electron (E > 1.02 MeV), then the Coulomb field of the nucleus can transform a photon into an electron-positron pair. The exceeding energy is almost equally distributed to the electron and the positron as kinetic energy. The pair production is always followed by annihilation of the positron, i.e. after its full slow-down, the positron unifies with an electron of the surroundings under emission of gamma radiation (annihilation radiation).

The ratio of occurrence of the different described effects is determined by the energy of the photon and the material of the detector (Fig. 3). The photo effect is dominating in case of low-energy photon radiation in combination with a high atomic number of the detector material. The Compton effect occurs mainly at intermediate energies at a range of about (0.4 ... 4) MeV.

Photons may also pass the detector without any interaction and re-enter the detector with much reduced energy after being backscattered in the surrounding material. The absorption of backscattered photons corresponds to a **backscattering maximum** at

$$E_{BS} = \frac{E_\gamma}{\left(1 + \frac{2 \cdot E_\gamma}{m_0 \cdot c^2}\right)} \qquad \textbf{(2)}$$

that overlaps with the Compton continuum.

In case of a gamma decay with emission of multiple gamma energies or in case of a mixture of various nuclides, the corresponding spectra will overlap. Hence, the identification may be more difficult.

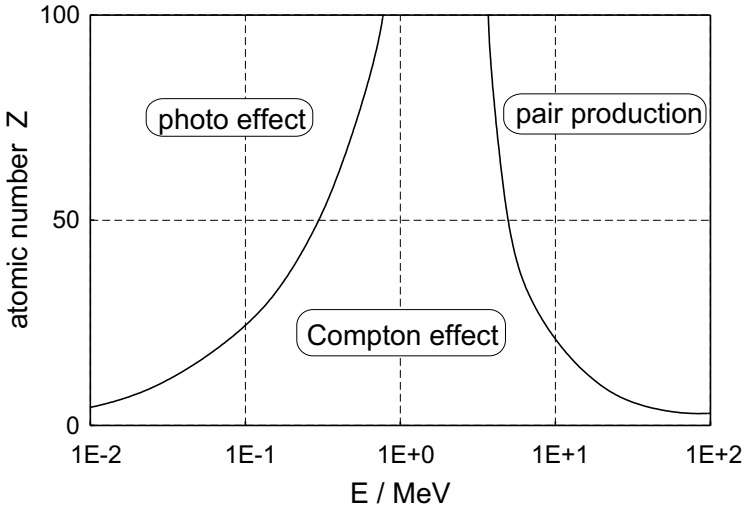


Fig. 3, Atomic numbers and energy ranges with dominant occurance of photoeffect, Compton effect and pair production

1.3. Measurement of the Radiation Energy

The measurement setup is presented in Fig. 4. The detector is coupled via a preamplifier and a spectroscopic amplifier to a multichannel analyser. In the charge-sensitive preamplifier, the primary charge pulses are converted into voltage pulses (output-voltage is proportional to the input charge). In the spectroscopic amplifier, the pulses are amplified up to a range of 1 ... 10 V. The multichannel analyser recognises the incoming pulses, evaluates them with respect to their height (voltage), which contains the information about the energy, and sorts them into consecutive pulse-height channels. The result is a pulse-height spectrum that contains the energy information of all recognised gamma quants. Each measured gamma quant is a count in the corresponding energy channel of the multichannel analyser. It should be realised that even discrete gamma energies do not cause sharp lines in the spectrum, because of statistical fluctuations of the elementary processes in the detector and in the post-processing electronics. This phenomenon creates from theoretically sharp lines GAUSSIAN bell-curves (peaks). The mean

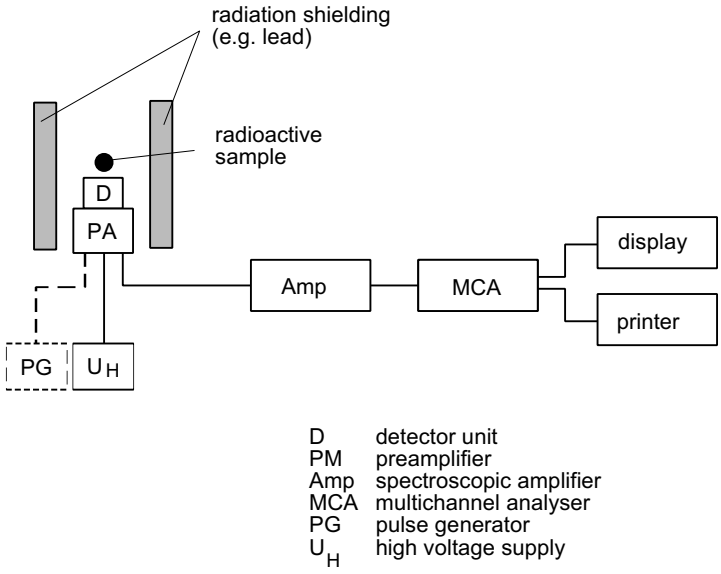


Fig. 4, General design of a gamma spectrometer

pulse heights correspond to the energy of the gamma quants. Using a PC programme, the gamma energy spectrum can either be observed on the screen, directly evaluated or saved. The required high voltage is fed to the γ-detector via the preamplifier.

Because temperature effects and voltage fluctuations can shift the electronic amplification of the measurement setup and correspondingly also the channel number of the photopeaks during the measurement, high-quality spectrometers have an automatic amplification control capability. This control uses voltage pulses of an exactly constant pulse height provided by a precision-pulse generator and fed as additional input signal to the preamplifier during the entire measurement. The electronics of the multichannel analyser automatically controls the internal amplification such that the generator peak steadily remains at a constant channel number. The generator pulses are set in such a way that the corresponding peak rises in a region of the pulse-height spectrum where photo lines of the sample are not expected. Usually this is at the very upper end of the pulse-height spectrum.

In most cases, the γ-detectors are surrounded by a radiation shielding, e.g. made from lead, to minimise the influence of the ambient radiation from the environment especially for low gamma activity measurements.

Gamma spectrometry is usually carried out with scintillation detectors (e.g. NaI(Tl)) or semiconductor detectors made from Ge(Li) or more recently made from high purity germanium (HPGe).

