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Research Reactor Exercises for Higher Education Programmes

Compendium

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RESEARCH REACTOR EXERCISES FOR HIGHER EDUCATION PROGRAMMES

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

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INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

The IAEA, via a Peaceful Uses Initiative project aimed at increasing the global supply of nuclear education and training programmes through research reactor facilities, has developed this compendium to make available reference materials for improved academic curricula at higher education institutions. It is intended to provide practical information on the development of research reactor exercises to be integrated into education programmes in nuclear science and technology. The information can benefit countries engaged in educating human capital for active or future nuclear programmes, whether intended to deploy nuclear power or for other peaceful applications of nuclear science and technology. The publication can also be used as a tool to promote and enhance the safe use of research reactors in the field of education and training.

Participants from more than 30 Member States contributed to the development of the publication and to the facility descriptions and experimental protocols included herein. The compendium provides the background and practical guidelines for the development and implementation of research reactor exercises. The exercises described are aimed primarily at advanced undergraduate or postgraduate level nuclear engineering or nuclear physics students. Nevertheless, with proper adaptation of the content and methodology, they can also be extended to education for non-nuclear students or to education and training of young professionals in the nuclear field.

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1. INTRODUCTION

1.1. BACKGROUND

Countries worldwide are pursuing or expanding their peaceful applications of nuclear technologies according to their national objectives. On the one hand, countries with long standing nuclear programmes need to maintain their capacity. On the other hand, countries embarking on nuclear programmes need to develop their knowledge and capabilities by initiating nuclear programmes or increasing interest in broader applications of nuclear science and technology.

For many countries, research reactors (RRs) have been and are a first step in their preparation for a national nuclear power programme. For other countries, RRs have been built to support various neutron applications, such as but not limited to basic and applied research, production of radioactive isotopes for medicine and industry, and characterization or testing of materials and samples for industry, archaeology, environmental studies and many other applications [1].

Regardless of the final goal (national nuclear power or nuclear science & application programmes), RRs are excellent tools to support nuclear capacity building, from academic education to training of scientists, engineers and technicians in many areas of nuclear science and engineering. In this matter, RRs play a role of paramount importance to link the theoretical knowledge gained in classrooms to practical hands-on experience.

This Compendium is developed with the aim of providing comprehensive overview of the utilization of RRs for education purposes in an academic environment. It compiles extensive references applicable to higher educational institutions including good practices and lessons learned from the IAEA Member States (MSs) in regard to nuclear capacity building with the objective to share and optimize RR use within and amongst MSs. As a result, this Compendium is meant to help in developing academic curricula based on the practical use of RRs, contributing to the preservation and the development of nuclear capabilities. It also gives reference to the basic safety concepts and standards in order to raise the awareness to the operational and educational aspects related to safe operation and utilization of RRs. Additionally, the general background and practical guidelines presented in the Compendium can be used with minor adaptation for the development of RR exercises dedicated to training of professionals in nuclear related fields.

1.2. OBJECTIVE

The Compendium is intended to serve as a reference which can provide insights to those working or intending to work on developing education or new educational exercises using a RR. Its main objective is to provide high level resource material for RR exercises[1] to support and improve experimental education and hands-on experiences at university level in nuclear science and technology. Furthermore, this resource material can also be used as starting point for the development of hands-on training activities dedicated to professionals in nuclear sector. In addition, the Compendium can be used to:

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[1] In this publication, RR exercises stands for RR presentation, demonstration, operation sequences and experiments (with known experimental results), which are performed for educational purposes to present, illustrate, demonstrate and discuss the principles and the practical aspects of RR operation and its utilization.

(a) Provide useful guide for MSs embarking on a new RR project to better identify the capabilities of each type of RR for educational purposes;
(b) Guide university professors and lecturers in identifying RR exercises that could be included in their academic curricula;
(c) Help RR managers to identify potential stakeholders and users in academia and to set up or further develop RR exercises at their facility.

1.3. SCOPE

The Compendium covers theoretical and practical aspects of RR exercises tailored toward academic education. It includes general background on the use of RRs as educational tools. It provides theoretical background and guidelines on how to develop 18 RR exercises in nine areas, ranging from introductory exercises to exercises dedicated to specific aspects of RR operation and applications. Since safety is a main pillar of the operation and utilization of RRs, this publication presents, in a synthetic way, the basic safety principles to raise the awareness of the reader on the various aspects of nuclear safety to be considered when developing RR exercises.

1.4. STRUCTURE

The Compendium is organized in twelve Sections and one Annex.

Section 1 presents the background, purpose and intended use, as well as the scope and structure of the Compendium.

Section 2 presents general background on how to use RRs as educational tools for human capacity building. After a brief description of different types and categories of RRs available worldwide, this Section gives background related to standard RR exercises commonly integrated in academic curricula, providing specific details on educational aspects of RRs. It also provides details on the types of students and curricula that can benefit from the use of RRs as educational tools and gives, in the form of simple tables, the relationship between reactor exercises and university curricula, as well as the relationship between exercises and requested RR power.

Sections 3 to 12 provide a comprehensive overview of the topics that can be addressed through practical exercises carried out at a RR. They provide general concepts and theoretical background related to standard RR exercises conducted for educational purposes, establish the associated learning outcomes for the students and describe the main principles of their integration into various types of academic curricula. Each Section also includes a guideline to perform exercises that can be used as a step-by-step guide to (further) develop RR exercises to a specific RR and to establish their corresponding protocols.

Section 3 addresses nuclear safety at RRs with related exercises, including basic concepts of nuclear safety, radiation protection and waste management. Section 4 is dedicated to RR instrumentation and control (I&C) systems. It focuses on neutron instrumentation typically used for the exercises. Section 5 focuses on neutron flux measurements, namely on neutron flux mapping as a typical exercise. Section 6 deals with criticality experiments, focusing on approach to criticality experiments. Section 7 is dedicated to reactor kinetics and its related exercises. Section 8 addresses reactivity control. It focuses on control rod calibration and safety parameters related to core reactivity and the influence of core components to reactivity. Section 9 is dedicated to reactor dynamics, presenting exercises on void effect, temperature effect and long-term reactivity feedback. Section 10 focuses on reactor power calibration, including the calorimetric method and heat balance. Section 11 is dedicated to neutron activation analysis (NAA). Finally, Section 12 presents applications of neutron irradiation including neutron radiography, radioisotope production, radiotracer analysis, nuclear chemistry and radiochemistry, neutron transmutation, geochronology and neutron beam experiments.

2. UTILISATION OF RESEARCH REACTORS IN EDUCATION

2.1. EDUCATION IN NUCLEAR SCIENCE AND ENGINEERING

One of the most pressing issues accompanying the peaceful use of nuclear energy worldwide is the need for experts and highly educated and skilled professionals in nuclear engineering, science and applications. These issues are caused by several factors. The first is the ageing of the staff in the existing nuclear power- and nuclear non-power programmes, which call for the constant renewal of the human capacity. The second factor is the need to develop human capacity for key operational staff either for countries expanding their nuclear programmes or for countries embarking on nuclear programmes. The construction of new RRs or nuclear power plants (NPPs) and the national development and expansion of other areas of nuclear science and technology result in the need of nuclear professionals all over the world.

An additional factor to consider is the ageing of RR staff and the RRs themselves, together with a trend of slow decrease in the number of RRs in operation. Indeed, RRs are an important source for capacity building, such as nuclear knowledge, experience, skill and qualified human resources, which all are needed for nuclear programmes (including nuclear power). In many countries, social factors are also to be considered, such as interest and availability of young people to enter nuclear careers. In particular, women are under-represented in Science, Technology, Engineering and Mathematics (STEM) education courses and career paths, and women make up less than a quarter of the workforce in the nuclear sector worldwide, which is detrimental to the industry's competitiveness and diversity [2].

All of these factors contribute to the lack of experts and highly educated and skilled professionals in nuclear sector. This problem presents new challenges to universities. During recent years, some new global trends in nuclear education have emerged, bringing new challenges and opportunities to universities and RRs.

One trend is connected with the demand of high quality nuclear education in a wide range of areas. Employer demand for graduates is very often one of the main motivations for the students choosing a particular university and academic programme. This potential employment in the nuclear sector involves, for example, the operation and maintenance of nuclear installations and other nuclear facilities, nuclear research as support to safe operation of nuclear installations and other nuclear facilities, technical expertise in support of national regulatory bodies, as well as staff renewal including professors and researchers at universities.

Another trend that can be seen worldwide is an expansion of research and development (R&D) at universities. Fast development of nuclear technology, nuclear applications and nuclear science brings new opportunities for universities, which are centres for basic and applied research, to meet the demands for new researchers.

Finally, the demand coming from nuclear industry for highly qualified courses and practical training for young professionals, lifelong learning for key personnel, and various short-term courses is increasing. Universities can also meet this demand and play an important role in practical staff training for nuclear installations and other nuclear facilities, national regulatory bodies and technical support organizations.

Lectures and courses at the universities are in general supplemented with practical exercises at experimental facilities. In nuclear education, the most appropriate experimental facilities are

RRs, representing attractive and high quality experimental tools for education, bringing new potential for attracting the students to study nuclear science and engineering.

The two words in the term ‘education and training’ are often used as synonyms, but education and training are two separate disciplines. Both disciplines use the same or very similar pedagogical methods, instruments and experimental equipment. In principle, they are very different from the view point of the target audience, and the range of knowledge transferred to the audience.

In the context of this publication, *Education* is a broad term that will be used only connected with students. During the educational process students have to obtain a broad overview of the studied field, together with specific knowledge of a specialist subject for any academic level, i.e. bachelor’s, master’s and doctoral.

Training is generally connected to a profession, and its main goal is to prepare professionals for specific tasks. It can apply to training of young professionals at the start of their career, as well as experienced workers for whom a lifelong retraining programme is required according to their positions. In the context of this publication, training is meant to entail mostly short-term courses with well-defined objectives. Preparation of training courses needs to take into account both initial training and regular refresher courses. The method of systematic approach to training is generally used in the nuclear sector, and aims at ensuring high quality outcomes.

The conditions and capabilities existing at RRs to perform experiments for education are also suitable for training; therefore, this Compendium can also be used for the development of RR exercises for training in nuclear science and technology.

The specific learning outcomes that can be obtained through education and training (E&T) on RRs have been reviewed in reference [1] for the particular case of nuclear engineering, showing that RRs provide specific inputs to education in that area, by:

(a) Linking theory and the actual events that may occur in reactor operation: address operational safety, and give practical background to understand the limitations in reactor operation which are, in turn, used to define operational limits and conditions (OLCs) or the characteristics of safety systems such as reactor protection system;
(b) Giving an example of the operator’s work: demonstrate its role, show a professional approach to the work and understand the importance of safety culture;
(c) Seeing what a reactor is: reactor building, reactor core, reactor control and monitoring system, shutdown system, main cooling systems, air conditioning and water make-up systems, facility instrumentation including sensors and systems used for reactor and environmental control and monitoring, and other systems;
(d) Obtaining experience on the constraints and real physical conditions in the operation of a nuclear facility, learn about mandatory documentation, procedures and identify their relation to safety, security, radiation protection and safety culture;
(e) Learning how to use specific measuring devices and determine the accuracy of experimental results;
(f) Carrying out in-depth studies on particular aspects related to reactor physics and operation (i.e. fuel loading plan and associated calculations, impact of moderation factor, between others.);
(g) Verifying theoretical concepts through experiments and learn about limitations of simulation and modelling;
(h) Self-operating a RR under the supervision of operational personnel.

Thus, the educational exercises with RRs provide practical learning and experience for a nuclear engineering programme, such as gradual buildup of knowledge, competences and skills, and practical experiments. The positive feedback obtained from students after RR laboratorial classes emphasises this, pointing out the importance placed in being able to have a direct contact with a nuclear reactor through the completion of the exercises and operation.

2.2. POTENTIAL USERS

One group of Compendium users are university professors and lecturers, providing education in nuclear science and technology, who have access to a RR. They can use this Compendium as a guide to extend educational programmes by adding an experimental component using the RR. Incorporation of experimental education to any academic curriculum is a long-term process which needs profound discussion between academic staff at the university and reactor operating staff.

Researchers working at universities, research institutions or working directly at a RR can use this Compendium to assist the work of young collaborators, i.e. students at bachelor, master or doctoral degree level, therefore enabling their research activities.

Students from various universities can use this Compendium as a reference during their academic study at all academic levels, i.e. bachelor’s, master’s and doctoral degrees.

Finally, policy decision makers at RR operating organizations, reactor managers and reactor lecturers who can use this Compendium to find ways of further extending the utilization of their RR and associated facilities.

As previously stated, to a broader extend, this Compendium can be used by organizations and trainers wishing to develop or to enlarge the scope of hands-on training activities using RRs for professionals who work in fields involving nuclear science and technology.

2.3. STUDENTS AT A RESEARCH REACTOR

2.3.1. Integration of research reactor exercises and university curricula

Research reactors are suitable for education at all academic levels, i.e. bachelor’s, master’s and doctoral. They can offer a wide portfolio of exercises which can be incorporated to a broad range of curricula, including cross-cutting along different disciplines. In general two main types of educational exercises can be performed at a RR.

The first type of exercises is related to the reactor itself as a complex engineering installation. Studying RR characteristics either in steady state or in transients allows students to understand not only the basic principles of RR operation and underlying physics and phaenomena, but also general concepts related to NPP principles and operational approaches. This group of exercises includes those described in Sections 4–10. These exercises are highly suitable for students studying nuclear engineering as their major. They are also suitable for students studying other nuclear related disciplines such as but not limited to radiological engineering, radiation protection, radiochemistry and nuclear chemistry, nuclear energy, nuclear safety and security and nuclear waste management, as well as for students studying nuclear relevant disciplines such as but not limited to power engineering, mechanical engineering and electrical engineering.

The second type of exercises is related to the use of a RR as a source of radiation, i.e. mainly neutron and gamma radiation. Neutrons from RRs can find applications in R&D in nearly every field, in industry, agriculture, in cultural heritage preservation, as well as environmental, food and health studies. The exercises described in Sections 11 and 12 belong to this group. In particular, this information is highly suitable for students in neutron science, neutron applications, or non-destructive techniques as their major discipline and for students in archaeology, geology, biology, earth and environmental sciences, cultural heritage and forensic sciences, and other subjects.

2.3.2. Level of study programmes

The level of the exercises (basic, intermediate or advanced) and their duration strictly depends on the curriculum of each group of users. The basic level usually covers exercises that are carried out only as demonstrations or observations, without the active involvement of students. The intermediate level usually covers exercises where a certain amount of active student work is requested, including experimental set-up, measurement and analysis of measured values. The advanced level usually covers exercises that are carried out under different conditions, using various methods, and allowing a deeper understanding of the phenomena under investigation. Extensive student involvement at the advanced level is requested for calculations, measurements, analysis of measured values and, in some exercises, a study of the regulatory framework and operating procedures applied for the reactor use.

The matrix showing the relationship between the reactor exercises and university curricula (refer to Section 2.6) can be used as a preliminary guide for those interested in using RRs for education. The incorporation of experimental education to any curriculum will need a close collaboration between the academic staff at university, the reactor operating staff and reactor management.

The importance of the safety aspects related to the safe operation and utilization of RRs is emphasized throughout this Compendium. This is the reason why nuclear safety and radiation protection aspects are explained in Section 3. Indeed, nuclear safety and radiation protection needs to be understood and effectively applied by all counterparts involved in educational activities at a RR, i.e. the professors, lecturers and students. This ensures that safety culture and good practices are properly followed and applied.

2.4. USE OF RESEARCH REACTORS AS EDUCATIONAL TOOLS

More than 840 RRs have been built and operated since 2nd December 1942, when the first reactor, called Chicago Pile 1 reached its first criticality. Currently, some 235 RRs in 54 countries are still in operation. Research reactors, together with NPPs are the most common nuclear installations in the world.

The general term ‘research reactor’ covers a wide range of non-power reactors, from subcritical assemblies through critical assemblies and low power RRs to high power RRs. The definition of RR varies, and there is no single definition. The most common definition used by the IAEA is based on potential use of a RR:

“A research reactor is a nuclear reactor used mainly for the generation and utilization of neutron flux and ionizing radiation for research and other purposes, including experimental facilities associated with the reactor and storage, handling and treatment facilities for radioactive materials on the same site that are directly related to safe operation of the research reactor. Facilities commonly known as critical assemblies and subcritical assemblies are included" [3].

Similarly, the IAEA Safety Glossary (2016) defines critical assembly as "an assembly containing fissile material intended to sustain a controlled fission chain reaction at a low power level, used to investigate reactor core geometry and composition" [4].

Research reactors have been constructed as thermal or fast reactors, mainly as heterogeneous but also as homogeneous systems. Almost all RRs can be classified into three major categories: pool type reactors, tank type reactors or tank-in-pool type reactors. Since the early 1950s, dozens of RRs have been constructed. Many of them have a unique construction and are represented by a single reactor only, while some RRs have been built at many locations and in several countries. Some RRs commonly used in E&T activities include [5] Aerojet General Nucleonics (AGN), Argonne's Nuclear Assembly for University Training (ARGONAUT), Miniature Neutron Source Reactor (MNSR), Safe LOW-Power Kritical Experiment (SLOWPOKE), Siemens-Unterrichtsreaktor (SUR), Water-Water Reactor (WWR), Standardized Research Reactor (IRT, from the Russian "Исследовательский Реактор Типовой"), and Training, Research, Isotopes, General Atomic (TRIGA) RRs, etc.