Scintillation detectors are relatively easy to use and have a high detection efficiency. Special kinds of such scintillators with a tight deep hole in the crystal (where the sample is placed within the detector at the bottom) enables a measurement with almost 4π-geometry.

A disadvantage of scintillators is the limited energy resolution, i.e. the ability to resolve two gamma peaks that are located close to each other as two separate lines.

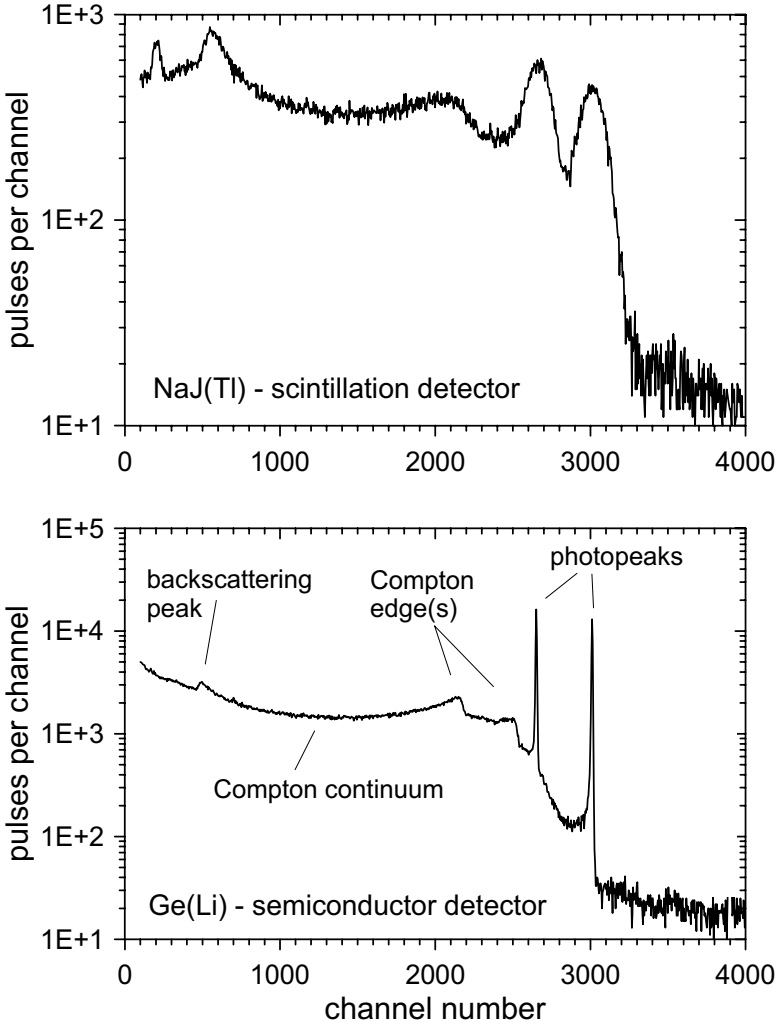


Fig. 5, Comparison of gamma pulse height spectra for Co-60, measured by means of a NaJ(Tl) scintillation detector (upper diagram) and a Ge(Li) semiconductor detector (lower diagram)

Semiconductor detectors have an excellent energy resolution. Therefore, they are preferably used for demanding analyses in laboratories. Though, the detection efficiency is in general considerably lower compared to scintillation detectors. Using higher efficiencies corresponds to rapidly rising detector costs, even though the efficiency of scintillation detectors cannot be reached. Furthermore, semiconductor detectors have to be cooled (permanently in case of Ge(Li) and at least during usage in case of HPGe) to the temperature of liquid nitrogen (77K). This means further physical and financial efforts.

Fig. 5 shows the pulse-height spectra of the radionuclide Co-60 (gamma energies at 1173 keV and 1332 keV measured with a NaI(Tl)-scintillation detector and a Ge(Li)-semiconductor detector, respectively. The quality of the semiconductor detector proves in form of a clearly better energy resolution of the photo peaks.

The resolution (of a detector) is defined as the width of a peak at half of its maximum height (full width at half maximum, FWHM). It is given either as absolute resolution in terms of channel numbers or energy units or as relative resolution if divided by the channel number or energy position, respectively, of the peak maximum.

1.3.1. Scintillation Detectors

Scintillation detectors consist of a combination of a luminescent material (scintillator) that is stimulated by ionising radiation to emit flashes (scintillations) and a photomultiplier that converts the flashes into electrical pulses (Fig. 6).

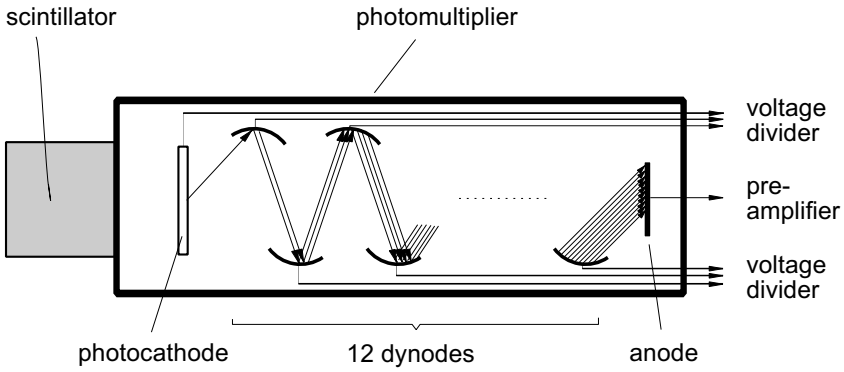


Fig. 6, General design of a scintillator-photomultiplier unit

Within the scintillator, the absorbed energy of a gamma quant is converted into light by excitation of the scintillator material and its subsequent return to the ground state. For spectrometric purposes, the absorbed gamma energy and the number of emitted light quants and the subsequent pulse height at the output of the photomultiplier must be proportional to each other.