There is no single, global classification of RRs according to their neutron flux or thermal power, which are equivalent. Two classifications used by the IAEA can be considered in this Compendium. In the IAEA Nuclear Energy Series No. NP–T–5.3, 'Applications of Research Reactors' [6], five power limits are defined according to the potential applications of RRs: up to 1 kW, up to 100 kW, up to 1 MW, up to 10 MW and above 10 MW. These power limits reflect the potential application of RRs, and this classification is useful for those interested in RR utilization. However, the IAEA Research Reactor Database (RRDB) implements a classification of RRs based on their power as follows: lower than 1 kW, higher or equal to 1 kW and lower than 1 MW and higher or equal to 1 MW [5].

The RRDB also uses a classification of RRs based on the neutron flux available for utilization at the RR. A low flux RR has a neutron flux of up to 10^{12} $cm^{-2}s^{-1}$; a medium flux RR has a neutron flux in the range of 10^{12} $cm^{-2}s^{-1}$ and 10^{14} $cm^{-2}s^{-1}$ and a high flux RR above 10^{14} $cm^{-2}s^{-1}$. These flux levels also define possible RRs applications. As the relation between neutron flux and power is highly dependent on the design of the reactor, an exact correspondence between the flux and the power cannot be established. A rough correlation can be used as follows: low flux RRs generally have a power rating up to 1 MW, medium flux reactors are in the range 100 kW to 20 MW and high flux reactors have power ratings above 10 MW. When considering educational activities particular aspects have to be taken into account. A large number of reactor exercises requires performing dedicated operating sequences or operating the reactor at a power level that can be considered as 'low power'. This is the case when performing exercises at powers below 1 kW to exclude temperature feedback effects, or 100 kW to exclude poisoning effects. Because medium flux and high flux reactors have high operational costs and are often involved in multiple activities, they may allow for a limited flexibility for educational activities. Taking into account these different aspects, the following terminology has been adopted in this Compendium. 'low power', 'medium power' and 'high power' referring to RRs with power of up to 1 MW, 1–10 MW and above 10 MW, respectively.

Research reactors are different from nuclear power reactors, and they are very diverse in their design, mode of operations, utilization, and associated risks. Research reactors operate at a lower power level, lower temperature and lower pressure than NPPs. There is also an essential difference in the construction of RRs, which are designed for operation in power ranging ten orders of magnitude (from mW up to hundreds of MW), and some RRs can operate both in steady state (standard mode) or in pulse mode. Another specific feature of RRs is their flexibility of operation (frequent different core configurations and power changes, frequent start-up and shutdown sequences), flexibility in the experimental instrumentation, and frequent changes to the instrumentation. Another specific feature of RRs, mainly low power ones, is that they are usually operated by a small group of reactor staff, comprising researchers and lecturers, and often also including students.

From the safety point of view, the activities carried out at RRs need particular attention since this type of nuclear facilities is subject to frequent and significant changes in their characteristics, including but not limited to:

(a) Handlings of reactor core components in the core or near the core;
(b) Important reactivity worth ;
(c) Fast and frequent changes in RR power;
(d) Delay in negative feedback effects at low power levels, i.e., below a few kW;
(e) Influence of experimental instrumentation and samples in the core and reflector;
(f) Potential interactions between experiments and reactor operation.

2.5. SPECIFIC REQUIREMENTS IN EDUCATION AT RESEARCH REACTORS

Based on long term experience in many MSs, education at RRs has specific requirements that can cause difficulties that may limit or even impede the establishment of an educational programme at a RR. Those difficulties can be of varied nature including available RR operational time used for education purposes, availability of RR staff, access to and availability of instrumentation, as well as economical aspects involved when running the RR solely for educational purposes.

Education could be an important source of income only for low power university reactors. Universities often lack funds to pay the running costs of RRs for educational purposes. Financial support for education is usually needed through government or industry, or from national and international research and educational grants and projects. The number of reactor staff at low power RRs or critical assemblies is usually very small, and every member carries out several duties connected with RR operation. For university operated RRs, professors and lecturers from the institution may join the RR operating staff as lecturers, and in some cases as part of the operating team.

Usually RRs are used for several different applications, and only a handful of them in the world are focused solely on education. In the strategic planning for utilization of a RR it is very important to address short term as well as long term priorities. A combination of several applications requires different types of instrumentations, which means more expensive investments, higher running costs and costlier maintenance and upgrades. It requires also more staff, e.g. lecturers for education, specialised researchers for R&D, reactor operation staff for specific activities, for example in radioisotope production, and, in case when RRs are significantly involved in the provision of commercial products and services, marketing staff may also be needed. The availability of more areas of application of a given RR also leads to different types of customers, with different work style and budgets, e.g. universities, research institutes, governmental and international organizations or industrial companies.

Education at a RR has several limitations that need to be resolved if management is considering the provision of education using the RR. The RR utilization time for education purposes is limited to that required for the lectures and experiments and needs to be coordinated with the planning of other activities. Typically, the operation scheme needed for education activities have a typical pattern in which peak hours are Monday to Friday from 9:00 to 13:00 and 14:00 to 17:00 during the academic year.

Planning RR operation schedule, including E&T activities, has to be coordinated taking into account other activities performed at the facility (research, irradiations, maintenance, or others). This is, together with the high operational costs, one of the reasons why high power multipurpose reactors are not commonly used to perform educational activities on a regular basis, with the exception of participation of graduate students in R&D projects.

Education at a RR is specific and differs from scientific and research work at RRs. Due to the investment and running cost of a RR, education at a RR is expensive compared with other types of laboratories at university. This means that the use of a RR for educational purposes has to be planned effectively, to avoid incurring in high and unnecessary running costs. State-of-the-art experimental equipment at universities and RRs and methodologies specifically developed for E&T are trends that can be noticed today all over the world. Providing an effective education at a RR involves adapting the educational methodology to the initial background of the students and using adequate experimental educational instrumentation. This ensures effective student education at the RR. To cope with high RR operating costs, it is necessary to use RR operation time as efficiently as possible.

Specific effort and sometimes specific educational instrumentation is needed for education of bachelor's and master's degree students. Standard experimental equipment can be either too complicated or its functioning principles are not clear for students. In this case, special educational instrumentation needs to be developed or acquired. It is best to use simple instrumentation to perform the experiment which should be focused on demonstrating one specific phenomenon at a time, and visualising or illustrating the studied phenomenon. Doctoral students are supposed to possess sufficient knowledge to understand complicated and complex phenomena, and standard experimental equipment for research may commonly be used.

When performing an educational experiment, it is effective to use several parallel measurement lines, each equipped with similar equipment. Each line can be used by an independent group of students, allowing several groups to perform the experiment at the same time, thus optimizing the use of RR time. As an example, flux measurements can be made with several detectors located at different positions, or several samples can be irradiated simultaneously, and their activity can later be measured in separate gamma spectrometers. The optimum number of students in one independent group usually is 2–3 because there is a lack of work for more students who may lose their interest, becoming bored or disturbing other students. The number of measurement lines is the main limiting factor for the number of students that could be accepted in such experiment. On the other hand, more measurement lines need more space, and more funds for investment.

A lecture room or a lecture zone is needed, where a lecturer can briefly repeat the necessary theory and describe the exercises. The best solution is to establish the lecture room (or zone) directly in the RR hall (or, if applicable, in the RR control room), where students can start practical exercises soon after the lecture is complete. If the lecture room is located in the RR building (or in the same building for those facilities sharing the building with other facilities), it is sufficient solution (although the entry procedure to RR hall may require significant amount of time that needs to be taken into account when programming the activities). Using a lecture room in another building is an ineffective solution because the logistics (e.g. putting on coats and shoes and walking to another building and repeating this process in the RR hall) would impair the students' ability to focus on the exercise.

For hands-on training on RR operation, enough space in the control room is needed. Some facilities can include a teaching control room. For several exercises, particularly studies about reactor kinetics and dynamics, direct access to operation data is needed; this may be done giving access to operation instrumentation, typically in the control room, or using information technology (IT). It is necessary to prepare for students upfront a hardcopy or electronic textbook containing a short introduction to theory, main goals of the exercises, measurement procedures, data processing and evaluation procedures including safety instructions that are needed before starting an educational activity. Visual differentiation between students and staff is recommended particularly in case of emergency. A tidy and clean RR hall and professional routine behaviour of lecturers and RR staff is mandatory in order to teach students safety culture. Practical aspects related to safe operation of a RR (radiation protection, safety and security issues) are an important part of education and should not be neglected.

It is said that 'a student's half-life is only one year', meaning that every academic year new students start working at the RR, and usually there is no sufficient time to perform several exercises with the same group of students. Exercises using RRs are often part of the standard university curricula. To carry out one reactor exercise, no more than three hours are generally allowed, in order to maintain the students' attention, for them to focus on the exercise and to ensure that teaching remains effective. Typically, no more than ten exercises are generally carried out within a course. Professors and lecturers are usually requested to carry out the same basic reactor exercises each year, which means that 80–90% of the time that is dedicated to education at RRs is filled with only a few exercises. The rest of the time is devoted to students with special interests. It is important to take this into account when developing the educational component of the use of RRs. High quality of educational exercises is essential for long term, effective and sustainable education at a RR. Focusing on developing a well-structured programme with a concise number of exercises will bring more benefits than putting efforts into developing a long list of exercises.

2.6. USE OF RESEARCH REACTORS IN EDUCATION

The following two tables can be used as a first guideline for developing educational exercises using a RR. It can also be a source of information for researchers working at universities, research institutions or RRs themselves, as well as for the students.

A simple matrix showing relations between reactor exercises and university disciplines is given in Table 1. The matrix showing relation between RR exercises and RR power is given in Table .

TABLE 1. MATRIX INDICATING THE INTENDED APPLICATION OF THE EXERCISES IN THIS COMPENDIUM IN SOME UNIVERSITY DISCIPLINES

Reactor exercise	Nuclear Eng.	Power Eng.	Mechanical Eng.	Electrical Eng.	Neutron Applications	Nuclear Sciences	Archaeology and Geology	Environmental and Earth Sciences	Biological Sciences	Cultural Heritage	Non Destructive Testing
Critical experiments	×	×	×	×							
Neutron flux mapping	×	×	×	×							
Reactor kinetics	×	×	×	×							
Reactor dynamics	×	×	×	×							
Long term reactivity feedback effects	×	×	×	×							
Reactivity measurements	×	×	×	×							
Reactor power calibration	×	×	×	×							
Reactor instrumentation	×	×	×	×							
Reactor safety (nuclear safety)	×	×	×	×							
Radiation protection	×	×	×	×	×	×	×	×	×	×	×
Neutron activation analysis	×				×	×	×	×	×	×	×
Neutron radiography					×	×	×	×	×	×	×
Radioisotope production, transmutation					×						
Nuclear chemistry and radiochemistry	×				×			×	×		
Radiotracers					×		×	×	×		
Geochronology					×		×				
Neutron beam experiments					×	×					

TABLE 2. MATRIX INDICATING RELATION BETWEEN THE EXERCISES IN THIS COMPENDIUM AND RR POWER

Reactor exercise	Subcritical assembly	Up to 1 kW	Up to 100 kW	Up to 1 MW	Up to 10 MW	Above 10 MW
Critical experiment		×	×	×		
Neutron flux mapping	×	×	×	×		
Reactor kinetics		×	×	×		
Reactor dynamics		×	×	×		
Long term reactivity feedback effects			×	×		
Reactivity measurements	×	×	×	×		
Reactor power calibration			×	×		
Reactor instrumentation	×	×	×	×	×	×
Reactor safety (nuclear safety)	×	×	×	×	×	×
Radiation protection	×	×	×	×	×	×
Neutron activation analysis		×	×	×	×	×
Neutron radiography		×	×	×	×	×
Radioisotope production, radiotracers				×	×	×
Nuclear chemistry and radiochemistry				×	×	×
Neutron transmutations			×	×	×	×
Geochronology				×	×	×
Neutron beam experiments			×	×	×	×

3. OVERVIEW OF NUCLEAR SAFETY AT RESEARCH REACTORS

As described in Section 2.3, nuclear safety should be well understood and effectively applied by all the counterparts involved in educational activities at RRs. This concerns the RR operating organization and the users, including professors and students. This Section provides an overview of the basic concepts of nuclear safety including the IAEA references, and addressing as well as radiation protection and waste management. The objective of this Section is to raise the reader's awareness regarding safety aspects in implementing and performing reactor exercises. It can be used as a base for dissemination of safety culture and good practices in reactor operation and utilization.

3.1. BASIC CONCEPTS OF NUCLEAR SAFETY

3.1.1. Background

Nuclear safety, which is often abbreviated as 'safety' in the IAEA publications, is defined in the IAEA Safety Glossary Terminology [4] as "The achievement of proper operating conditions, prevention of accidents and mitigation of accident consequences, resulting in protection of workers, the public and the environment from undue radiation risks.". Nuclear safety encompasses both risks under normal circumstances and as a consequence of incidents or accidents, including possible direct consequences of a loss of control over a nuclear reactor core, chain reaction, radioactive source or another source of radiation. Considering the high priority that must be given to safety, it is important to focus on all safety aspects related to reactor operation and utilization in any educational activity (including visits) conducted in reactor facility.

Safety is also commonly associated with security and safeguards[2] in relation to establishing an adequate legislative and regulatory framework to ensure the peaceful uses and prevent the non–peaceful uses of nuclear energy and ionizing radiation. These are included in the concept referred to as 3S (safety, security and safeguards), which can be applied to RRs and their activities. Safety, security and safeguards must be considered not only during the operation of a RR, but during all stages over its lifetime (planning, siting, design, manufacturing and construction, commissioning, operation, decommissioning, release from regulatory control or closure), to which should be added the transport of radioactive materials and management of radioactive waste. Depending on the curricula and pedagogical objectives, educational activities at a RR can address particular aspects of safety, security and safeguards.

It should be mentioned that safety and security have similar objective - to protect individuals, the public, and the environment from harmful effects of ionizing radiation. A properly managed interface between safety and security is therefore essential for ensuring the protection of people and the environment from security-related threats to, and radiological hazards associated with, RRs. According to the IAEA Fundamental Safety Principles, Safety Fundamentals No SF-1 (2006), "Safety measures and security measures must be designed and implemented in an

[2] According to IAEA Safety Glossary [4], the term security is defined as the "The prevention and detection of, and response to, criminal or intentional unauthorized acts involving nuclear material, other radioactive material, associated facilities or associated activities." The term *safeguards* is related to non-proliferation of nuclear weapons by early detection of the misuse of nuclear material or technology by providing credible assurance that each country is ensuring its safeguards obligations. Safeguards aspects have to be taken into account for example when performing reactor experiments involving fuel loading or unloading from a reactor.

integrated manner so that security measures do not compromise safety and safety measures do not compromise security".

Operation and utilization of RRs must comply with the license conditions and the OLCs (see the definitions in Annex). It must be performed in accordance with the facility safety documentation and approved operating procedures as well.

The following Sections provide descriptions concerning the fundamental principles and approaches in nuclear safety, including in particular the organization of nuclear safety control, the culture for safety (also called safety culture), concept of defence in depth, safety and operational documents, safety of RR utilization and graded approach. Safety considerations for specific types of utilization and activities performed in RRs are also included in relevant Sections of this Compendium.

3.1.2. Fundamental safety objective, principles, and concepts

3.1.2.1.Fundamental safety objective and principles

According to 'Fundamental Safety Principles' published in 2006 by the IAEA [7], "the fundamental safety objective is to protect people — individually and collectively — and the environment from harmful effect of ionizing radiation without unduly limiting the operation of facilities or the conduct of activities that give rise to radiation risks. To ensure that facilities are operated and activities conducted so as to achieve the highest standards of safety that can reasonably be achieved, measures have to be taken to:

a) Control the radiation exposure of people and the release of radioactive material to the environment;
b) Restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation;
c) Mitigate the consequences of such events if they were to occur."

The Fundamental Safety Principles publication [7] is the basis for establishing safety requirements. It addresses nuclear and radiological safety, and it is applicable in its entirety for all facilities and activities, and for all stages over the lifetime of a facility. This includes the associated transport of radioactive material and management of radioactive waste.

Concerning the effective application of these objectives and principles in the case of implementing an educational programme at RRs, the basic principles stated in Fundamental Safety Principles [7] are the following:

(a) Principle 1 Responsibility for safety: "The prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risks". The licensee keeps this responsibility during the lifetime of the facilities and activities and cannot delegate it.
(b) Principle 2 Role of government: "An effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained".
(c) Principle 3 Leadership and management for safety: "Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks".
(d) Principle 5 Optimization of protection: "Protection must be optimized to provide the highest level of safety that can reasonably be achieved". "To determine whether radiation

risks are as low as reasonably achievable, all such risks, whether arising from normal operations or from abnormal or accident conditions, must be assessed (using a graded approach) a priori and periodically reassessed throughout the lifetime of facilities and activities".

3.1.2.2.Safety culture

Safety culture is fostered in the organization to ensure that the attitudes of personnel and the actions and interactions of all individuals and organizations are conducive to safe conduct of activities during operation of the facility. According to the IAEA definition, the culture for safety, commonly referred as the safety culture, is "the assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, protection and safety issues receive the attention warranted by their significance" [4]. Application of safety culture at a reactor corresponds to the highest priority upon safety and all reactor staff, from top management through operating staff, maintenance staff, utilization staff, administrative and supporting staff understanding this primary priority.

Safety culture is relevant to all nuclear facilities and their activities. It should be noted that operation and utilization of RRs should be considered as two activities, each one exhibiting specific but interdependent constraints. These constrains must be considered by the operating and experiment staff (e.g., limited neutron beam time for performing an exercise, strict adherence to the planning for radioisotopes production, and others) and they may be at the origin of conflict between production and safety. For this reason, safety culture is very important and should be promoted for both the operators and users of RRs. A particular attention must be brought to the need for the prevention of human failures, and to the importance of considering human and organizational factors in the management system.