Inorganic mono-crystals made from sodium iodide (NaI) doped with Thallium for activating the light emission has proved a suitable scintillator material in spectroscopy of gamma radiation (NaI(Tl)-scintillators). Because of their high physical density $\rho = 3.67$ g/cm³, their high content of iodine of 85 wt%, and due to the ordinal number of $Z = 53$, they have a high absorption capability for gamma radiation. Clear crystals that are homogeneously doped and transmissible for their own fluorescent light can be produced in dimensions up to a thickness of e.g. 300 mm or diameter of 400 mm. Also high-energy gamma quants are sufficiently absorbed by these crystals. Commercially available crystals are hermetically sealed (because NaI is hygroscopic) e.g. within aluminium cases, having a glass or plastic window for the light emission. In order to avoid light losses all surfaces of the crystal except for the emission window are not polished but surrounded with a reflecting material, e.g. MgO.

The energy resolution of NaI(Tl)-scintillators is usually given with regard to the gamma line of the radionuclide Cs-137 ($E_\gamma = 662$ keV). Best energy resolution obtained with selected NaI-scintillation crystals and photomultiplier combinations are in the order of about 6 %, common values for commercial scintillator crystals and photomultipliers range between (8...12) % FWHM.

The scintillations from the crystal need to be transmitted to the photocathode of the photomultiplier with only as small as possible losses. Immersion layers (e.g. silicone oil) between scintillator surface and photomultiplier reduce total reflection.

By the photoelectric effect, a few free electrons are emitted on the photocathode. Between the photocathode and the anode of the photomultiplier, a graduated high voltage is applied via a voltage divider and a series of intermediate electrodes (dynodes). In the electric field between each two of the dynodes, the electrons accelerate and liberate further electrons by collision ionisation at the next dynode. This effect leads to a low-noise amplification of the initial number of electrons (factor $10^5...10^9$). Thus, scintillation flashes are transformed into energy-proportional electric current and voltage signals and are amplified in the photomultiplier.

Since ambient light would completely overlap the scintillation effect of the detector and also stimulate the photocathode of the photomultiplier to emit electrons, the entire scintillator-photomultiplier system needs to be light-tightly sealed.

1.3.2. Semiconductor Detectors

In the past, mainly scintillation detectors have been used for the determination of gamma spectra. Nowadays, almost exclusively semiconductor detectors are applied for laboratory use. Their main advantage is their high energetic resolution, which is about 10 to 20 times better than that of a scintillator. The energy resolution of a semiconductor detector exceeds that of all other radiation detectors because the energy needed to produce a pair of charge carriers is very low (only about 2.96eV) with the consequence that for a certain amount of absorbed energy a very high number of charged particles with corresponding excellent statistic accuracy is produced. There are the older Ge(Li)-semiconductor detectors and the increasingly used high-purity germanium detectors (HPGe).

Manufacturing of a Ge(Li)-semiconductor detector:
In a p-conducting germanium crystal (doped with boron), at 400 °C, lithium atoms diffuse and build up a high Li-concentration at the surface of the crystal. The Li-atoms act as donors, accumulate at inter-lattice positions and form a p-n-junction. Already at room temperature, the inter-lattice atoms are ionised. Consequently, the crystal contains ionised positive lithium atoms. When applying inverse voltage, these ions drift from the n-region to the p-region of the crystal as long as their concentration is higher than that of the acceptors. Due to the formation of boron and lithium ion pairs in the crystal, a high-resistance intrinsic layer (i-layer) is achieved. In this layer, the impurities are eliminated because the negative ions got compensated. Such an arrangement is called pin-structure. The i-layer is an effective barrier with a voltage-independent thickness.

Mechanism of Ge(Li)-semiconductor detectors:
High voltage is applied to the contacts (n and p) in inverse direction. As long as no ionising particle penetrates the crystal no electric current arises, since the i-layer has a large electric resistance. If a gamma quant enters the crystal, it generates electron-defect-electron pairs along its trajectory due to photo effect, Compton effect and pair production. These charges are collected at the respective poles by the applied voltage. At the poles, a temporary gain of electric charge, i.e. a potential difference, comes up and causes an electric pulse in the detector circuit (electrical discharge). In a charge-sensitive preamplifier, the charge pulse is converted into a voltage pulse (with an output signal proportional to the input charge).

Cooling:
One disadvantage of semiconductor detectors is that at room temperature, even lattice vibrations cause ionisations in the crystal. Therefore, semiconductor detectors need to be cooled during measurements. The cooling is usually made with liquid nitrogen (boiling temperature 77 K or -196 °C, respectively) or recently also by electric cooling units based on the Peltier effect. Ge(Li)-semiconductor detectors need to be cooled even when they are only stored and not in use because at room temperature, lithium atoms are very mobile and would diffuse out of the crystal. Thus, the detector would be distroyed.

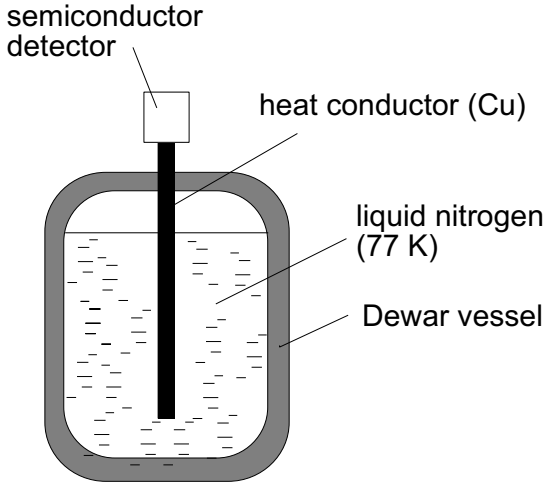


Fig. 7, Cooling arrangement of a semiconductor detector

The liquid nitrogen is kept by a thermally well-isolated Dewar vessel (principle of a thermo, see Fig. 7). The semiconductor detector is connected to the nitrogen by a copper finger that removes heat from the detector via heat conduction. To reduce heat losses at the detector, the detector is additionally put into a vacuum vessel. Because heat losses are not entirely avoidable the nitrogen gradually evaporates and needs to be refilled regularly.

1.4. Screen Display of the Multichannel Analyser

Fig. 8 shows a screen display of the multichannel analyser computer program for an arbitrary spectrum. The vertical cursor in the spectrum marks the selection of a certain channel of the measured spectrum (middle) and the cursor region (left and right). The number of the selected channel is displayed below the spectrum (marker). If the channel axis of the spectrometer has been calibrated with respect to energy, also the corresponding energy of the channel is displayed. Next to this information, the number of counts in the selected channel is displayed.

Important displays within the column on the right hand side of the screen are:

Display	MCR/Buffer	Display of the spectrum from the current measurement or from the buffer of the multichannel analyser
	Full/Expand	Display of the whole spectrum or only of that part selected in the cursor region
	Vt	Scaling of the ordinate axis (vertical)
	Hz	Number of displayed channels on the abscissa axis (horizontal)
Presets	Rl Tm	Option for preselecting a RealTime for the measurement
	Lv Tm	Option for preselecting a LiveTime for the measurement
Time	Rl Tm	Elapsed RealTime of the measurement
	Lv Tm	Elapsed LiveTime of the measurement
	Dead Tm	DeadTime of the multichannel analyser (in %)

The terms RealTime, LifeTime and DeadTime need to be explained. RealTime is the time period of the measurement according to the clock time. During processing a detector pulse, the multichannel analyser cannot accept additional pulses (it is "busy"). The gate of the analyser is said to be closed. Only after processing the pulse, the gate will be re-opened again to enable the multichannel analyser to accept a next pulse. The time period of an actually open gate is called

LifeTime. Consequently, the DeadTime is that time period with the gate closed. The LifeTime is essential for any absolute measurement of spectra or activity determination. Preferably, the LifeTime should be close to the RealTime, i.e. the DeadTime should be kept small. This can be realised by either fast electronics in the multichannel analyser or by adjusting small count rates.

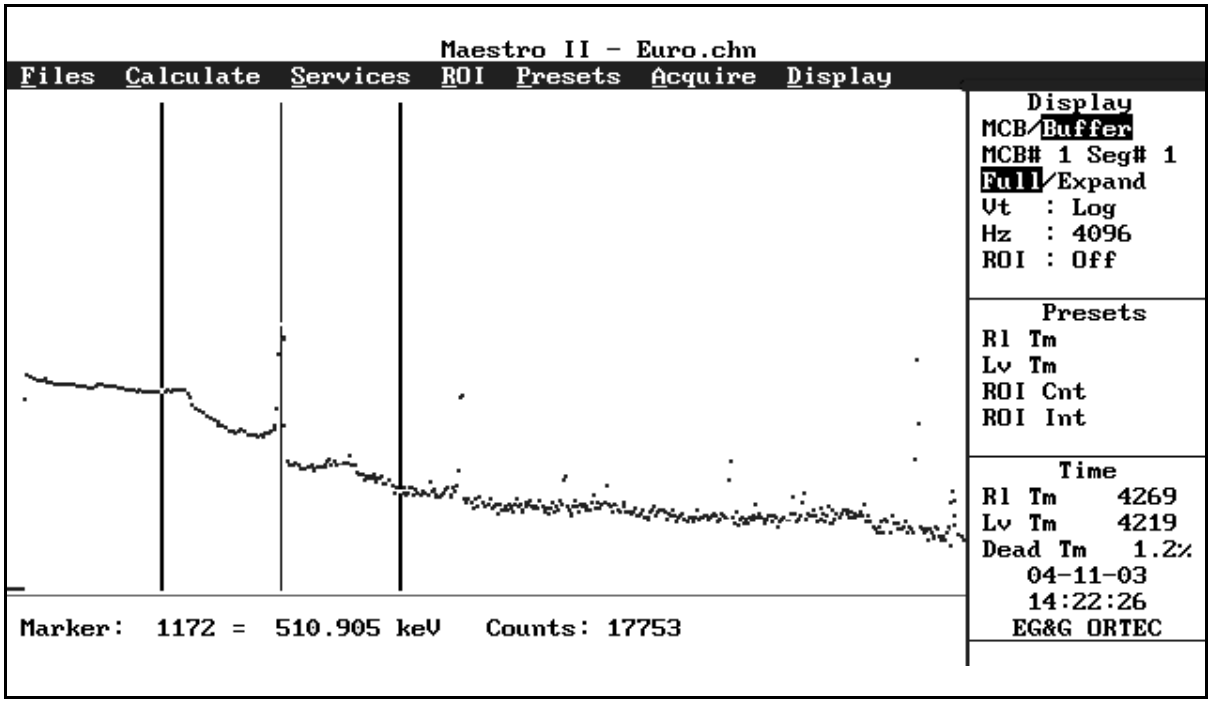


Fig. 8, Monitor screen of a multichannel analyser

2. Tasks

2.1. Comparison of the Properties of a NaI(Tl)-Scintillation Detector and a Ge(Li)-Semiconductor Detector

The gamma spectra of a Co-60 source, which is characterised by emission of two gamma energies, are measured by both the detectors consecutively.

The setup of the scintillation spectrometer is given in Fig. 9 and that of the semiconductor spectrometer is given in Fig. 10. The parameters of the electronic adjustment are provided by the supervisor. A description of the most important buttons for operating the spectrometer is given in Sec. 3.1.

Task: *Place the gamma source Co-60 right onto the respective detector. Discuss the shape of the measured spectra of both the detectors and determine the resolution of the photopeaks. Compare the results of the measurement with the given literature.*