The concept of safety culture should be also explained and promoted in lecturers to the students and trainees during E&T activities using RRs, with a focus on its practical application.

3.1.2.3.Concept of defence in depth

The application of the concept of defence in depth for the design, operation and utilization of RRs provides protection against anticipated operational occurrences and accidents, including those due to equipment failure, inappropriate human actions and events induced by external hazards.

Defence in depth provides levels of defence based on inherent features, equipment and procedures for preventing accidents and ensuring protection of people and the environment against the harmful effects of radiation and mitigating the consequences if an accident does occur. The independence of the different levels of defence is a key element aimed at preventing a failure of a certain defence level n leading to the failure of defence level $n+1$.

The general objectives of the concept of defence in depth are:

(a) To mitigate the consequences of equipment and human failures;
(b) To maintain the efficiency of confinement barriers through prevention of damages in the facility and damages to the barriers themselves;
(c) To protect the public and the environment in case of a failure of the barriers.

Application of the concept of defence in depth for a RR or an experimental device includes consideration at the design stage and implementation of a series of physical barriers, as well as inherent safety features that contribute to the effectiveness of these barriers. For example, experimental devices containing fissile or radioactive materials, or liquid metals such as sodium–potassium or sodium, are equipped with two barriers separating the irradiated materials from the reactor coolant in order to prevent chemical reactions that could damage the core fuel and lead to significant release of fission products to the environment. The space between the two barriers of the device is usually filled with a pressurized inert gas. The surveillance of the leak tightness of each barrier is ensured by the monitoring of the variation of inert gas pressure, which enables for alarms and initiation of safety actions in case of a detection of gas leakage.

3.1.2.4.Deterministic and probabilistic approaches to safety assessment

For most of the RRs, the consideration of safety objectives and safety requirements in the design of RRs and their demonstration are mainly based on the application of a deterministic approach using envelope data for the most conservative core and experiments' configurations. This is consistent with the IAEA Safety Standards Series No. SSR-3 [3], 'Safety of Research Reactors', which considers for the safety analysis a selection of postulated initiating events (PIEs) resulting from equipment failure, system malfunction, human error or internal or extreme hazard.

A probabilistic safety assessment (PSA) could also be used in a complementary manner to the deterministic methods. In this regard, it should be noted that probabilistic studies are useful to identify weak points in the design of the RR facility or to evaluate in a quantitative manner the improvements due to its modifications. Use of PSA methods could also facilitate a better understanding of the systems, structures and components important to safety and their interactions.

3.1.2.5.Safety and operational documents

An important document to be prepared by the operating organization is the safety analysis report (SAR). The purpose of this document is to demonstrate the safety of the RR design and provide the basis for its safe operation. It is requested during the interaction between the operating organization and the regulatory body in the licensing process. The IAEA Safety Standards Series No. SSG-20 'Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report (Rev. 1)' [8], provides detailed guidance on the content and preparation of the SAR. This document comprises the safety analysis which investigates the response of a RR to a range of PIEs. It serves as a basis for the determination of the OLCs, as well as to the safety requirements to be applied in the design of components and systems important to safety.

The SAR also contains additional relevant information related to safety such as site characteristics, building characteristics, safety objectives and engineering design requirements, detailed description of the facility (reactor, coolant systems, instrumentation and control systems, radiation monitoring systems, electric power supply,…), reactor utilization and conduct of operations, radiation protection and environmental impact assessment, commissioning and decommissioning, emergency planning and preparedness, as well as management and quality assurance system.

Visiting a RR, for an insight into its technology, experimental instrumentation and operation is a very valuable experience to learn and understand the basic principles of nuclear reactors and become familiarised with the safety approaches applied in reactor design and operation. As an example, a reactor start–up with the completion of the check-list procedures can be used to show the practical application of the operating procedures and methodology developed for the implementation of a basic safety approach at a RR with the concepts of defence in depth, redundancy, diversity, segregation or physical separation, use of passive systems, quality control as well as use of a conservative approach in reactor technology or operation.

3.1.2.6.Graded approach

This important topic is addressed in IAEA Specific Safety Standards No. SSG-22, 'Use of a Graded Approach in the Safety Requirements for Research Reactors' [9]. For each RR, the design provisions, application of the defence in depth concept and effort and level of details in safety analyses, documentation, operating procedures for applications of OLCs and resources dedicated to safety and its supervision should be commensurate with the potential hazards of the reactor.

Grading should consider the inventory of radioactive material, the means for their dissemination, site characteristics, quality of confinement/containment building and proximity of population.

The objective is to ensure that the operator and the safety organization's efforts are deployed in a commensurate manner with the importance of the safety issues to be addressed.

3.1.2.7.Safety aspects of research reactor utilization

The experiments or irradiations performed at RRs are very diverse, depending on the characteristics of the involved facilities. Experimental devices are generally placed in the reactor core, in its reflector or near its periphery. One or several barriers separate the irradiated samples from the reactor coolant. Irradiations in experimental devices may involve:

(a) Various materials for industrial applications, R&D or E&T;
(b) Targets for radioisotopes production for medical use or for other applications;
(c) Fuel samples contained in experimental loops in which the thermohydraulic condition correspond to incident or accident situations of NPPs. In such an experiment the fuel sample can undergo a cladding failure or melting.

Experiment and modification projects for RRs have several common aspects from the safety management point of view, including organization, safety analysis and management of authorizations and commissioning tests. For this reason, the IAEA published in 2012 the Safety Standards No. SSG-24 [10], 'Safety in the Utilization and Modification of Research Reactors'. This document is applicable to every experiment and education exercises performed at RRs.

According to these safety standards, the RR operator should maintain responsibility for the safety aspects related to experiments and educational exercises, even if their design and their implementation fall with other entities such as research organizations, universities, hospitals and industries.

The safety committee must review the safety of any new experiments and educational exercises, and it may formulate recommendations to the operating organization.

The IAEA Safety Standards No. SSG-24 recommends that:

(a) The regulatory body establishes and implements an authorization process for experiments in the RR, including the possibility of internal authorizations within the operating organization under well-defined criteria;
(b) Procedures should be established for safety analysis and approval of experiments;
(c) Experiment projects should be categorized according to their importance to safety in the frame of a graded approach;
(d) Experiments having major or important safety significance should be designed following the same safety principles as the reactor itself (i.e., application of the defence in depth concept and the single failure criterion) and should be submitted to the regulatory body for review and approval;
(e) Experiments with minor or no impact on reactor safety could be subject to internal authorization within the operating organization.

These safety standards also provide a list of specific safety aspects that should be examined for each experiment such as:

(a) Reactivity worth of the experimental device, which should comply with the OLCs (shutdown margin);
(b) Protection system associated with the experiment to ensure the protection of the reactor;
(c) Heat generated in the experimental device and the adequacy of the cooling system for removal;
(d) Risks associated with a pressurized experimental device, mainly for the items important to safety;
(e) Compatibility among the different materials in the experimental device (risk of corrosion and risk of eutectic formation);
(f) Possible interaction between the experimental device and the reactor (neutron flux perturbation, mechanical interactions);
(g) Updating of safety documents (SAR, OLCs, emergency procedures, etc.).

The above elements highlight the importance of a careful examination from the safety point of view of all possible interactions among the experimental devices and the reactor.

3.1.2.8.Overview of the risks associated with utilization of research reactors

Many RRs located in university campuses or research institutes are utilized for E&T of students, engineers and professionals from the nuclear industry, including operating personnel of RRs and NPPs as well as specialists from regulatory bodies. For training activities in which the trainees can acquire practical experience on the operation of RRs (e.g., approach to criticality, reactor start–up, reactor shutdown, control rods movement), it is important to use a specific core configuration with low excess reactivity[3] to prevent reactivity accidents in case of handling errors.

Research reactors are also tools for basic and applied research. They can produce a wide variety of radioisotopes for medical and industrial applications, as well as bulk doping of silicon by neutron transmutation for the electronics industry. RRs are also used for testing different types of nuclear fuel and materials by irradiation or simulating accidental conditions.

[3] The amount of surplus reactivity over that needed to achieve criticality; it is built into a reactor (by using extra fuel) in order to compensate for fuel burn up and the accumulation of fission-product poisons during operation. The excess reactivity existing in a freshly loaded reactor is balanced by the position of shim and control rods.

Among experimental devices used at RRs, two typical devices are irradiation capsules, which are not instrumented, and irradiation loops, which are instrumented and may be cooled by various fluids (pressurized water, gas, or molten metal). Irradiation loops are used for studies on the behaviour of NPP fuels under normal and accident conditions. Operating parameters of experimental devices such as pressure, temperature and coolant flow are monitored in a continuous manner during irradiation. Safety actions trigger an automatic shutdown of the reactor when safety limits are exceeded.

Safety analysis related to the implementation and utilization of a new experimental devices must address not only the risk relative to the device itself, but any additional risk associated with its possible interactions with reactor operation and utilization, considering the other experimental devices. This addresses the possible impact on reactor safety and PIEs, as adopted for the reactor design.

The safety of irradiation capsules is based on calculations of thermal heating and pressure increase as well as on their chemical and thermal compatibility with irradiated samples. Several incidents, reported in the IAEA Incident Reporting System for RRs, have occurred with irradiation capsules, including loss of barriers, leak tightness or complete rupture of the capsule, resulting in radioactive contamination of a reactor pool and structures, as well as doses to operating personnel.

The main risks associated with these activities include the following:

(a) Melting of the tested fuel, followed by a steam explosion;
(b) Loss of confinement barriers and contamination of the facility;
(c) Inadvertent exposure to radiations of operating personnel and experimenters;
(d) Radioactive release to the environment;
(e) Reactivity accident that could result in acute irradiation of personnel and damage to the reactor core.

Such events have occurred in a number of RR facilities [11], motivating the implementation of provisions to prevent their reoccurrence and mitigate their consequences if they do occur. The most efficient measure for preventing such accidents is the strict application of OLCs and approved procedures for handling operations in the core. Specific safety considerations are presented in the following subsections for particular utilization programmes and activities.

3.2. RADIATION PROTECTION

Radiation protection, which is related to protection of staff, society and environment from radiation generated in the reactor and its utilization, constitutes a part of reactor safety. The IAEA Safety Standards No. NS-G-4.6, 'Radiation Protection and Radioactive Waste Management in the Design and Operation of Research Reactors' [12] provides guidance on radiation protection and radioactive waste management programmes for RR facilities, with recommended good practices in implementing such programmes and their optimization.

Radiation protection is based on three main principles, justification, optimization of protection and dose limits [13]. The justification of use of radiation is a process for determining whether a practice is overall beneficial, i.e., whether the benefits to individuals and to society from introducing or continuing the practice outweigh the harm (including radiation detriment) resulting from the practice.

The limitation of risk to individuals is the process of controlling radiation risks to ensure that no individual bears an unacceptable risk of harm from radiation. The main aim of limitation is to prevent an exposed person from deterministic effects and minimize stochastic effects.

The optimization of radiation protection is a process for determining the highest level of safety that can reasonably be achieved, with doses from radiation exposure and exposure due to any planned radioactive release kept below dose limits and kept as low as reasonably achievable.

Monitoring at a RR consists of facility monitoring and individual monitoring. Facility monitoring focuses on areas in the reactor hall and the surroundings. Various types of stable or portable radiation monitors are typically used for monitoring surface contamination, airborne contamination, liquid contamination, monitoring of solid waste or monitoring of shipments of radioactive material. Individual monitoring, also known as personal monitoring, is based on the measurement of external doses at a RR where usually neutron, β (beta) and γ (gamma) radiation levels are measured. Various electronic dosimeters, film and thermoluminescent dosimeters (TLDs), nuclear track detectors, activation detectors and bubble detectors are used as neutron detectors for individual monitoring.

Research reactors are often equipped with experimental and irradiation facilities that use neutrons or other radiation types, such as γ, emitted from the core. Burned fuel as well as irradiated and activated materials are also produced and manipulated at RRs. These facilities and activities may pose a significant radiation hazard to personnel. Thus, special features should be enforced to provide radiation monitoring and protection. This applies specifically to neutron beam tubes and thermal columns, even for reactors of low power levels. Irradiation loops or rigs can present a significant radiological hazard owing to the increased risk of a release of radioactive material caused by high pressures and temperatures. Issues related to possible melting of fissile materials that are usually present in loops or rigs should be considered in the planning of the radiation protection programme at the facility.

Radiological zones are defined on the basis of radiation sources in different areas within the facility. Clear demarcation should be in place between different radiological zones.

Concrete blocks and lead shielding are very often installed around experimental zones equipped with neutron guides in order to allow safe access to these zones. The experimenters and operators are provided with dosimeters relevant to the type of radiation in each experimental zone. Audio and visual alarms are triggered in cases when a preset dose or dose rate is reached in the working area.

For further information on radiation protection, please consult the bibliography.

3.3. RADIOACTIVE WASTE MANAGEMENT

Radioactive releases from RRs could be in gaseous, liquid and solid form. Each of these types of releases ought to be controlled and monitored to keep the amounts and concentrations of radioactive releases as low as reasonably achievable and below authorized limits on discharges. For all types of releases, consideration should be given to minimizing both the radioactivity content and generated volume. Monitoring of releases should be ensured to check compliance with specified and approved criteria as well as with corresponding regulatory requirements. Design of experiments should ensure that radioactive waste generated as a result of RR utilization is restricted to a minimum practicable level. Documentation indicating in detail the nature of radioactive waste, location of the waste, as well as safety and security measures needs to be established and maintained.

3.4. EDUCATIONAL ASPECTS

Nuclear safety and provisions to be implemented for ensuring safe operation and utilization of a RR must be the primary objective of the operating organization. Disseminating and applying the concepts of nuclear safety, safety culture and good practices should also be one of the main learning objectives of any educational exercise to be performed at a RR.

Either RR exercises specifically dedicated to nuclear safety should be developed, or RR exercises should address the safety issues related to a given aspect of RR operation and utilization. Indeed, any exercise is an opportunity to practice operational safety and develop an interrogative attitude regarding the activities performed at the RR.

In both cases, exercises should provide a clear understanding of the basic principles and concepts of safety, as well as methodology used for their practical implementation at a RR. For example, learning objectives can include the understanding of:

(a) The need to consider and apply the safety principles and concepts at all stages of a facility (design, operation, modification …) and in all activities (operation, utilization);
(b) The need for a constant application of an interrogative attitude during reactor operation and utilization as a key component of the safety culture;
(c) The safety considerations of RR design, the definition of the OLCs and the establishment of the operating procedures applicable to the RR;
(d) The design and functioning principles of the I&C, including the instrumentation and protective systems.

Specific exercises can be dedicated to radiation protection in order to provide students with a clear understanding of issues and protective measures to be taken in this domain. Exercises performed at a reactor are an opportunity to measure radiation doses at a nuclear facility and practice the concepts related to radiation protection.

Exercises, or specific considerations as a part of an exercise, on nuclear safety are, to a certain extent, mandatory for students who are studying curricula related to reactor design, physics, operation or safety analysis. This applies to students at all three degree levels — bachelor's, master's and doctoral. The content of this education activity can be at the basic, medium or advanced level, depending on the level of knowledge of the students and the learning objectives.

These exercises related to nuclear safety are also essential for students studying various engineering majors in master's and doctoral study programmes, including but not limited to power engineering, mechanical engineering or electrical engineering, with future assignments in conjunction with a minor curriculum in nuclear engineering. The level of the exercises in this case is usually basic or medium, keeping in mind that the dissemination of the concept of nuclear safety, safety culture and good practices is a key objective in nuclear capacity building.

For the basic level, the objective is to introduce the concepts of nuclear safety and raise the awareness of importance to the students, providing practical examples of the I&C architecture, the reactor protection system and the associated protective actions. Practical procedures, online measurements, and protection systems ensuring the safe operation and utilization of the reactor can be emphasized. Also, the concept of defence in depth can be briefly explained, and general rules related to radiation protection presented. It is important to show that reactor operation and utilization are practiced with the application of strict safety rules to minimize risk. Reactor exercises are the place for lecturers and reactor operating personnel to share knowledge and experience in nuclear safety and disseminate safety culture. Highlighting the qualification and professionalism of the reactor operating staff is very important. This introduction to nuclear safety at a RR can easily be included into the presentation of the facility as well as in basic demonstration and study of reactor operation and utilization. The duration of this specific introduction to nuclear safety can typically range from ½ to 1 hour.

For the intermediate level, the practical application of nuclear safety can be addressed and studied. A detailed presentation of safety concepts and principles, I&C system(s), reactor protection system, radiation protection rules, radiation monitoring system and environmental monitoring is an effective tool. In addition, specific exercises can be dedicated to the study of the OLCs, measurement of the safety margins, I&C system and its function, measurement of radiation doses at the reactor and application of radiation protection measures. Some specific case studies can be conducted, and feedback on incidents and accidents that have occurred at nuclear facilities can be discussed. The main objective is to provide the students with practical experience in the analysis and application of nuclear safety. The duration of an exercise can typically be 1–3 hours. Another and complementary approach is to take the opportunity during an exercise at the RR to illustrate and study the safety concepts relative to a given aspect of reactor operation and utilization.

For the advanced level, exercises can include a detailed study of the safety issues related to specific aspects of reactor operation and utilization. This can include the analysis and explanation of existing operating procedures. It can also take the form of an exercise in which the students must conduct their own safety analysis and propose adequate OLCs or operating procedures that will be compared with the actual ones at the facility. The duration of such exercises can typically be 3–6 hours. Additional projects can be extended to the use of calculations, in relation, for example, to core characteristics, thermal-hydraulic parameters or radiation doses, as a base for the safety analysis and establishment of OLCs or operating procedures at the RR. Such an exercise can be organised as a week-long project for the students.