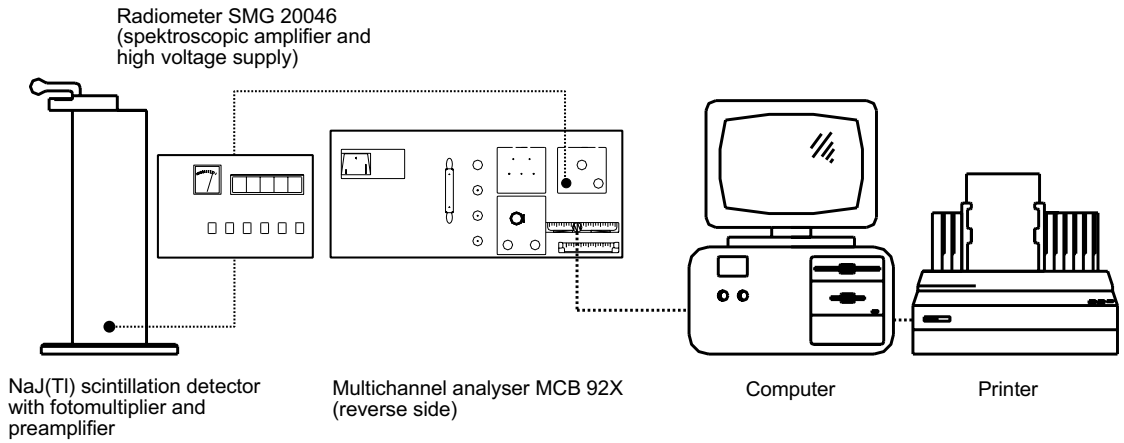


Fig. 9, Measuring setup with scintillation detector

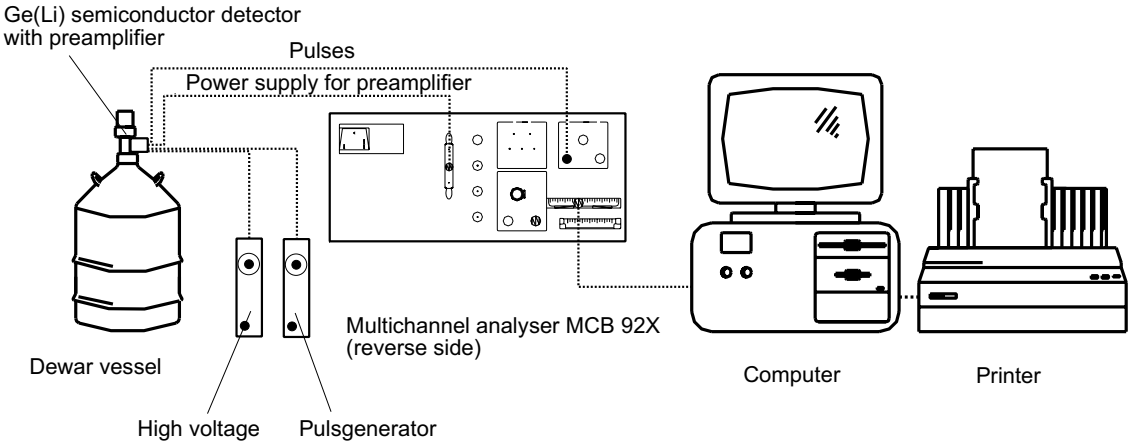


Fig. 10, Measuring setup with semiconductor detector

2.2. Energy Calibration of the Spectrometer with Ge(Li)-Detector

Analysing spectra of pulse heights, only the channel numbers of peaks and the respective number of pulses per channel are available. In order to link the channel number of the multi-channel analyser to the corresponding gamma energy an energy calibration of the spectrometer is necessary.

For the calibration, pulse height spectra of several well known gamma sources (calibration sources) are measured with the multichannel analyser and the channel numbers of the measured photopeaks are related to the well known photopeak gamma energies. A set of such calibration sources is available at the AKR-2 consisting e.g. of nuclides given in Tab. 1.

Nuclide	Energy [keV]
Co-60	1173 1332
Cs-137	662
Kr-85	514
Ho-166m (main lines)	184 280 411 712 810

Tab. 1, Energies of the gamma radiation of some calibration sources usable in the experiment

Task: *For the given gamma sources, determine the relation between channel numbers and energies of the photopeaks and draw a graphical representation of the function E = f(K). What functional relation can be found?*

It is also possible to let the computer calculate the relation between the channel number and the energy, and subsequently also to display the energy of each channel directly on the screen. Therefore, two points (with the channel number and the respective energy each) taken from the determined function E = f(K) have to be entered in the computer program, which calculates a linear relation. For achieving the best accuracy, the two most-separated point (the lowest and the uppermost) should be used.

The required **keyboard inputs** are as follows:
- Place the cursor onto the channel of the first chosen point (photopeak of the calibration source with the lowest gamma energy)
- Switch to the pull-down menu "Calculate": Alt-C
- Activate "Calibrate": Alt-C
- Type the energy of the chosen channel: ... <CR>
- Repeat the procedure with the second pair
- Type the unit for the energy (keV): ... <CR>

After these steps, the abscissa is calibrated with respect to the gamma energy as can be seen at the screen displaying the respective energy value for each channel number (compare Fig. 8).

Task: *Calibrate the spectrometer as explained. Check the calibration with the help of the photopeaks of other calibration sources.*

2.3. Analysis of the Time Dependence of the Activities in a Mixture of Isotopes

The element copper comprises the two naturally occurring isotopes Cu-63 and Cu-65 (Tab. 2). When activating copper (e.g. a copper sheet) in a neutron field, the two radioactive isotopes Cu-64 and Cu-66 are being generated. These two isotopes decay with considerably differing half-lives and emit gamma radiation with the energies given in Tab. 2.

Because of the superposition of the two exponential decay curves in the measurement, a simple integral pulse counting does not allow a clear determination of the half-life (see also reactor training course experiment "Activation and Decay of Radioactive Isotopes"). Using gamma spectrometry the measured photopeaks can be related to the respective isotope and their time-dependent intensity can be analysed isotope-specifically.

Element	Nuclide to be activated	Abundance in element	$\sigma(n_{th},\gamma)$	Activated nuclide	$T_{1/2}$	$E\gamma$
Cu	Cu-63 Cu-65	69.1 % 30.9 %	4.3 b 2.1 b	Cu-64 Cu-66	12.8 h 5.1 min	511 keV and 1348 keV 1039 keV

Tab. 2, Activation data of copper

Task: *Activate a copper sample for about 10 min at 2 W reactor power in the tangential experimental channel no. 3 of the AKR. The end of the irradiation time marks the start of the decay time (use a stop watch for the time measurement!). Pre-select an appropriate LifeTime at the gamma spectrometer (see also Sec. 3.1) and measure a few gamma spectra for several decay times each. Relate the measured photopeaks to the two copper isotopes. Determine the net peak areas (that correspond to the intensities) of the photopeaks (see Sec. 3.2). Write the results and also the standard deviations into a table according to Tab. 3 and calculate the ratios of the peak areas of the photopeaks Cu-64/Cu-66. What are conclusions of the measurement?*

Decay time [min]	LiveTime of measurement [s]	Net peak areas [pulses]		Ratio of the peak areas Cu-64 / Cu-66
		short-lived isotope Cu-66 (5.1 min) $E\gamma$ = 1039 keV	long-lived isotope Cu-64 (12.8 h) $E\gamma$ = 511 keV	

Tab. 3, Protocol of the peak areas of the photopeaks of an activated copper sample

2.4. Identification of an Unknown Radionuclide or a Radionuclide Mixture

The nuclide identification of an unknown radioactive sample is the measurement of the gamma energies of the photopeaks and the comparison of the results with the literature. Because all radionuclides have a unique decay scheme (example in Fig. 1) a definite identification is possible. The data of all known radionuclides are available graphically or tabulated in printed or digital form.