Besides RR exercises integrated in higher education programmes, common activities at RRs are presentations and tours at the facility. Such visits can be organized for students, including high school students, professionals involved in activities linked to nuclear applications, decision makers, journalists and public.

3.5. EXERCISES ON SAFETY ASPECTS OF REACTOR OPERATION

3.5.1. Objective of the exercises

The objective of the exercises is to provide insight into practical aspects of safe operation and utilization of a RR. Exercises are an opportunity to disseminate safety culture and good practices, as well as initiate the development of adequate behavioural competencies regarding the safe operation and utilization of nuclear facilities. Provided hereafter are useful information to develop exercises dedicated to study about safety principles, rules and procedures, as well as radiation protection and environmental monitoring.

3.5.2. Equipment and conditions

Exercises addressing safety aspects are usually conducted at normal operating conditions of the RR.

For the study of safety principles, rules and procedures, no specific equipment is needed. For the medium and advanced levels, the facility's safety documentation could be made available to the students in the control room.

For training related to radiation protection and environmental monitoring, the measuring systems implemented at the facility can be used. They can be complemented if needed by additional measuring devices such as portable detectors and radiation dose monitors to measure radiation and dose rates in different locations within and outside of the facility.

All rules and procedures to enter, to perform the training and to exit facility ought to be strictly followed. It is important for educational purposes as well as to disseminate good practices to show strict adherence to safety and radiation protection rules.

For educational proposes, even if there is no risk of contamination at a facility, it can be decided as a training exercise on the application of radiation protection rules to apply a mock procedure entailing the protocol, including, e.g., wearing special individual equipment such as coats and gloves, surveying for contamination and making input to a logbook.

3.5.3. Methodology

Since topics related to safe reactor operation and utilization are very broad, this Section gives only two key examples of training activities that can be conducted.

3.5.3.1.Safety principles, rules, and procedures

After a general introduction to safety principles, a description of the reactor's I&C system can illustrate the practical application of these principles. The I&C system provide monitoring, control, supervision and protection at a RR. It should ensure three main safety functions: control of reactivity, control of heat removal from the core and from the fuel storage, control of confinement of radioactive materials. The control panels of the I&C system are often designed so that a specific part of one control panel is dedicated to each of the three main safety functions.

Learning about reactor safety can be conducted as follows:

(a) Introductory presentation about, for example, the safety principles ;
(b) Description of the detectors and associated measurements ensuring the control of the reactor;
(c) Presentation of the I&C system, the control desk and its information panel(s) that displays reactor parameters and associated alarms;
(d) Provide a practical example on how reactor operation is controlled; for a given state of a reactor, the students can be asked to fulfil a data sheet (similar to the information recorded by the operators in the logbooks) with the reactor parameters such as: power, position of the control rods, counting rate or current in neutron detectors, doubling time, water temperature (inlet and outlet), flow rate of the primary circuit, negative pressure in the reactor hall, β and γ dose rates in different locations and at the air exhaust;
(e) Recorded parameters can be checked according to the OLCs;

(f) Safety reasons related to definition of the OLCs can be presented to, and then discussed with the students;
(g) Action(s) taken by the protection system when a reactor parameter is outside its OLCs can be presented to, and then discussed with the students.

For example, the use of a doubling time for safe operation of a reactor can be explained accordingly:

(1) The doubling time T_d is a key parameter to control reactivity; it is measured at least by two different detectors for redundancy.
(2) T_d is deduced from the evolution of the counting rate or the current given by the low level of power or the high level of power neutron detection system, respectively.
(3) T_d is the main parameter to be followed by the operators when the reactor power is increased. Thus, the T_d given by each operating detection system is shown on the control desk (analogic or digital display). Two thresholds are usually defined in the OLCs, a first level corresponding to an alarm (for example for $T_d = 10$ s) and a second level that induces the reactor SCRAM [4] ($T_d = 5$ s).
(4) The T_d can be observed during reactor operation checking that it is compatible with its expected value and that it is far from the limits previously defined. The T_d measured by two detection systems, in application of the redundancy, can be compared.
(5) The limits defined for T_d result from the need in limiting the speed at which the neutron density (and power) increases. Standard T_d values are above 20–60 seconds depending on the reactor. Thus, the first limit corresponds to the detection of an abnormally fast increase of power. The second limit corresponds to a safe shutdown of the reactor resulting from an unsafe increase of its power. Indeed, it is important to keep a significant margin ensuring that reactor is shut down before reaching criticality accident conditions.
(6) Actions taken by the protection system can be different for the first and second level. The first one may result in an audible and visual warning that can be associated with the inhibition of the rod extraction. The second one results in the drop of all the control rods in order to ensure the safe shutdown of a reactor.

For the advanced level, with the objective to give students some experience in operational safety, students can study the general operating rules of the reactor and find by themselves the standard operational conditions, limits of the OLCs and actions taken by the protection system.

3.5.3.2.Radiation protection at the reactor

Radiation protection at the reactor is related to the protection of the reactor staff, public and environment from radiation generated in the reactor and its applications. The exercises should focus on understanding of the basic principles of radiation protection, its justification and on the limitation and optimization of radiation doses, which can be achieved in practice using radioactive decay time, distance and shielding. Exercises can include the monitoring of dose rates (β, γ, neutrons) and control of contamination at different locations of the facility.

[4] The term *SCRAM rods* is widely used also to name the safety control rods. This term refers to the year 1942 and to the first reactor, the Chicago Pile-1. During the reaching of the first criticality, the safety control rod was called SCRAM. The acronym stands for ***S**afety **C**ontrol-**R**od **A**xe-**M**an*. On that design, the rod hung from a rope, and one member of the reactor staff was ready to cut the rope by the means of an axe to allow a fast rod drop into the reactor and therefore stop a chain reaction.

From a practical point of view, focusing on radiation protection measurements around the core and at reactor equipment is easier.

The study of radiation protection can be conducted as follows:

(a) A presentation about radiation protection principles such as justification, limitation and optimization of radiation doses, as defined in Section 3.2.
(b) Identification of the risk related to radiation exposure at different locations in the facility. For example, the following origins of risk can be addressed: reactor core, water or concrete shielding breach during reactor operating; the fuel during handling and in a storage facility; activated and potentially contaminated core components; activated and contaminated water from the primary circuit, beam ports and ancillary facilities; radiation sources, neutron start–up source, for example; activated devices and samples; radioactive releases in gaseous, liquid and solid form. Using a simplified schematic of the reactor facility, the students can be asked to identify the components and circuits of the reactor concerned with potential radiation exposure and contamination. They can also be asked to define the types of measurements that should be performed to monitor the risk at different locations. This can include, for example: the measurement of neutron or gamma dose rates, sampling to control the activity of water from the pool or from the primary circuit, surface contamination control, as well as control of activated dust on filters placed on the air circuit.
(c) Description of the different types of detection systems used for radiation measurements in the reactor hall, at reactor equipment (beam port, etc.) and in the surrounding outside area. Fixed or portable radiation monitors are used for dose rate and contamination measurements around the core, in the reactor hall, around water and air circuits, as well as to control irradiated samples and wastes. Personal monitoring is based on measurement of the external doses (usually neutron and beta–gamma radiations). For workers this is usually done by two complementary means: by electronic dosimeters, for the continuous monitoring, and by film or TLDs, for monthly or quarterly integrated doses. Students can be asked to indicate which type of detection system should be used for radiation protection according to the potential risks previously identified.
(d) Measurement of the radiation dose and contamination in different locations and its use for the protection of reactor staff, public and environment.
(e) Explanation of the OLCs which generally arise from the national regulation related to radiation protection of the workers (and public), as well as the authorization given by the regulatory body. The latter concerns for example the annual limit of radioactive gaseous released into the environment.
(f) Explanation of the action(s) taken by the protection system of the reactor or the radiation protection officer when a parameter is beyond the OLCs or normal range. This can concern, for example, an upper limit of the beta–gamma dose rate that can induce an alarm and an automatic shutdown of the reactor.

Table 3 gives examples of radiation protection measurements that can be fulfilled by the students during the exercises. The table, which is more applicable to medium and advanced level students, can be simplified for the basic level students.

3.5.4. Safety considerations

Exercises on nuclear safety must be performed in accordance with the safety documentation and operating procedures of the reactor. Concerted care should be taken to strictly apply the procedures since these exercises are an opportunity to disseminate and apply the concepts of nuclear safety, safety culture and good practices.

All exercises must be performed in accordance with the safety documentation and operating procedures of the reactor. Reactor start-up and changes to reactor power should be performed using standard operating procedures.

For further information on nuclear safety, please consult the bibliography.

TABLE 3. EXAMPLE OF RADIATION PROTECTION MEASUREMENTS AT A RR

Area of interest	Type of measurement	Type of detection system	Measurement to be completed by students	Normal range	Action when out of range
Reactor hall – position 1	β–γ dose rate	Fixed radiation monitor			Alarm (SCRAM)
Reactor hall – position 2	β–γ dose rate	Portable radiation monitor			Alarm (SCRAM)
…					
Beam port	Neutron dose rate	Fixed radiation monitor Completed by measurement with a portable radiation monitor			Alarm Beam cut off
Water – primary circuit	Water activity	Water sample activity measurement by Scintillation or evaporation method			Incident report
Water – pool	Water activity	Water sample activity measurement by Scintillation or evaporation method			Incident report
Air circuit	Cartridge filter's activity	α–β–γ detection on the filter			Incident report
Air circuit	Air activity	Comparative system with ionization chambers			Alarm (SCRAM)
…					
Personal dosimetry	β–γ– neutron dose and dose rate	Electronic and film dosimeters			Alarm Incident report

Before entering the facility, the basic rules, instructions on actions to be taken in case of an incident and a meeting point in case of emergency evacuation of the facility, should be discussed with the visitors. It should be stated if the use of mobile telephones or taking photos is prohibited. A logbook should be used to record the entrance and exit of all visitors and students at the facility. The students may practice fulfilling a table with their name, surname, organization, check-in time, check-out time, personal dose recorded during the exercises and signature. In the case of an extended exercise with a duration of more than half a day, it is advised that the students check in and out for each half day period.

From a radiation protection point of view, the participants must comply with the basic rules of the facility when entering the facility, reactor hall and control room. Dosimetry, lab coat, overshoes and gloves are to be considered depending on the risk. Additional risk may be considered when performing measurements of radiation doses at different locations in the facility such as neutron beam ports or in the vicinity of the primary circuit (water activation). Definitely, the measurement of dose rates, activity or contamination around the reactor should not result in a significant dose or contamination risk for the students, as the ALARA principle (see Section 3.2) must apply. When exiting the facility, the personal or group dosimetry must be analysed to verify the absence of significant radiation exposure and control of contamination should be carefully performed. In some facilities, control of contamination is also performed when the students enter the facility, as this is used as a reference measurement before the students and trainers perform reactor exercises. Indeed, this would enable identification of contamination of a participant prior to his entry to the facility cannot be totally excluded.

Concerning the entry to the facility or reactor operation in the presence of visitors, some particular conditions may apply according to national regulations. For example, restrictions related to reactor operation may apply if the students are minors or if visitors have specific medical conditions.

From a security or safeguard points of view, some specific rules and restrictions may apply. Access may be restricted in areas where fresh fuel is stored, for example.

The safety considerations made in this Section apply to all exercises in this Compendium.

3.5.5. Documentation

In order to perform exercises related to nuclear safety, the following documents can be given to the students:

(a) Background: reactor principle and operation, safety, radiation protection;
(b) Schematic of the facility, which may include the points of interest and the localization of the reactor components and the monitoring systems;
(c) Description of the I&C system, list of the OLCs and actions taken by the protection system;
(d) Any, safety document and procedure that is needed to illustrate operational safety or to complete the exercise;
(e) Specific information and rules, such as the operating mode of the radiation measuring systems, specific rules for the access to the vicinity of a beam port, or interdictions to access some areas when the reactor is in operation (for example technical room where the primary circuit is located);
(f) Step by step procedure to complete the task, including tables to be completed by the students (see Table 3 for example).

3.5.6. Questions to the students

As an example, the following set of questions can be addressed to the students:

(1) List the three main safety functions for a RR.
(2) For each safety function, give at least two reactor parameters that are continuously measured to ensure safe operation. Give the OLCs related to each of these parameters and explain the way they have been established according to safety considerations.
(3) Give at least three examples of abnormal situations (a parameter exceeding the OLCs), that should result in reactor shutdown by SCRAM or manual shutdown.
(4) What is redundancy, and why do we apply this concept for parameter measurement in a reactor?
(5) Give at least five parameters that are commonly measured in a reactor for radiation protection considerations.
(6) Describe the detection system used to measure each of these parameters.
(7) Give the maximum value of the daily dose authorized for a radiation worker, i.e., under specific medical surveillance, and if authorized by national regulations, for a public visitor.

Further evaluation of the impact of the experiment can be obtained through a deliverable such as a report or a presentation on the objectives, methodology and results obtained from these measurements, prepared by the students and delivered to the teacher.

4. REACTOR INSTRUMENTATION AND CONTROL SYSTEM

4.1. BACKGROUND

Research reactors are equipped with an adequate I&C system to ensure reactor safety in normal operation, which includes start-up; operation at any power; shutdown; refuelling and maintenance; as well as in incidental and accidental conditions. The architecture of the I&C system has to anticipate all foreseeable operational occurrences and post-event conditions. The I&C system provides protective actions such as automatic reactor shutdown, emergency core cooling, residual heat removal and confinement of radioactive material. The quality and reliability of the I&C system and reactor equipment have to be commensurate with their safety classification.

Safe operation of a RR requires appropriate and reliable reactor instrumentation which is a key factor in safe operation. Reactor instrumentation ensures the achievement of proper operating conditions, the prevention of accidents and the mitigation of accident consequences, resulting in protection of workers, the public and the environment from undue radiation hazards.

The I&C system and its instrumentation have to comply with the concept of defence in depth that implies redundancy, diversity and segregation of items important to reactor safety (see 3.1.2). Such provisions are aimed at preventing the risks of common mode failures.

4.2. THEORY

4.2.1. Instrumentation and control systems of research reactors

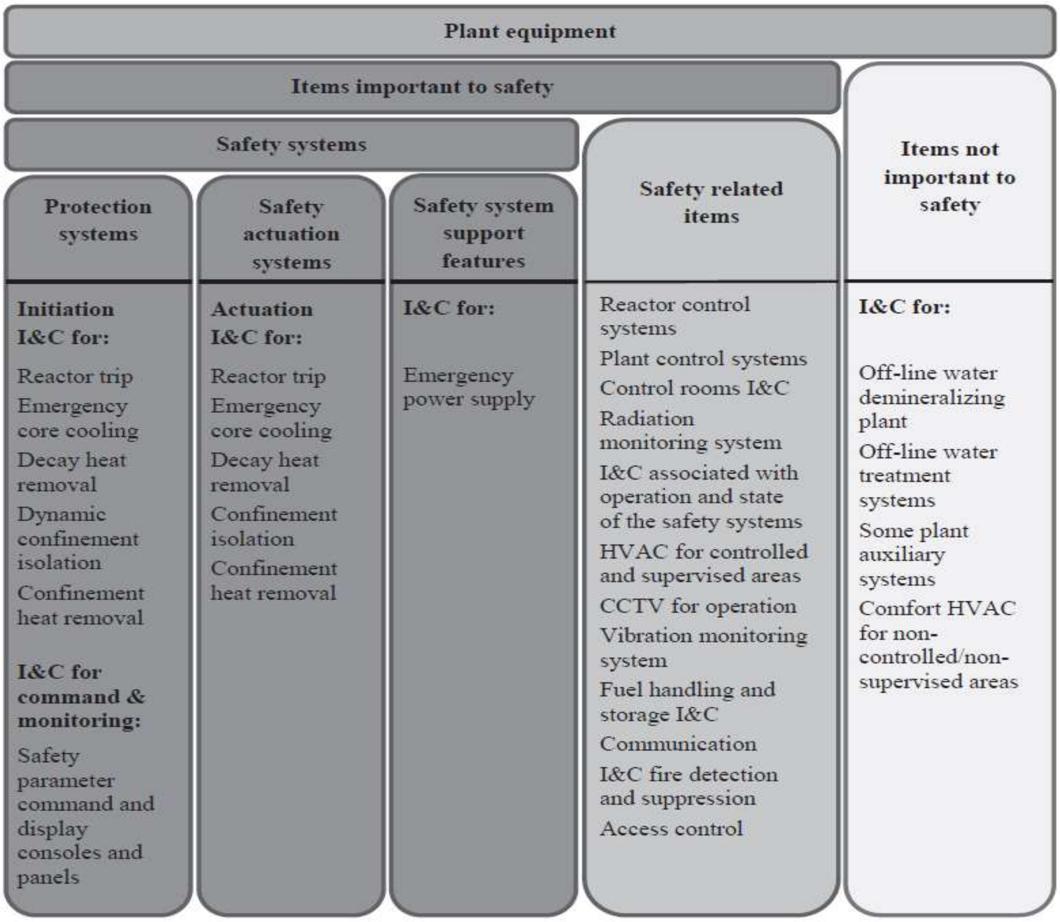
Reactor equipment consists of items that are important to safety and some that are not important to safety. Items or systems that are important to safety are divided into safety systems and safety related items or systems. Some examples of safety systems are protection systems, safety actuation systems and safety system support features. Reactor instrumentation is integrated in the I&C system to ensure the monitoring, control, supervision and protection of the RR. A number of vital functions are performed by the I&C system to ensure safe and efficient RR operation.