In the case that all photopeaks of a measured spectrum can be related to one single nuclide, a pure isotope was measured. If additional photopeaks are identified, these are related to other nuclides and consequently the sample is an isotope mix.

For simple nuclide identification as done in the described reactor training experiment, only the ingredients of the sample have to be determined. An advanced analysis (see Sec. 2.5) allows to calculate the quantitative composition of the sample by analysing additionally the photopeak areas of the spectrum (taking into account the absolute energy-dependent efficiency of the spectrometer, the gamma-emission probability and, if necessary, the activation parameters).

In the training at the AKR, either an unknown sample gets activated in the reactor or an already present long-lived nuclide is measured.

Task: *Measure the gamma spectrum of a sample that is provided by the supervisor using the Ge(Li)-semiconductor detector. If necessary, the sample has to be activated in the experimental channels of the AKR before the measurement. The energies of the measured photopeaks have to be recorded in a table according to Tab. 4 (column 1). Using the provided literature data at the experimental place also write both, the gamma energies and the possibly matching nuclides into this table (columns 2 and 3). To account for a slight shift in the energy calibration include also neighbouring gamma energies for comparison, i.e. one keV higher and lower of each measured photopeak. What is the result of your nuclide identification?*

Measured photopeaks Eγ [keV]	Possibly matching nuclides from the literature	
	Eγ [keV]	Nuclide(s)

Tab. 4, Protocol sheet for nuclide identification

2.5. Determination of the Absolute Efficiency of the Ge(Li)-Semiconductor Detector and Measurement of the Activity (part of the extended experiment only)

The number of gamma quants of a particular energy emitted by the source differs from the number of detected pulses at this energy in the detector, because

- not all emitted quants penetrate the sensitive volume of the detector (geometry effect),
- not all quants that penetrate the volume of the detector interact within the sensitive volume (detector efficiency effect), and
- not all interactions lead to a complete release of the quantum energy and consequently the resulting pulse does not contribute to the photopeak area of the actual gamma energy (i.e. Compton effect).

The efficiency calibration determines the quantitative relation between the number of gamma quants emitted by the source and the number of pulses being counted in the corresponding photopeak by the detector. The detector efficiency is strongly dependent on the photon energy. At low gamma energies, much more pulses can be registered than at high gamma energies (due to energy dependence of interaction cross sections). Additionally, the efficiency is influenced by the material and the dimensions of the detector, by the geometry of the measurement setup and by the distance between source and detector.

The number of photons that are emitted by the source per time unit is

$$N_{emitted} = A \cdot \gamma \qquad \textbf{(3)}$$

with A being the activity of the source (i.e. the number of nuclei that decay per time unit) and γ being the emission probability (or intensity). The emission probability is the fraction of photons which has a certain energy and which is released by decays of the corresponding nuclei. These intensities are provided by the decay scheme of the respective nuclide or appropriate tables.

The number of detected pulses per time unit can be calculated from the measured net peak area and the LiveTime according to

$$N_{detected} = \frac{N}{t_{LT}} \qquad \textbf{(4)}$$

The efficiency at a certain gamma energy is the ratio between the numbers of detected and emitted photons:

$$\varepsilon = \frac{N_{detected}}{N_{emitted}} = \frac{N}{t_{LT} \cdot \gamma \cdot A} \qquad \textbf{(5)}$$

ε = efficiency
N = measured net photopeak area
t_{LT} = LiveTime of the measurement
γ = photon emission probability
A = activity of the gamma source at the time of the measurement

The detector efficiency at various energies is calculated using this formula for several measurement geometries (e.g. using point sources in a certain distance from the detector or using standardised vessels with a constant volume for liquid samples). If the geometry setup of the measurement is changed, the efficiency calibration needs to be updated.

When plotting the detector efficiency versus the gamma energy with log-log scale, the values get arranged linearly in a wide energy region. Thus, the relation between detector efficiency and gamma energy can be expressed as

$$\ln \varepsilon = a + b \cdot \ln E \qquad \textbf{(6)}$$

For minimising the error of the efficiency calibration, a linear fit should be generated from an appropriate set of pairs of values. This can be achieved by using nuclides with multifold gamma lines or by using a set of several calibration sources. In any case, the absolute activity of the source at the time of calibration needs to be known.

After obtaining the coefficients a and b in equation (6) by an appropriate calibration, the absolute activity of an unknown sample can be calculated from the measured gamma spectrum by rearrangement of equation (5).

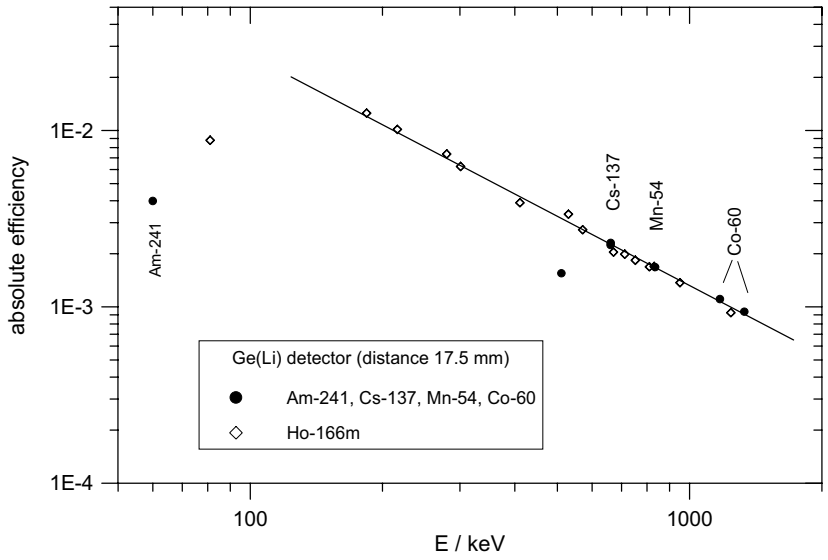


Fig. 11, Example of an absolute efficiency calibration of a Ge(Li) semiconductor detector

Examples of calibration sources are given in Tab. 5. The activities of the sources at the time of their production that are needed for the efficiency calibration can be found in the corresponding certificates. These activities need to be converted to the actual activities present at the time of the measurement using the half-lives of the nuclides. An example of an energy calibration is shown in Fig. 11.