The I&C system fulfils three main safety functions: the control of reactivity, control of heat removal from the core and confinement of radioactive materials.

The first safety function consists, for example, in controlling reactivity within safe limits, preventing unacceptable reactivity transients and shutting down the RR to prevent anticipated operational occurrences or mitigate the consequences of accident conditions.

The second safety function involves removing heat from the core during operations, by means of sufficient coolant inventory for core cooling; after reactor shutdown through adequate removal of residual heat; and in accidental conditions, for example following a loss of coolant. The third safety function consists in the confinement of radioactive materials and is achieved by maintaining the integrity of the fuel cladding, maintaining the integrity of the boundary of the cooling system, and limiting the release of radioactive materials to minimize the exposure of the public and personnel to radiation.

There are four primary I&C system functions important to safety: protective, control, monitoring and display, and testing. Protective functions of an I&C system provide a line of defence against failures in other reactor systems. Control functions provide assurance that the reactor is controlled and kept within its operating envelope under normal and abnormal conditions. Monitoring and display functions provide the interface between the reactor and the operation and maintenance personnel. Testing functions provide assurance of the availability and effectiveness of other functions important to safety and confirm that these have not been degraded. Fig 1 represents the typical structure of the I&C system of a reactor.



Note: CCTV — closed circuit television; HVAC — heating, ventilation and air-conditioning; I&C — instrumentation and control.

FIG. 1. Examples of instrumentation and control systems of a RR classified according to their importance to safety, as presented in IAEA Specific Safety Guide No. SSG-37 [13].

The overall I&C system is usually divided into three types of subsystems: protection systems, control systems, and information systems. At high power RRs, like at NPPs, three other types of subsystems are defined: limitation systems, interlock systems and risk reduction systems.

The protection system is a particularly important component of an I&C system and is designed to perform the following functions:

(a) Automatic initiation of appropriate actions performed by other systems, including as necessary the reactor shutdown systems, in order to ensure that specified design limits are not exceeded as a result of anticipated operational occurrences;
(b) Detection of design basis accidents and initiation of actions performed by other systems necessary to limit the consequences of such accidents within the design basis;

(c) Overriding unsafe actions of the control system.

The control systems encompass all equipment and components used automatically and manually to control facility parameters, from the connection to the process sensors to the actuation devices that have a direct impact on the physical processes affecting the values of the parameters to be controlled.
Information systems encompass equipment and components such as sensors, equipment to convert signals from the sensors to those suitable for display or recording, sound transmitters, lights, visual display units, recorders, printers and solid state display devices.

Instrumentation of a RR includes both non-nuclear and nuclear instrumentations. Non-nuclear instrumentation measures conventional technological parameters such as temperature of moderator and coolant, flow rate through the reactor pool, tank and water management system, water level in the reactor pool, pressure in the different parts of the facility (such as reactor hall or hot cells) and various parameters measurements related to experiments and exercises.

4.2.2. Nuclear instrumentation

Nuclear instrumentation is dedicated to radiation detection with a special focus on neutron and gamma detection. Neutron detectors are a crucial element for reactor operation, as they provide information about power and its rate of change. In addition to these key measurements for the I&C, additional neutron detectors may be used for neutron field characterization, providing information about neutron flux distribution and energy spectrum.

Fast and frequent changes in power are common at RRs. Thus, I&C neutron detectors provide quasi-instantaneous information about the reactor's state. Also, the operational power range of RRs is often wider than that of nuclear power reactors. There are several different types of neutron detectors, classified into two main groups: active and passive neutron detectors.

Active neutron detectors react immediately to reactor power changes, and their output signal is proportional to these changes. Typical examples of active detectors used at reactors are gas-filled detectors mainly based on the use of a boron or uranium converter. In such detectors, neutron interactions with the converter produce ionizing particles that are detected in the gas. These detectors are used for neutron density and reactor power measurements. For radiation protection or cladding failure measurements, gas detectors with a helium converter are commonly used.

Passive neutron detectors exhibit a response proportional to the neutron flux integral, but their evaluation is delayed. Passive detectors cannot be used for reactor control and instantaneous power measurement, but they are still valuable for long term measurements. Typical examples of passive neutron detectors are activation detectors based on the activation of specific foils or wires, such as Au or Mn, which are used to measure neutron flux magnitude and spectrum. Thermoluminescent detectors are also used for radiation dose monitoring of personnel and the environment.

As explained above, a specific feature of gas-filled neutron detectors is that neutrons cannot directly ionize the gas filling of the detector, i.e., they are called indirectly ionizing particles. Thus it is necessary to convert neutrons to other particles capable of generating a measurable charge in the detector volume. This conversion can either be realized by the gas filling ($^{10}BF_3$ or ^{3}He) inside the detector or the detector walls being lined by a suitable material (^{10}B or ^{235}U). These detectors are either ionization chambers or proportional counters [15]. The probability of a neutron interaction with the converter greatly depends on neutron energy. Detection of neutrons with kinetic energy below 1 eV — thermal neutrons is especially important for RR operation.

Several factors influence the successful conversion of thermal neutrons into ionizing particles. The reaction cross-section of the converter should be high so its size may be reduced. This is especially important for gas-filled neutron detectors, in which the conversion is directly achieved in the gas. Detection of thermal neutrons relies on the production of heavy charged particles. If a thermal neutron interacts with a target nucleus, it typically results in the production of proton or alpha particles or fission fragments. All of these reactions are sufficiently exothermic, and the energy of the reaction, which is typically in the range of 0.1–100 MeV, greatly surpasses the incoming energy of the thermal neutron (below 1 eV). Also, should be taken into account that neutrons frequently coexist with gamma radiation, which can also produce an output signal. Using a converter, such as ^{235}U, that produces high energy ionizing particles will result in a better capability to discriminate the output signal arising from neutrons from the one arising from gammas.

For reactor control two (or more) types of detection systems are generally implemented. A first system working in the pulse mode is used at low power (LP) levels. This detection system is capable of measuring each neutron interacting in the detector, giving rise to a counting rate, i.e., a number of electrical pulses per second proportional to neutron density. It is used from the source level, when neutrons are mainly supplied by the source before reactor start-up, up to a maximum value of the power, which is determined by the dynamic range of this system, typically five decades. A second system working in the current mode is used at high power (HP) levels. This detection system is measuring a current that is proportional to neutron density. It is used from the minimum power level, which is determined by the dynamic range of the system) to the Nominal Power (NP) plus a certain margin: NP + X% (typically 10–20% above the nominal power). These detectors usually cover up to seven decades of power evolution.

The LP level detection system is usually equipped with ionization fission chambers or boron proportional counters. The HP level detection system is usually equipped with ionization fission chambers or ionization boron chambers.

The information given by both LP and HP level detection systems is used to measure and control a certain number of parameters and safety indicators such as the counting rate or current proportional to neutron density, rate at which this neutron density changes (period or doubling time), reactor power and proper variation of these parameters within the OLCs.

Since these neutron detection systems are the only tool capable of giving instantaneously an image of the neutron density and its variation, these are key systems for the control of the chain reaction.

4.3. EDUCATIONAL ASPECTS

I&C systems are key components ensuring safe operation and safe utilization of a RR. Corresponding exercises should provide an understanding of the basic principles of reactor I&C and also offer practical applications of the safety concepts in the design, functioning and protective action of the system.

Research reactor exercises specifically dedicated to the I&C system can be developed. Alternatively, other exercises (for example a temperature effect experiment, see Section 9.4.3) can be used to analyse the design, functioning and protective actions of the I&C system's related functions. Any exercise should be seen as an opportunity to practice operational safety and develop an interrogative attitude regarding the activities performed at a RR.

The learning objectives related to I&C can include the understanding of:

(a) Design and functioning principles of the I&C from the conceptual and practical points of view;
(b) Key role of I&C in safe RR operation and utilization, ensuring operation within the conditions set by the OLCs;
(c) Methodology applied in the establishment of the actions taken by the I&C protective system;
(d) Importance of applying the fail–safe criteria, redundancy and diversity in relation to the performance requirements in safety related I&C systems;
(e) Importance of applying the concept of defence in depth in the design of the I&C;
(f) Functioning principle of a specific part of the I&C system and its associated instrumentation, such as the neutron instrumentation;
(g) Need for a constant application of an interrogative attitude in reactor operation and utilization when considering the information given by the I&C.

These objectives are highly suitable for students choosing nuclear science as the major curriculum in all three study programmes — bachelor's, master's and doctoral; the level of the content from basic to advanced (see Section 2.3.2) being adjusted to the level of knowledge of the students and the learning objectives. This knowledge is also suitable for students in various major engineering curricula in master's and doctoral study programmes such as power engineering, mechanical engineering, electrical engineering, the level of the content being usually basic or medium.

With an objective of observing and analysing the data measured by the detection system of the I&C, no specific experimental instrumentation is needed for the above mentioned activity related to reactor instrumentation. A RR with its standard technology and experimental instrumentation, including appropriate neutron and gamma detectors, is sufficient. If the objective is to observe the electrical signal at different stages of a neutron detection system and study the settings of the neutron detection systems, additional detectors and detection systems will have to be implemented independently of the I&C system, in order to ensure that there is no interference with the I&C system and the reactor safe operation.

For the basic level, the objective is to introduce the concept and architecture of the I&C system, providing practical examples of the role of the I&C in ensuring the safe operation and utilization of the reactor. It can either take the form of a brief description of the I&C system integrated into a presentation of the facility or a more detailed description in the frame of reactor exercises. In the latter case, application of the concept of defence in depth can be explained and illustrated through analysis of the I&C characteristics, emphasising the role of the protection system and its protective actions. The functioning of a specific type of instrumentation and the corresponding OLC's can also be further addressed. Depending on the level of detail that is addressed, the duration of the introduction and exercises can typically range from ½–2 hours.

For the intermediate level, deeper insight into the concept, architecture and practical aspects of the I&C system can be addressed. A reactor exercise can also focus on a specific type of instrumentation and signal processing such as the neutron instrumentation, which plays a key role for safe operation. In such a case, exercises can provide a basic understanding of the physical processes involved in the elaboration and processing of the electric signals given by the neutron detection systems both in the pulse and currents modes. The duration of such an exercise is typically 1–2 hours.

When focusing on neutron instrumentation for the advanced level, additional objectives can include the characterization of the signals from the neutron detection systems, such as the duration of the pulses in the counting mode, establishment of the operating range of each system with their associated power level, as well as establishment of the thresholds that can be set on each detection system in order to ensure the reactor protection. Duration of such an exercise is typically 2–3 hours.

4.4. EXERCISES ON NEUTRON INSTRUMENTATION

4.4.1. Objective of the exercise

Neutron detection systems are essential for reactor operation since they provide immediate information about reactor power and rate of power change, ensuring safe reactor operation and control. The objective of this exercise is to study the neutron detection systems, addressing successively the study of the two main operating modes, i.e., the pulsed and current mode; analysis of the neutron instrumentation signals during reactor operation at different power levels; and understanding the use of these signals by the protection system, ensuring the control of reactivity and heat removal.

4.4.2. Equipment and conditions

To perform a study of the neutron detection systems, two approaches are possible. In the first approach, the existent instrumentation of the I&C system can be used to provide the data to be analysed. In this case, it is usually not possible to modify the settings of the neutron detection systems for the purpose of the exercise.
Alternatively, additional detectors can be installed in the vicinity of the core and connected to a data processing and acquisition system. This latter approach gives the possibility to modify the settings of the neutron detection systems and study in more detail the operation of such a system. In this case, the following equipment needs to be installed:

(a) Neutron detectors for the LP level, such as fission chambers or boron proportional counters, and the HP level, such as fission chambers or ionization boron chambers;
(b) Cables, power supply, preamplifier and discriminator for the LP detection system;
(c) Oscilloscope to observe the signals at the different stages of the LP detection system;
(d) Cables, power supply and ampere meter (typical range: nA to mA) for the HP detection system.

The neutron detectors should be located in the vicinity of the core, ensuring that the counting rates and current given by the additional detection system allow proper measurement of the neutron density from the source level to the nominal power. As for the detection systems from the I&C system, there should be appropriate overlap, typically at least two orders of magnitude, between the information given by the LP and HP detection systems.

4.4.3. Methodology

As indicated previously, the exercise can be performed in three stages. For each stage, this Section provides first a methodology applicable when using the I&C instrumentation. Complementary indications are then provided as additional instrumentation is available.

The first stage of the exercise concerns the explanation of the principle of the neutron detection systems and, whenever possible, the observation of the electrical signal and its fluctuation.

Schematics such as those given in Fig. 2 can be used to describe the functioning of the LP and HP neutron detection systems working respectively in the pulse and current mode. A schematic of the core and its vicinity showing position of neutron detectors should also be available. The characteristics of the detectors (such as type, size, operating and nominal voltage, and sensitivity) and its electronics (gain of the amplifier, typical voltage supply and threshold values) should also be provided.

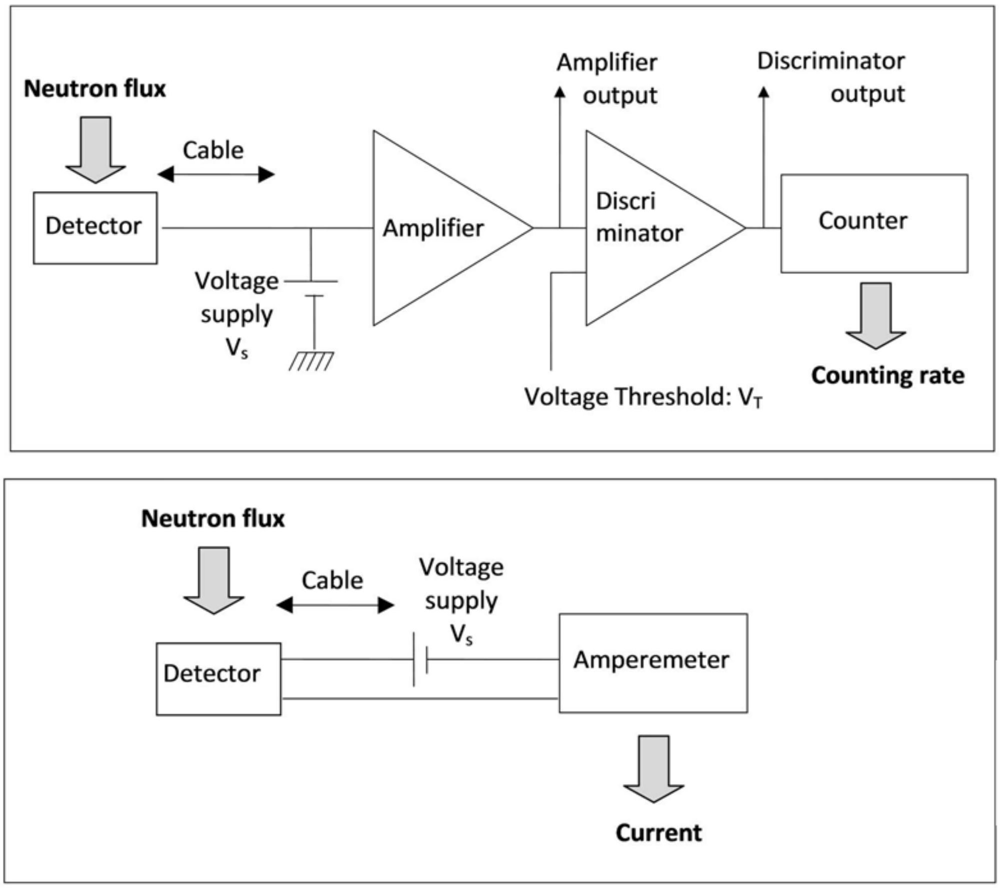


FIG. 2. Schematic of the LP and HP detection systems, respectively, operating in the pulse and current mode, and providing a counting rate and a current proportional to the neutron density and reactor power. (Courtesy of the National Institute for Nuclear Science and Technology, CEA Saclay, France)

For this first exercise, the LP and HP neutron detection systems should be set at their nominal operational values. While explaining the principle and operation of the neutron detection systems the following issues can be addressed:

(a) Position of the detector, which influences its operating range in relation with reactor power;
(b) Neutron energy spectrum and the environment of the detector, which influences the response of the detection system;
(c) Values chosen for the voltage supply for the LP and HP detection systems, which mainly depends on the operating range provided by the detector supplier;
(d) Voltage threshold of the discriminator of the LP system, which highly depends on the operating conditions at the reactor;
(e) Observation of the LP detection signal at the output of the amplifier, which can include the measurement of the typical height of the noise and neutron pulses as well as of the full width at half maximum (FWHM) of the pulses;
(f) Role of the discriminator, which is to suppress the contribution of noise and lower amplitude signals arising from gamma radiation, and the proper setting of the discriminator voltage threshold;
(g) Observation of the signal at the output of the discriminator, which is usually a transistor–transistor logic (TTL) shaped signal, whose amplitude and FWHM can be measured;
(h) Observation of the fluctuations of the counting rate at different power levels to exhibit the statistical character of neutron counting and its influence on the uncertainty of the measured counting rate;
(i) Determination of the maximum counting rate (τ_{max}) of the LP system, which can be deduced from the FWHM of the pulses at the output of the discriminator, which is given by $\tau_{max} = 1/10 \times \text{FWHM}$ for a random signal [17];
(j) Measurement of the dark current of the HP detection system at zero power and observation of the fluctuations of the current at high power.

The second stage of the exercise is concerned with recording the signals given by the LP and HP detection systems during reactor operation at different levels of power. Table 4 gives an example of the type of results than can be obtained.

TABLE 4. EXAMPLE OF RESULTS OBTAINED WITH THE LOW AND HIGH POWER LEVEL NEUTRON DETECTION SYSTEMS

Reactor power	Counting rate (cps) [1]	Current (A)
Source level	10	2×10^{-9}
1 W	10^3	3×10^{-9}
10 W	10^4	5×10^{-8}
100 W	10^5	5×10^{-7}
1 kW	7×10^5	5×10^{-6}
10 kW	2×10^6	5×10^{-5}
100 kW	—	5×10^{-4}
1 MW	—	5×10^{-3}

[1] The counting rate is given in counts per second (cps).

As an example, the following points can be addressed according to recorded data:

(a) Linear increase of the signals, i.e. the counting rate and current, with the reactor power in the respective operating range of the LP (source level to 100 W, in the example given in Table 4) and HP (10 W to 1 MW, in the example) detection systems;
(b) Saturation of the counting rate of the LP system when the power increases (above 100 W, in the example), which arises from the overlap of the pulses with a large increase in the number of neutron interacting in the detector per unit time;
(c) Consistency between the observed maximum value of the counting rate and the value calculated from the FWHM of the pulses in the first stage;
(d) Calculation of the neutron flux on both LP and HP detectors, which is given by the ratio of the counting rate or current, respectively, to the sensitivity of the detector;
(e) Comparison of the neutron fluxes calculated by different detectors, which usually shows non-uniform neutron flux distribution around the core, resulting in a specific calibration factor (i.e., power to signal ration) for each detector;
(f) Establishment of the operating range of both systems according to the recorded data: marked in grey in Table 4;
(g) Overlap between the operating range of the LP and HP systems and its importance for safe operation;
(h) Discuss the link between the counting rate, current and reactor power, a corresponding calibration factor being defined for each detector. This calibration factor is periodically adjusted following a power calibration campaign (see Section 10.1);
(i) Explain the use of the rate of change in the counting rate and current for the calculation of the period or doubling time, which is a key parameter that ensures safe RR operation.

The third stage of the exercise concerns the understanding of the use of these signals by the reactor protection system. The following points can be addressed:

(a) Emphasis on the fact that neutron detection systems play an essential role in ensuring safe RR operation and control;
(b) Definition of a low level threshold for the counting rate at the source level, typically 5 cps, which ensures the proper operation of the LP detection systems before reactor start-up;
(c) Definition of a high level threshold for the counting rate corresponding to the upper value of the operating range of the system, 10^5 cps according to Table 4. This threshold triggers the reactor SCRAM if the LP system is used out of its operating range;
(d) Definition of a low level threshold for the current, which correspond to the lower value of the operating range of the system, 5×10^{-8} A according to Table 4. This threshold triggers the reactor SCRAM if the HP system is used out of its operating range;
(e) Definition of a high level threshold for the current, equivalent to the nominal power plus a given margin, typically 10%, which would correspond to a current of 5.5×10^{-3} A. This threshold induces a reactor SCRAM;
(f) Definition of thresholds for the period (or doubling time) of the reactor, for example, an alarm if the doubling time is less than 10 s and a reactor SCRAM if the doubling time goes down to 3 s;
(g) Explanation of the I&C system's logic, which often initiates a manual switch from the LP to HP detection system when increasing the power.

Depending on the facility, additional thresholds used to trigger additional alarms may be defined. A focus on the activity related to the periodic testing of the detection systems may also be provided to the students.

4.4.4. Safety considerations

The safety considerations given in Section 3.5.4 apply. This Section describes additional safety considerations specific to this exercise.

When considering the implementation of additional detectors to perform the exercise, a safety analysis should be conducted, referring to Section 3.5 and to Safety Standards No. SSG-24 'Safety in the Utilization and Modification of Research Reactors (Rev. 1)' [10]. It is important to check that the detection setup, which may include a specific holder or beam tube, does not influence reactivity nor the measurements made by the detectors of the I&C system. The latter issue may arise if an additional detector is installed in the vicinity of a detector of the I&C system. Implementation of additional detectors may also lead to supplementary radiation protection issues. These can include changes in the shielding at the facility when using neutron beam ports (which may impose a limit on the maximum operating power to restrict the dose rate in the reactor hall) or precautions when handling and storing activated devices such as detectors, cables and holders. For all these reasons, implementation of additional detectors may require review and authorization by the safety committee of the RR or the regulatory body.

From a security or safeguards point of view, handling fission chambers with highly enriched uranium (for instance, while implementing additional detection channels) can be subject to specific rules and restrictions.

4.4.5. Documentation

In order to conduct this exercise, the following documents can be given to the students:

(a) Background documents on neutron detectors and neutron detection systems working in the pulse and current mode;
(b) Technical specifications of the detectors, including the type, size, operating and nominal voltage and sensitivity;
(c) Technical specifications of the electronics, such as the gain of the amplifier and the values of the voltage supply and discrimination voltage threshold;
(d) Schematic of the core and its vicinity showing the position of the neutron detectors;
(e) Step by step procedure to complete the task;
(f) Specific experimental related information and rules when needed.

4.4.6. Questions to the students

Following the exercise on neutron instrumentation, the following set of questions can be addressed to the students:

(1) Explain briefly the principle of the LP detection systems, including the role of the discriminator.
(2) What limits the maximum counting rate of the LP detection system and what is the link between the maximum counting rate and the FWHM of the pulses at the output of the discriminator?
(3) A LP detector with a sensitivity of 0.1 cps for a unitary flux of 1 $n.cm^{-2}\cdot s^{-1}$ gives a counting rate of 2580 cps. What is the neutron flux impinging on this detector?
(4) Give two reasons why it is important to have a wide overlap (at least 2 decades in the instrumentation LOG scale) between the operating range of the LP and HP detection systems.

(5) Propose a practical way for the protection system to SCRAM the reactor when its power reaches 115% of the nominal reactor power. We assume that at the nominal power, the current of the HP detection system is equal to 2×10^{-3} A).

(6) Explain how the HP and LP detection systems can be successively calibrated following a power calibration campaign.

Further evaluation of the impact of the experiment can be obtained through a deliverable such as a report or a presentation on the objectives, methodology and results obtained from these measurements, prepared by the students and delivered to the teacher.

For further information on the reactor instrumentation exercises, please consult the bibliography.

5. NEUTRON FLUX MEASUREMENT

5.1. BACKGROUND

The exercises regarding neutron flux[5] distribution in RRs, which are also referred to as 'neutron flux mapping' exercises, are common at low power RRs. Two principal methods are used for thermal neutron flux distribution measurement.

The first method consists of using various types of small-size neutron detectors such as self-powered neutron detectors [15] or ionization chambers, which are inserted into a certain position in the core and are gradually moved in one direction, usually vertically. The signal (pulse, or current mode) measured at the detector, which is sensitive to thermal neutrons, is proportional to the thermal neutron flux. Therefore, the position dependent signal is used to establish the thermal neutron flux distribution in the core.

The second method is based on neutron activation of samples, similarly to the NAA technique[6]. Various activation detectors, e.g. activation foils or wires, can be placed into the core and irradiated. The resulting measured sample activity, which is proportional to thermal neutron flux, is used to establish the neutron flux distribution.

5.2. THEORY

The neutron flux distribution in the core of a RR results from interaction processes which depend on the neutron energy. The neutron energy spectrum in a thermal reactor is affected by processes connected to the neutron life cycle in the core, from its origin as a fast neutron[7], through its slowing down, i.e. moderation to thermal energy[8], until the neutron is either captured by the fuel or by other materials in the core, or until it leaks out from the core. The neutron spectrum, which covers the wide range of energies from approximately 10 MeV to approximately 10^{-4} eV is usually divided into three regions: thermal, epithermal, and fast energy region. The neutron flux density[9] of the thermal reactor as a function of neutron energy E can be described as follows:

$$\varphi(E) = \varphi_{th} \frac{E}{(kT)^2} e^{-\frac{E}{kT}} + \varphi_{epi} \frac{1}{E} \Delta\left(\frac{E}{kT}\right) \tag{1}$$

where

$\varphi(E)$: neutron flux density at neutron energy E

φ_{th}: thermal neutron flux density

φ_{epi}: epithermal neutron flux density

T: neutron temperature

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[5] Neutron flux is defined as the total path length of all the neutrons in a cubic centimetre in a second, or as the number of neutrons crossing through some arbitrary cross-sectional unit area in all directions per unit time.

[6] See Section 11 on neutron activation analysis for further information.

[7] The average energy of fission neutrons is approximately 2 MeV.

[8] The average energy of thermal neutrons is approximately 0.025 eV at a temperature of 290 K.

[9] Product of neutron density and neutron velocity integrated over all directions of neutron movement. Unit: $/cm^2 \cdot s^1$.

$\Delta(E/kT)$: joining function

Eq. (1) is often rewritten as a function of the thermal neutron flux density as:

$$\varphi(E)=\varphi_{th}\left\{\frac{E}{(kT)^2}e^{-\frac{E}{kT}}+\frac{\lambda}{E}\Delta\left(\frac{E}{kT}\right)\right\} \tag{2}$$

where $\lambda = \varphi_{epi} / \varphi_{th}$ is the ratio between the epithermal and thermal neutron flux densities.

The thermal neutron flux distribution in a RR is usually measured given that it has the highest influence on reactor safety because it is proportional to the distribution of heat generated in the core. This means that high peaks in the thermal neutron flux cause hot regions in the reactor core. The other two flux distributions, i.e. of epithermal and fast neutrons, are measured less frequently.

The thermal neutron flux distribution can be determined from diffusion theory for an infinite one-dimensional bare slab reactor with thickness a (i.e. reactor without reflector). The diffusion equation can be written as:

$$\frac{\partial^2\varphi}{\partial x}+B^2\varphi=0 \tag{3}$$

where B is the buckling factor. The thermal neutron flux density in the core can be determined using boundary conditions that describe the fact that the thermal neutron flux density on extrapolated surfaces $\tilde{a}$ reaches a value of zero:

$$\begin{aligned}&\varphi(\frac{\tilde{a}}{2}) = \varphi(-\frac{\tilde{a}}{2}) = 0\\&\tilde{a}=a+2d\end{aligned} \tag{4}$$

where d is the extrapolation length.

Eq. (4) can be solved using the boundary conditions in Eq. (4), and the thermal neutron flux density φ in the bare reactor is equal to:

$$\varphi(x) = A\cos(\frac{\pi x}{\tilde{a}}) \tag{5}$$

In the case of a reactor with a reflector, a similar approach can be used using a two-group diffusion equation, with one group for the core and the second group for the reflector. From the solution of the diffusion equations it can be found that the thermal neutron flux rises near the core–reflector border and a peak can be found in the reflector. The peak is caused by thermalization of fast neutrons in the reflector, where the thermal neutrons are slightly absorbed. Thus thermal neutrons are accumulated in the reflector before they escape out or returned back to the core, causing the flattening of the neutron flux distribution in the core. An example of the neutron flux density distribution in the bare reactor and the reactor with reflector is shown in Fig. 3.

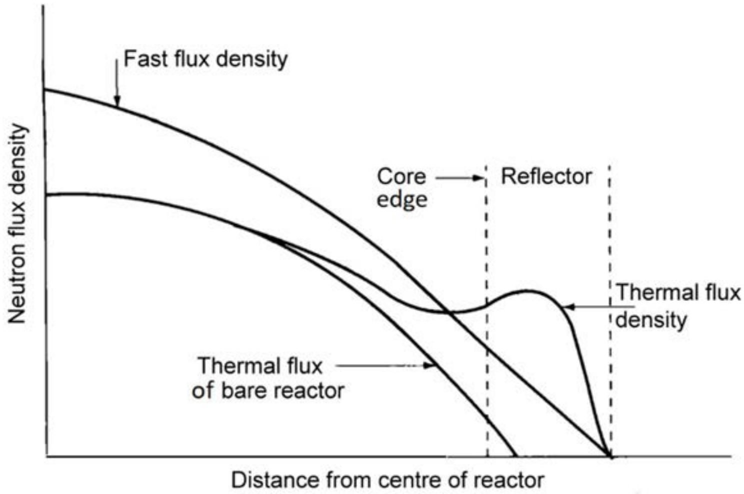

FIG. 3. The neutron flux density distribution in the reactor with and without reflector. (courtesy of the CTU, Czech Republic)

The reflector in the reactor redirects some thermal neutrons back into the core; this reduces the critical mass, and thus the critical dimensions of the core. In order to consider this effect, the reflector savings δ is defined as in Eq. (6):

$$\delta = \widetilde{R}_0 - R \tag{6}$$

where $\widetilde{R}_0$ is the critical diameter of the bare reactor and R is the critical diameter of the reactor with a reflector. For a water moderated and reflected reactor, the following formula can be used to roughly estimate the reflector savings:

$$\delta \cong \frac{\overline{D}_C}{\overline{D}_R} L_R \tag{7}$$

where

$\overline{D}_C$: diffusion coefficient in the core

$\overline{D}_R$: diffusion coefficient in the reflector

L_R: diffusion length in reflector

Neutron detectors (based on ^{10}B or ^{235}U convertor) and Au or Mn activation samples are sensitive not only to thermal neutrons but also to epithermal neutrons. This sensitivity depends of the cross section of the convertor or activation sample for the neutron kinetic energy. In order to obtain pure thermal neutron flux density distribution, it is necessary to cut off the epithermal neutron part from the measurement. The boundary energy between thermal and epithermal neutrons has a value of approximately 0.1 eV. Cadmium (Cd), which is often used in the reactor, has significantly high absorption cross section for neutrons at low energies with a resonance peak at energy of approximately 0.18 eV and significantly low absorption cross section for neutrons at energies above this peak. That is why Cd is often used as a filter for thermal neutron detection. An initial measurement with a bare detector is carried out, and both thermal and epithermal neutrons are detected. In the second step the measurement with the same detector but in cadmium casing is carried out, and only epithermal neutrons are detected. Through comparison of both measurements, the thermal neutron flux is determined. The cadmium ratio R_{Cd} is used in the calculations, defined as:

$$R_{Cd} = \frac{C_{bare}}{C_{Cd}} = \frac{C_{th} + C_{epi}}{C_{epi}} \tag{8}$$

or:

$$R_{Cd} = \frac{\frac{C_{bare}}{M_{bare}}}{\frac{C_{Cd}}{M_{Cd}}} = \left(\frac{C_{bare}}{C_{Cd}}\right).\left(\frac{M_{Cd}}{M_{bare}}\right) = \left(\frac{C_{th} + C_{epi}}{C_{epi}}\right).\left(\frac{M_{Cd}}{M_{bare}}\right) \tag{9}$$

where

C_{bare}: count rate measured at the bare detector
C_{Cd}: count rate measured at the detector with Cd casing
C_{ep}: count rate corresponding to the epithermal neutrons
C_{th}: count rate corresponding to the thermal neutrons
M_{bare}: mass of foil or wire
M_{Cd}: mass of foil or wire with Cd casing

Eq. (8) is used for the determination of R_{Cd} by the ionization chambers, and Eq. (9) by the activation detector where different mass of foils or wires are considered. In practice, unless measurements of C_{bare} and C_{Cd} can be done simultaneously, the calculation will have to take into account the radioactive decay time after sample irradiation.

5.3. EDUCATIONAL ASPECTS

From the reactor point of view, no specific experimental instrumentation is needed for carrying out the neutron flux mapping.

When using small-sized self-powered neutron detectors or ionization chambers, associated reading electronics and cadmium filtering device are necessary to carry out this exercise. When using activation based measurements, appropriate activation samples and a gamma ray spectrometry system are needed for NAA.

Depending on the level of the exercise, the students should have a prior minimum background on:

(a) Reactor principles and the role of its major components (fuel, moderator, reflector, absorbent);
(b) Neutronics (neutron energy spectra and distribution);
(c) Principle and use of the neutron detectors;
(d) Principles and measurements based on neutron activation (use of different materials and cadmium case).

Contents on this Section, Section 4 and Section 11 and their references can be used for the establishment of this minimum background.

This exercise is appropriate for bachelor’s, master’s and doctoral programmes. The level of the exercise can be adjusted according to the student’s background and pedagogical objectives. It is usually basic or intermediate level, but can also be applied to an advanced level programme.

For the basic level, the objectives of the exercise can be limited to the measurement of thermal neutron flux distribution (with one technique), followed by the understanding of the shape of neutron flux distribution according to the configuration of the core. The duration of such an exercise is typically three hours.

For the intermediate level, the following additional points can be investigated:

(a) Comparison of the neutron flux distribution given by two techniques (direct measurement and through neutron activation);
(b) Investigations about neutron spectrum and understanding of the neutron flux distribution according to their energy range;
(c) Study the influence of perturbations in the core configuration based on neutron flux distribution measured. Duration of such an exercise can range between 6–12 hours.

Finally, in the frame of an intermediate to advanced level micro project, students can develop a validation process of neutronics codes by the means of the comparison of the result obtained through calculation and the neutron flux measurements made on the reactor by one or the two flux mapping techniques.

5.4. EXERCISES ON NEUTRON FLUX MAPPING

5.4.1. Objective of the exercise

The neutron flux mapping exercise can easily be conducted in RRs as stated in Section 5.3. This exercise can have two complementary aims:

(a) Learn how to use detectors to measure neutron flux distribution;
(b) Understand the shape of the neutron flux distribution and its relationship to the reactor physics and the core design.

Measurement of neutron flux distribution can be carried out with two techniques as stated in Section 5.1: (1) direct measurement of neutron flux with small size detectors (usually gas detectors such as fission chambers or self-powered detectors) or (2) post-measurement of neutron activation of foils or wires placed in or/and around the core. Additional investigation about neutron spectrum (thermal, epithermal, fast) or on the modification of neutron flux distribution by modifications of the core configuration (removal of a reflector or insertion of an experimental device for example) can also be carried out.