Calibration source	Half-life [a]	Gamma energy [keV]	Intensity [%]
Am-241	432.1	60	35.90
Cs-137	30.17	662	89.92
Mn-54	0.856	835	99.98
Co-60	5.27	1173 1332	99.90 99.98

Calibration source	Half-life [a]	Gamma energy [keV]	Intensity [%]
Ho-166m	1200	81 184 216 280 301 411 530 571 671 712 752 810 831 951 1241	14.00 100.00 3.80 40.79 4.80 15.25 10.00 6.80 7.00 74.48 16.00 78.66 12.00 3.60 1.20

Tab. 5, Examples of gamma calibration sources (gamma energies and intensities of the lines)

Task: *Determine the energy-dependent absolute efficiency of the Ge(Li)-semiconductor detector for a given geometry and present the results in a diagram according to Fig. 11. Calculate the fit function according to equation (6) and give the approximate range of validity with respect to energy. Determine the absolute activity of a given radioactive sample using the result of the efficiency calibration and by rearrangement of equation (5).*

3. Operation of the Gamma Spectrometer

3.1. Main Control Keys

			Key
Measurement	Start		Alt-1
	Stop		Alt-2
	Delete		Alt-3
Display	Vertical scaling of the spectrum		Cursor buttons up/down
	View the whole spectrum/cursor region		F3
	Shift measured spectrum to buffer		Alt-5
	Switch between measurement display and buffer		Alt-6
Cursor	Cursor movement	slow	Cursor buttons left/right
	"	fast	PageUp / PageDown
	"	to begin/end of spectrum	Pos1/End
	Cursor region	wide	+ (NumPad)
	"	narrow	- (NumPad)
Save	Shift spectrum to buffer		Alt-5
	Pull-down menu "Files"		Alt-F
	Save		Alt-S
	Type in file name (without extension)		...
	Write a comment (if needed)		...
Presets	Pull-down menu "Presets"		Alt-P
	(e.g.) LiveTime preset		Alt-L

3.2. Calculation of the Peak Areas of Photopeaks

The area of a peak is the difference between its entire area and the respective background. The background comprises all pulses that do not belong to the photopeak but have been registered by the multichannel analyser in the channels of the peak. The main reasons for the background are:

- natural radiation at the place of measurement (e.g. cosmic radiation, terrestrial radiation),
- artificial radiation at the place of measurement (e.g. operation of the reactor, storage of radiation sources close to the place of measurement),
- background of Compton scattering originating from higher gamma energies of the sample being analysed.

The first two components of the background can be minimised by shielding the gamma detector (e.g. with lead) and/or by storing other gamma sources as far as possible from the place of measurement. If the sample being analysed has more than one gamma line, the third background component is often dominating and cannot be avoided.

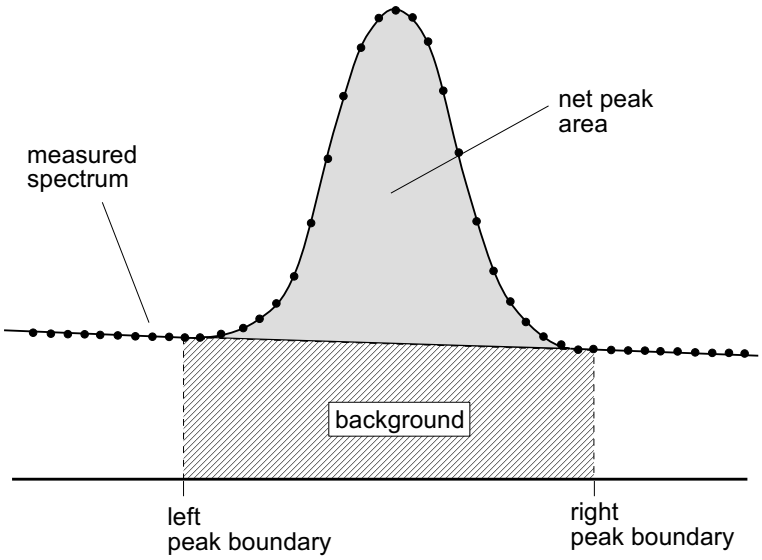


Fig. 12, Principle of net peak area determination

For correcting the background, the "ceiling" is assumed to have a linear shape (see Fig. 12). In this case, it can be written

$$B = \left(\frac{C_1 + C_n}{2} \right) \cdot n \qquad \textbf{(7)}$$

$$A_g = \sum_{i=1}^{n} C_i \qquad \textbf{(8)}$$

$$A_n = A_g - B = \sum_{i=1}^{n} C_i - \left(\frac{C_1 + C_n}{2} \right) \cdot n \qquad \textbf{(9)}$$

B = area of the background
A_g = gross peak area
A_n = net peak area
C_1 = content of the channel at the left boundary of the peak
C_i = content of the i-th channel
C_n = content of the channel at the right boundary of the peak
n = number of channels between left and right boundaries

The software of the multichannel analyser can calculate automatically the gross and net peak areas as well as the respective standard deviations, which are due to the counting statistics. For these calculations, the following **keyboard commands** are necessary:

1.	Shift the measured spectrum to the buffer	Alt-5
2.	Display the content of the buffer	Alt-6
3.	Place the cursor at the left boundary of the peak	
4.	Mark the left boundary of the peak	Alt-R, then Alt-B
5.	Place the cursor at the right boundary of the peak	
6.	Mark the right boundary of the peak	Alt-R, then Alt-E (peak sector is in red color, now)
7.	Move the cursor inside the peak range	
8.	Calculate net peak area	Alt-C, then Alt-A