5.4.2. Equipment and conditions

Except for the instrumentation needed to measure the neutron flux distribution in the core or its vicinity, no specific equipment is needed for this exercise.

When a direct measurement is carried out with neutron detectors, the following equipment is needed:

(a) Small size gas detector, such as fission ionization chambers, or self-powered detector with their associated cable and electronics for signal reading;
(b) Specific detector holder to move the detector in or around the core.

Guidelines for equipment related to the neutron detection with gas detectors is given in Section 4. Some limitations related to safety or radiation protection issues may apply when handling the detectors in the core. It is preferable to carry out the experience at very low power in order to limit the dose rate around the core, as well as to limit the activation of the detector, especially if the detector has to be removed later on from the core or pool of the reactor (the time for activity decay should be considered).

When carrying out the measurement through activation of foils or wires, the following equipment is needed:

(a) Activation foils or wire detectors with known mass(es);
(b) Sample holder (often Plexiglas or aluminium) to place the activation detector(s) in the core or vicinity;
(c) 2π counter or spectrometry detection system device (with a NaI(Tl) or HPGe detector) for the measurement of the activity;
(d) When neutron energy filtering is needed, Cd cases for the activation foils.

The type of activation detector is chosen according to its capture cross section, in the energy domain to be investigated, and to the type of isotope that will be produced under neutron irradiation. The activation detector is also chosen according to the type of radiation emitted by the produced isotope, its energy and its period (or half-life). The cross section should be high enough to give measurable activity. Periods are usually in the range of a few minutes to a few days. Short period (a few minutes) materials need only short irradiation times to have a significant activity. In that case, the measurements will have to be corrected taking into account the time interval between irradiation and measurement. From a radiation protection point of view such samples need caution in handling because of their potentially high activity, and the mass of the sample will be limited to the minimum sufficient to reach significant counting. Common activation detectors (radioactive isotope produced, half-life) include copper (^{64}Cu, 12.7 hours), gold (^{198}Au, 2.69 days), copper–gold alloy (typically 98% Cu and 2% Au) and Fe (^{56}Mn, 2.58 hours).

Some limitations, related to safety or radiation protection issues, may apply when handling the activation detectors. In particular, the dose rate for the students should be kept as low as possible when removing the irradiated samples and measuring their activity . Specific sample holders can be developed to limit the exposure during the experiment. Standard sources may also be used for the calibration of the measuring system in order to get a precise determination of the neutron flux. This is the case of ^{54}Mn standard gamma emitting sources (835 keV) that can be used for the calibration of the detection system for ^{56}Mn (gamma at 847 keV).

Most of the neutron detectors will preferentially result in the establishment of the thermal neutron flux distribution. Indeed, most of the absorption cross sections vary as one over the square root of neutron kinetic energy. For example, fission chambers are about 600 times more sensitive to thermal neutrons at 0.025 eV than to 1 MeV neutrons.

5.4.3. Methodology

This subsection describes a three-stage exercise that can be developed for neutron flux mapping. These stages can be carried out individually or as a single exercise, according to the level of the students and the curriculum. Since a wide variety of detectors and detection systems can be used for neutron mapping, this paragraph, as an example, considers only the use of one type of neutron detector (small size fission chamber) and two types of activation detectors (gold and iron) that can be used for this exercise.

Stage 1 — Neutron flux distribution

This stage concerns the measurement and understanding of the global shape of the neutron flux distribution.

Using a small size fission chamber, the detector connected to its detection system is moved step by step (e.g., every 5 cm) in the vicinity of or in the core. This exercise can be carried out along the vertical or the horizontal axis, using a beam channel for example. For each position of the detector, the signal is recorded in order to plot the position dependent signal along the considered axe. Using a calibrated detector of well-known sensitivity, it is also possible to obtain the corresponding value of the neutron flux.

Activation detectors, such as gold foils or wires are commonly used because of the simple decay pattern of ^{198}Au and its suitable half-life (2.7 days). Foils (e.g., one every 5 cm) or wire to be activated are fixed within the sample holder that can be introduced in the core or its vicinity. After irradiation, the irradiated sample(s) can be measured either with a 2π counter or a gamma spectrometer to measure the sample activity which is proportional to neutron flux. From a practical point of view, it is possible to cut the wire, for example in 5 cm long samples, in order to measure the individual activity of each part of the wire. Activity versus position of the irradiated samples will result in neutron mapping along the investigated direction.

Fig. 3 gives the typical neutron flux distribution in the core of the reactor with and without a reflector. The neutron density exhibits a maximum in the centre of the core and decreases to its periphery. This behaviour can be modelled using neutron diffusion theory (see Section 5.2). In the presence of a reflector around the core a local increase of the overall neutron density, which is related to the increase in thermal neutron contribution, is observed in the vicinity of the reflector.

Stage 2 — Determination of thermal and fast neutron flux distributions

The second stage concerns a simple investigation about thermal and fast neutron flux distributions.

To obtain the pure thermal contribution to the neutron flux, a procedure that implies carrying out two successive measurements can be used. One measurement needs to be made without a cadmium shield, i.e. using the full neutron spectrum, and the second with a cadmium shield, i.e. removing the thermal component from the spectrum. Then, the purely thermal contribution can be calculated based on the two measurements. The experiment can be carried out with a fission chamber successively self-standing and enclosed in a cadmium container. The quantity of cadmium utilized for the shielding needs to be limited to the minimum necessary, because the strong neutron absorption by cadmium may result in significant perturbation of the neutron flux and, potentially, a change in reactivity. Based on the same principle, the procedure can be implemented utilizing activation detectors instead of fission chambers. Fig. 4 shows the energy dependent cross sections of cadmium and gold and can be used to illustrate the principle of this subtracting method using gold samples.

The fast neutron flux distribution can be measured with iron samples whose cross section is significant for neutrons with energy above 6 MeV. Comparison of the thermal and fast contributions to the neutron flux distribution is expected to give a result similar to the one shown in Fig. 3. However, it has to be noted that accurate measurement of fast neutron flux distribution is more complex than for thermal neutron flux distribution as a result of low values of the cross sections in the fast energy region.

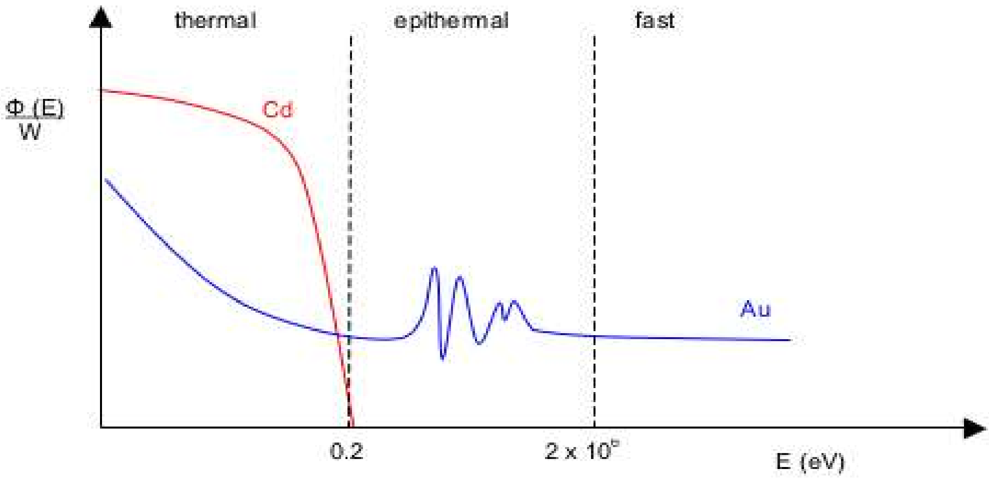

FIG. 4. Comparison of the Gold and Cadmium absorption cross sections as a function of neutron kinetic energy. (Courtesy of Atominstitut, Technical University, Vienna, Austria)

Stage 3 — Influence of core perturbations on neutron flux distribution

The technique previously described for establishing the neutron flux distribution (total, thermal or fast energy domains) can be used in this stage. Starting with a reference state studied in phases 1 and 2, one or more core perturbations to the core can be inserted, in order to study the corresponding changes in the neutron flux distribution. Two perturbations that can be studied are:

(a) Insertion of a control rod that locally lowers thermal neutron density;
(b) Addition of reflecting elements at the periphery of the core, increasing the thermal neutron density in the core periphery.

5.4.4. Safety considerations

The safety considerations given in Section 3.5.4 apply. This Section describes additional safety considerations specific to this exercise.

When implementing an additional device (such as detectors, activation samples, cadmium shield or sample holder) placed in or around the core (including beam channels) it is important to check for its possible impact on core reactivity and the measurement of reactor power.

From a radiation protection point of view, implementation of additional devices submitted to neutron activation can lead to additional risk of exposure either during or after irradiation. It is common, for radioisotopes having a period of a few hours , to wait for an hour or more for the decay of short lived radioisotopes (aluminium for example).

Specific procedures apply when students are involved in handling and implementing additional detectors. In particular, when carrying out the experiment with activation detector some basic rules apply:

(a) Calculate the expected activity of the sample to be activated, limit its mass consequently and define the time for decay before the samples are removed from the irradiation position and placed for counting;
(b) Each person participating in the exercise (teachers, students and reactor operators) should have a personal dosimeter;
(c) Activity of the irradiation device should be systematically checked when taking the activated sample out of its irradiating position;
(d) Use lab coat, gloves and tweezers for the manipulation of irradiated samples,
(e) While not measured, samples are kept under biological shielding;
(f) Each participant should check for possible contamination (using hand–foot monitor) after the exercise;
(g) In order to reduce the radioactive waste, short half-life isotopes should preferably be used. Reusing the samples is advised, ensuring a good traceability of the samples' history;
(h) Caution should be taken with cadmium because of the associated biochemical risk.

5.4.5. Documentation

In order to conduct this exercise, the following documents can be given to the students:

(a) Background (according to the level of the exercise): reactor physics, reactivity, neutron density and its geometrical dependence, neutron detection systems, nuclear interaction, cross section, neutron activation, radioactive decay, basic rules in radiation protection;
(b) Characteristics of the neutron detectors, activation detectors and detection system (including their setting) as needed;
(c) Schematic of the core and its vicinity, showing the position where detectors can be placed;
(d) Step by step procedure to complete the tasks;
(e) Specific experimental related information and rules, especially for the use of activation detectors, (see Section 5.4.4).

5.4.6. Questions to the students

A first evaluation of the impact of the experiment can be obtained using the following set of questions:

(1) Explain briefly the principle of technique(s) used for neutron measurement.
(2) By which mean is it possible to determine the thermal neutron flux distribution without the epithermal and fast contributions?
(3) What should be the characteristics of a proper activation detector considering the investigated neutron energy range, the activity measuring aspects and the radiation protection issues?
(4) What is a typical shape of thermal neutron flux distribution from the centre of the core toward its periphery? Which mathematical model can be used to establish such a distribution?
(5) What is the difference in the general shape of the fast and thermal neutron flux distributions?
(6) Assuming that graphite elements are placed around the core, what would be the change in neutron flux distribution? Why is the graphite installed surrounding the core?
(7) What would be the effect on the neutron flux distribution when a control rod is inserted? Is this effect identical for thermal and fast neutrons?

Further evaluation of the impact of the experiment can be obtained through a deliverable such as a report or a presentation on the objectives, methodology and results obtained from these measurements, prepared by the students and delivered to the teacher.

For further information on the neutron flux measurement exercises, please consult the bibliography.

6. CRITICALITY EXPERIMENT

6.1. BACKGROUND

During standard reactor operation, maintaining the reactor at criticality or initiating transients through subcritical or supercritical states to modify the reactor power is a routine part of the operators' work. These deviations from criticality are well described in the operating procedures, and they are executed taking into account critical configuration(s) previously achieved during reactor operation, usually a few minutes or a few hours before. Operators are trained to maintain the RR in a safe state within the OLCs.

A completely different situation can occur when surveying the reactor state during refuelling or when starting up the reactor after refuelling or after implementation of new equipment, in the core or its vicinity, that significantly influences core reactivity. In such cases, checking or approaching criticality is always connected with some uncertainties that cannot be avoided even with the most accurate criticality calculations available and with experienced reactor staff. Indeed, discrepancies from the predicted critical parameters can lead to incidental or accidental conditions. The most severe accident that can occur if a criticality experiment is not performed in a proper and safe way is prompt criticality. Therefore, the exercise related to the critical experiment, also known as the approach to criticality, has to be performed very precisely, with a deep understanding of all phenomena that can affect the experiment and with the appropriate methodology. Almost all RRs in the world use the methodology that is described below.

Approaching criticality in light water moderated reactors can be achieved by changing one of the three following reactor characteristics: (1) the amount of nuclear fuel in the core, i.e. by adding fuel; (2) the neutron absorption rate in the core, i.e. by control rod withdrawal or boric acid dilution; or (3) the moderation in the core, i.e. by increasing the moderator level.

The criticality experiment is one of the most frequent exercises in student instruction. Indeed, the critical experiment constitutes the preliminary step to RR operation, and it also permits students to become familiar with reactor design and control. Additionally, this exercise needs simple theoretical basis and, in practice, it is simple to conduct by the control rod withdrawal or water level increasing methods, which require no specific experimental equipment or instrumentation. Slightly more complicated and more time consuming are criticality experiments based on adding fuel to the core. Whichever method is used, such exercises aim to provide students with the knowledge, method and skills for conducting a criticality experiment.

6.2. THEORY

The main objective of the criticality experiment is to safely approach and determine by extrapolation the reactor's critical state. Reactivity is increased in a step-by-step and controlled manner, for example by the staged withdrawal of a control rod or the loading of fuel elements in the core. At each step the neutron counting rate is measured, and, using the counting rates from current and previous steps, the critical state (i.e., the critical position of the rod or the number of fuel elements necessary to make the reactor critical) is predicted. Using this predicted critical state and the corresponding value of the so-called critical parameter (i.e., position of the rod or number of fuel elements), the reactivity can be further increased while ensuring that the reactor is maintained subcritical, until a good estimate of the critical state is obtained. This iterative process is repeated until a very small subcriticality of typically a few cents is reached, i.e., k_{eff} nearly approaches 1. At this stage, the criticality experiment is finished, and an operator using a routine procedure can safely achieve a critical and later a supercritical state to start up the reactor.

The critical experiment is based on the application of the subcritical multiplication method, which is commonly described in reactor physics textbooks or RR exercise manuals. The evolution of the neutron population relates to the effective multiplication factor k_{eff}, which is an essential parameter of the reactor. According to its definition k_{eff} represents the ratio between the number of neutrons in the current neutron generation n_i and the number of neutrons in the previous generation n_{i-1}. The value k_{eff} is also the ratio between the number of neutrons produced in one generation $n_{production}$ and the number of neutrons absorbed in the core $n_{absorption}$ and leaked from the core $n_{leakage}$ in the previous generation. Lastly, the effective multiplication factor can be defined by the six factor formula, where η is the thermal fission factor, ε is the fast fission factor, p is the resonance escape probability, f is the thermal utilization factor and P_F and P_T are the fast and thermal neutron non leakage probabilities, respectively. The three definitions are provided as follows:

$$k_{eff} = \frac{n_i}{n_{i-1}}; \qquad k_{eff} = \frac{n_{production}}{n_{absorpsion} + n_{leakage}}; \qquad k_{eff} = \eta\varepsilon p f P_F P_T \tag{10}$$

In a subcritical state $k_{eff} < 1$, but when an external neutron source is added to the core, the neutrons from the source are multiplied in the core, and after a period of stabilization a steady state neutron flux is achieved. In this steady state, the thermal neutron flux is proportional to the effective multiplication factor. Let's assume that the reactor can be described by a single point approximation, meaning that thermal neutron fluxes in the core and reflector are proportional. Then, in any position of the core or reflector, a neutron detector would give a measured signal n that is directly proportional to the neutron flux. Let's assume that at a starting point of such a measurement, when the reactor is in a steady state, the thermal neutron flux exhibits a value n_0.[10] When a critical parameter is changed, e.g., a control rod is withdrawn or fuel is added, an increase of k_{eff} from its initial value k_{eff0} to the value k_{eff1} results, increasing the thermal neutron flux until a new steady state is reached with a measured value n_1:

$$n_1 = n_o + n_o k_{eff\,1} + n_o(k_{eff\,1})^2 + n_o(k_{eff\,1})^3 + \ldots \; n_o(k_{eff\,1})^m \tag{11}$$

where m is the number of neutron generations. As the reactor is subcritical, i.e., $k_{eff1} < 1$, the final number of neutrons is given by the sum of the geometric series with the quotient of k_{eff1}. Thus, it is:

$$n_1 = n_o \frac{1-(k_{eff1})^m}{1-k_{eff1}} \tag{12}$$

If the value of m approaches infinity and $k_{eff} < 1$, the second term in the numerator approaches zero.

$$\text{if} \quad k_{eff\,1} < 1 \quad \text{and} \quad m \to \infty \quad \text{then} \quad (k_{eff\,1})^m \to 0 \tag{13}$$

[10] In this case $n_0 = \varepsilon \cdot N$, where N is the neutron flux in the core, and ε is the efficiency of the neutron detector.