The net peak area is calculated by the PC program according to following method. For balancing statistic errors at the lower and upper limits of the chosen peak range, the calculation of the background B averages both, the first and the last 3 channels inside the peak range and uses them for calculating the "ceiling". Hence,

$$B = \left(\sum_{i=1}^{l+2} C_i + \sum_{i=h-2}^{h} C_i \right) \cdot \frac{h - l + 1}{6} \qquad \textbf{(10)}$$

$$A_b = \sum_{i=l}^{h} C_i \qquad \textbf{(11)}$$

$$A_n = A_{ib} - \left(\sum_{i=l}^{l+2} C_i + \sum_{i=h-2}^{h} C_i \right) \cdot \frac{h - l - 5}{6} = A_b - B \cdot \frac{h - l - 5}{h - l + 1} \qquad \textbf{(12)}$$

B = area of the background
l = channel number at the left boundary of the ROI (**r**egion **o**f **i**nterest)
h = channel number at the right boundary of the ROI
C_i = content of the i-th channel
A_b = gross peak area
A_{ib} = inner gross area (without both, the first three and the last three channels)
A_n = net peak area
σ_{An} = statistical error of the calculated net peak area

Time-dependent changes in the isotope activities in an activated copper sample

Decay time [min]	Measuring time (LiveTime) [s]	Peak areas [pulses]		Ratios of peak areas Cu-64 / Cu-66
		short-lived isotope Cu-66 (5.1 min) Eγ = 1039 keV	long-lived isotope Cu-64 (12.8 h) Eγ = 511 keV	

Determination of peak areas:

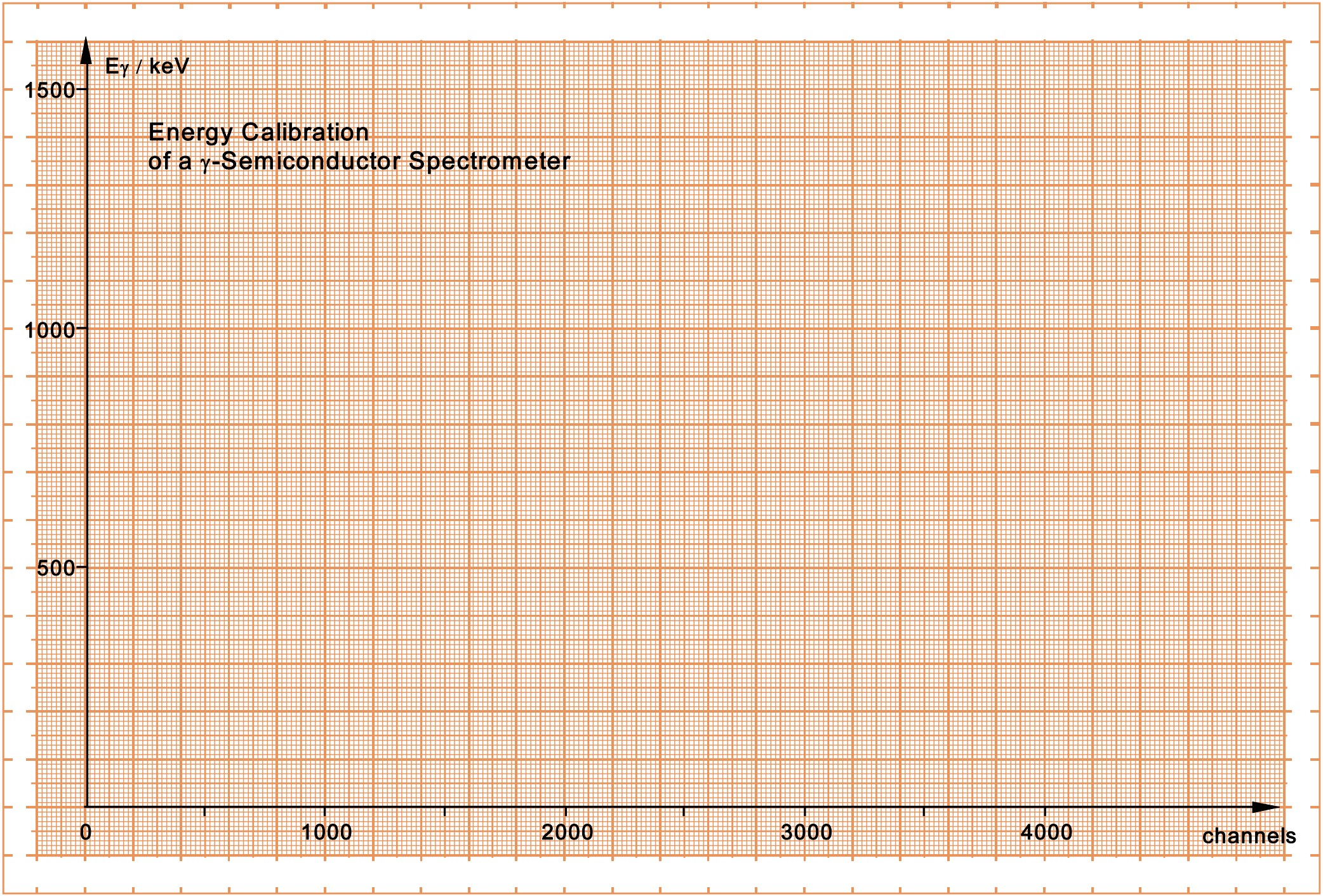
1. Shift measured spectrum into the buffer memory (Alt-5)
2. Show buffer content (Alt-6)
3. Move cursor to left boundary of the peak
4. Fix left boundary of the peak (Alt-R, afterwards Alt-B)
5. Move cursor to right boundary of the peak
6. Fix right boundary of the peak (Alt-R, danach Alt-E)
   (channels in peak region are displayed in red color)
7. Move cursor into the peak region
8. Calculate peak area (Alt-C, danach Alt-A)

Results:

Identification of an unknown activated sample

Measured γ-peaks Eγ [keV]	Possible nuclides	
	Eγ [keV]	Nuclides

Results:


Eγ / keV
1500
1000
500
Energy Calibration
of a γ-Semiconductor Spectrometer
0
1000
2000
3000
4000
channels