Therefore, the final equation has the form of:

$$n_1 = n_o \frac{1}{1 - k_{eff1}} \tag{14}$$

The same procedure is applicable for a subcritical state with k_{eff2}, and Eqs. (10) to Eq. (15) remain similar, replacing k_{eff1} with k_{eff2}. Thus, Eq. (13) can be rewritten in a general form using k_{effi}:

$$n_i = n_0 \frac{1}{1 - k_{effi}} \tag{15}$$

If the reactor is approaching the critical state, the value of k_{effi} approaches 1. In effect the value of the fraction in Eq. (15) increases to infinity, and its inverse value approaches zero.

$$\text{if } k_{effi} \to 1 \quad \text{then} \quad n_i \to \infty \quad \text{and} \quad \frac{1}{n_i} \to 0 \tag{16}$$

When the inverse value $1/n_i$ from Eq. (16) is plotted on the y axis as a function of the critical parameter x (position of the rod, for example), the curve intersects the x axis (i.e. $1/n_i = 0$) when criticality is reached (see Fig. 5). In practice, while the critical state is closely approached but not reached yet, with the extrapolation of this curve the critical value of the critical parameter can be predicted at the moment when criticality is reached.

$$\text{if } \frac{1}{n_i} \to 0 \quad \text{then} \quad \frac{n_0}{n_i} \to 0 \quad \text{if} \quad n_0 = konst \tag{17}$$

When reaching criticality, $1/n_i$ approaches zero and intersects the x axis. Thus, any constant multiple of it also approaches zero. Therefore, a common practice is to plot n_0/n_i as a function of x; in this case the initial value on the y axis is 1 (maximum value for n_0/n_i), and there is no need to adjust the scale of the y axis.

Starting from an initial state with the neutron thermal flux stable and equal to n_0, the reactivity is increased (step *i,* modified parameter at value x_i). As a result of the increase (k_{eff} approaches 1), the neutron thermal flux increases and stabilizes to a value n_i. The value of n_0/n_i is plotted on a curve (see Fig. 5), and, by extrapolation, a first estimate of the critical parameter is made. This value may be compared with the critical parameter determined by calculation. Taking into account the extrapolated value of the critical parameter, reactivity can be further increased (step $i+1$). For safety reasons, a requirement stipulates that the modified parameter (position of the rod, number of elements, or water level) should not be increased by more than one half of the difference between its present value and the smaller value of the critical parameter that was determined either from the previous extrapolation (see safety margin in Fig. 5) or calculation. In practice and for pedagogical purposes, supposing that the general trend of the curve (see Fig. 6) is not known, a value of x_{i+1} much smaller than that imposed by the safety margin is chosen. Applying this x_{i+1} value, after reactivity addition, the neutron flux increases and reaches a new equilibrium at n_{i+1}. This value is entered in the graph, and the whole process is repeated to reach the value of $n_0/n_i \approx 0.1$, i.e., when the reactor is slightly subcritical by typically a few cents.

Then, the last extrapolation is made, and the operator can reach the critical state using a routine reactor procedure.

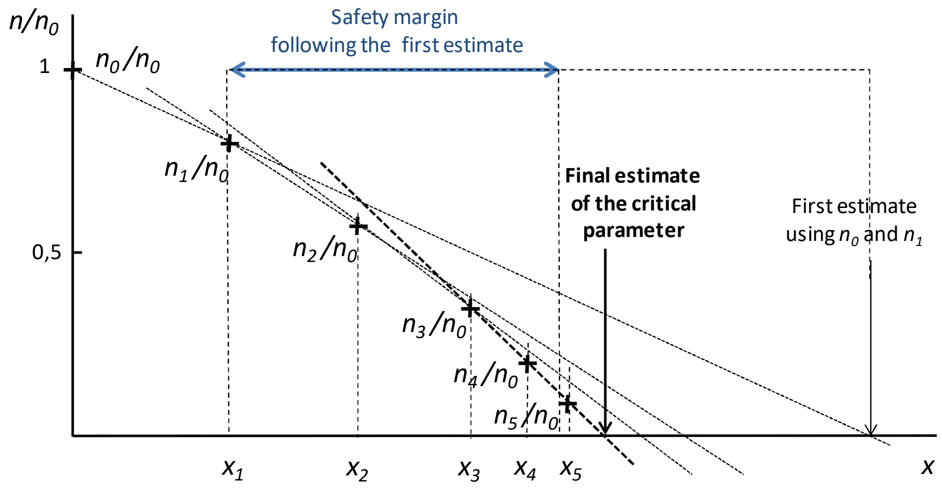

FIG. 5. Step-by-step measurements carried out during a criticality experiment.

Figure 6 represents three different cases or curve shapes that can occur when the reactor approaches criticality. The ideal case follows curve 3. Unfortunately, such a straight curve 3 is generally not attainable in standard reactor conditions. From the nuclear safety perspective, concave curve 1 is disadvantageous because the extrapolated value is higher than the subsequent real value. It is usually achieved when a neutron detector is located very close to the core. Convex curve 2 is more advantageous from a safety point of view, but the angle under which it intersects the x axis leads to an inaccurate intersection point and thus to a less accurate critical state forecast. The shape of the curves depends on a number of factors, the most important of which are the mutual position of the detector, neutron source, and fuel and its change during assembly of the core. Convex curve 2, which is more desirable, is reached when a neutron detector is far from the core. However, in this case, two major issues often arise, as placing a detector at a distance may not be spatially feasible and when possible, the count rate provided by the detector is generally too low to provide good statistics.

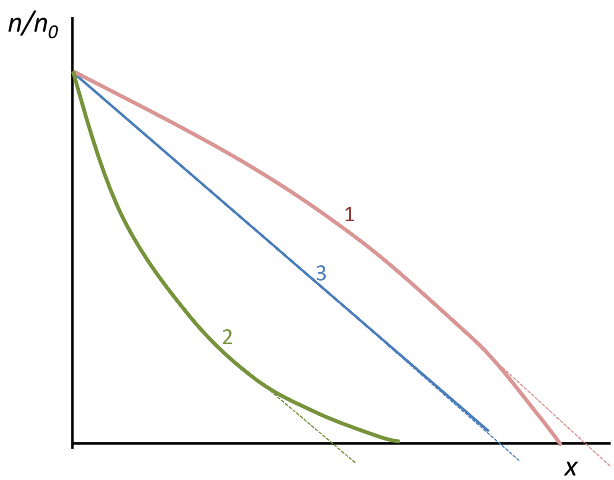

FIG. 6. Three ways observed experimentaly when approaching critical state.

6.3. EDUCATIONAL ASPECTS

The criticality experiment belongs to the group of reactor physics exercises whose general objective is to understand basic principles of reactor physics and their relation to safe operation of a research or power nuclear reactor.
The learning objectives specific to this experiment should include the understanding of:

(a) Neutron kinetics in subcritical, critical and supercritical states in the presence of a neutron source;
(b) Influence of fuel, moderator, and absorber and reflector materials, on reactivity;
(c) Safety issues related to an uncontrolled reactivity increase that can potentially lead to a criticality accident;
(d) Need for the determination of criticality by extrapolation from the subcritical state, after any change in reactor configuration (usually while reactor is shut down) that leads to an unknown reactivity state;
(e) Technique used for the determination of criticality, including the way experimental data can be used to safely conduct the experiment and obtain an accurate estimate of the critical configuration;
(f) Importance of safety culture and an interrogative attitude while conducting the experiment (experimental data should be given the priority over calculated parameters, and human error should always be considered as a possible reason for unpredictable results);
(g) How the estimated critical configuration can be further used to start up and operate the reactor.

This exercise is essential for students who are studying curricula related to reactor design, physics, operation or safety. This exercise is highly suitable for students studying nuclear science as the major curriculum in all three study programmes — bachelor’s, master’s and doctoral. The level of the exercises can be basic, intermediate or advanced, depending on the level of knowledge of the students and the learning objectives.

This exercise is also suitable for students studying various major engineering curricula in master’s and doctoral study programmes such as power engineering, mechanical engineering or electrical engineering with future assignments in conjunction with the minor curriculum in nuclear engineering. The level of exercises in this case is usually basic or intermediate and depends on the level of knowledge of the students and the learning objectives.

For the basic level, the approach to criticality can be conducted by the operators during the presentation of the reactor. The method is explained, and the recorded values of the counting rates N are used to find the critical configuration (plot of $1/N$ as a function of the changed parameter). The main objective is to make the students understand that starting up a reactor by increasing the reactivity requires having a clear knowledge of what will be the resulting increase in the neutron density. The duration of such an exercise is typically 1–2 hours.

For the intermediate level, the approach to criticality is conducted step-by-step, making the students participate in the decision concerning the changes in core parameters that change reactivity. A deeper analysis of reactor characteristics can be carried out to explain the role of the source and the non-linearity of the $1/N$ plot. The duration of such an exercise is typically 2–3 hours.

For the advanced level, the critical experiment can be conducted in the form of a core building experiment. For example, it can be performed during the loading of the last 4 or 5 fuel elements into the core before the reactivity of the core is increased by withdrawing the control rods. The duration of such an exercise can typically be 5–6 hours. An extended exercise can also combine core calculations accomplished with a computational code using experimental data recorded during the practical exercise at the reactor. Such an exercise can be organised as a one or two week mini-project for the students.

6.4. EXERCISES ON CRITICALITY EXPERIMENT

6.4.1. Objective of the exercise

The objective of the critical experiment or approach to criticality is to find by extrapolation a critical configuration of a reactor while it is maintained subcritical ($k < 1$). It is conducted whenever the conditions to make the reactor critical are not known accurately. This includes conditions in which the core configuration or core vicinity (including e.g. the core periphery, irradiation devices, cold and hot sources, and neutron beams) has been modified following, for example: fuel loading; modification of reflecting elements; maintenance work or change of a control rod; or introduction, removal or modification of an experimental setup.

In the critical experiment, the reactivity can be changed by different means such as changing the number of fuel elements, water moderator level or control rod position. Once the approach to criticality has been conducted and the critical configuration of a reactor is known, this configuration can be used to start up the reactor in a controlled and safe way.

6.4.2. Equipment and conditions

Since performing a critical experiment constitutes a standard step in safe reactor operation, no additional or specific equipment is generally needed for conducting this exercise for academic purposes.

To perform the approach to criticality and later to start up the reactor, a neutron source (e.g., an Am–Be source) has to be available to supply neutrons to be multiplied in the core. In addition, the following equipment and conditions are needed:

(a) Fuel elements in sufficient quantity to reach criticality;
(b) A way of measuring the critical parameter to be changed (such as rod position or water level);
(c) At least one neutron detection system to measure the counting rate, which has to be proportional to neutron density in the whole measuring range associated to the exercise;
(d) The neutron source should exhibit a sufficient activity and be placed at a suitable position to obtain a significant counting rate on the neutron detection systems. If necessary and if possible, the source can be moved to increase the counting rate.

For this exercise, it is important to ensure that no time-dependent parameter can interfere with reactivity other than the critical parameter that is changed on purpose to conduct the approach to criticality. Thus, the approach to criticality cannot be conducted after the reactor has been run at a power above a few kW for some time, generally more than a few minutes. Indeed, a time-dependent decrease in water temperature significantly influences reactivity during the exercise. For reactors operated for a long period of time, typically more than 10 hours, above a few hundred kilowatts, core poisoning (see Section 9.2.4), which results in a time-dependent reactivity change for tens of hours after reactor shutdown (48 hours in the case of ^{135}Xe), also entails inappropriate conditions to perform the exercise.

In order to ensure a precise estimate of the critical state the source level has to remain constant during the exercise. In the case of RRs relying on (γ,n) reactions with beryllium elements to produce neutrons, an approach to criticality is not recommended a few hours after reactor operation at medium power, typically a few tens of kW, as the residual power, related gamma dose rate and consequently source level decreases exponentially with time.

Additionally, for this type of reactor, loading burned fuel elements, elements exhibiting a significant residual power and gamma dose rate, importantly results in a simultaneous increase of reactivity and source level. Since such an effect is common in NPPs, the exercise provides a good example of a comprehensive data analysis in the case of such a core loading.

6.4.3. Methodology

The critical experiment can be achieved on any reactor since it is the basic step to be fulfilled before reactor start-up. Its method can be applied to fuel loading, the addition of moderator (e.g., water level) or reflectors (e.g., graphite, beryllium), as well as the removal of absorbers (e.g., control rods).

For pedagogical purposes changing only one parameter during the approach is preferred in order to find the critical value of this parameter. For example, withdrawing only one rod is better than moving different control rods sequentially to similar positions.

It is important to conduct the approach to criticality using a method that can be applied to any RR. For example, the procedure sometimes used at TRIGA reactors, where the entire set of control rods is moved incrementally until a given reference current is obtained on the neutron detection systems, and without plotting the approach to criticality graph, should be avoided, as it is not applicable to other types of RR.

From a practical point of view, the neutron detection systems installed at the reactor gives a counting rate N proportional to the neutron density n. The N values are used to plot the curve $1/N$ as a function of reactivity, or more commonly as a function of a physical parameter x that is changed. When plotting this curve, it is possible to check step-by-step that the rod can be withdrawn further to approach criticality without passing it. Consequently, when approaching criticality, the critical value of the parameter x can be found by extrapolation (see Fig. 5).

When more than one detection system is available, from a pedagogical point of view comparing the different $1/N$ curves to show that they yield a unique critical state, taking into account uncertainties, is advisable.

As an example we consider here the search for the critical position of a control rod. The reactor is initially in a subcritical state with the control rod fully inserted at height $H = 0$. The value of the counting rate $N(H)$ is measured for $H = 0$. The value of $1/N(0)$ is calculated and plotted on a graph of $1/N$ versus the position of the rod H.

The rod is withdrawn to a position H_1. For the first values of H, it is important to give some guidance to the students, as increases in reactivity have to be controlled. The best way to do so is to give the students a table (see Table 5), where the first values of H to be used are fixed (H_1,

H_2, H_3 and H_4 for example here). Based on the knowledge of the reactor characteristics and previous operational state, this will ensure that the reactor is subcritical for these values of H.

The value of the counting rate $N(H)$ is measured for $H = H_1$. In order to obtain the proper value of $N(H)$ it is necessary to wait for the counting rate to reach an equilibrium. Depending on the type of I&C system, this can be done either by comparing the counting rates after some time interval (e.g., every 10 s) or graphically if the I&C can be used to plot in real time N as a function of time. An average value of the counting rate can be established with three successive measurements; for example, an average value is reported in the previous table and used to calculate $1/N$. $1/N(H_1)$ is plotted on a graph of $1/N(H)$ versus H (graph similar to that of Fig. 5). This is the A step of the method.

TABLE 5. REGISTER OF THE COUNTING RATE AND ITS INVERSE VALUE: EXAMPLE WITH FIXED VALUES OF THE INITIAL POSITIONS OF THE ROD

Height of the rod H (mm)		N	$1/N$
0			
$H_1 = 100$			
$H_2 = 170$			
$H_3 = 230$			
$H_4 = 270$			
… *Value to be defined by the students*			
… *Value to be defined by the students*			
… *Value to be defined by the students*			

From the first two points of $H = 0$ and $H = H_1$ and the extrapolation of the curve, it has to be confirmed that the control rod can be moved to the position H_2 without making the reactor supercritical. This is the B step of the method. In the extrapolation, the nonlinear variation of reactivity with control rod height, according to the control rod calibration curve, needs to be considered. This is why offering guidance with fixed values of H to the students is preferable.

Once the possibility to move the control rod to H_2 has been checked, it can be withdrawn to that position. The A and B steps are then carried out at position H_2 in order to check that the rod can be moved to position H_3. The same process (A and B steps) is again completed before proceeding to position H_4.

Upon reaching position H_4 and according to the A and B steps, it is possible to decide to which position the rod can be further withdrawn in order to approach the critical position while the reactor is maintained subcritical.

The approach to criticality is executed until a precise estimate of the critical position of the rod is found. From a pedagogical point of view adjusting, if possible, the configuration of the reactor (positions of the other rods, for example) so that the reactivity of the core varies linearly with the position of the critical rod is advisable. This can be obtained when the critical position is determined to be around the middle of the height of the core. In this case, a better estimate of the critical position is obtained since the extrapolating curve is linear for the last measured data.

Importantly, the plot of the $1/N$ curve gives only qualitative information on the distance to criticality. Such an approach can be completed by an absolute measurement of the reactivity using reactivity measurement methods such as the source jerk method (see Section 8.2 and its references).

6.4.4. Safety considerations

The safety considerations given in Section 3.5.4 apply. This Section describes additional safety considerations specific to this exercise.

As the critical experiment is a standard step before reactor start-up, it is generally part of the standard procedures for the reactor. If the exercise involves fuel handling, special care is needed to ensure safe handling and loading, namely, the prevention of mechanical damage of core components, and strict application of the loading procedures and loading map. Handling of fuel elements in the reactor core is from the safety point of view an important operation that is always performed by trained, qualified and authorized operating staff. Changes in core configurations should be based, prior to their implementation, on neutronics and thermal hydraulics calculations demonstrating the safety of the configuration. They should comply with the values in the OLCs on shutdown margin and with thermalhydraulic safety criteria. Several reactivity accidents occurred in the past at, for instance in the following RRs: NRX (Canada, 1952), RB (Yugoslavia, 1958), Stationary Low Power Reactor (SL-1) (USA, 1961), VENUS (Belgium, 1965) and RA-2 (Argentina, 1984), some resulting in human casualties [11]. Thus, as indicated in Section 3.1, emphasis should be given to raise the awareness of the students on the safety issues related to fuel loading and the approach to criticality in order to strengthen the safety culture and interrogative attitude among the students.

From a radiation protection point of view, conducting the approach to criticality by the withdrawal of a control rod should generally not give rise to additional constraints. On the contrary, specific rules may apply when the approach to criticality is carried out by handling and loading fuel elements or by changing the water level in the core, as additional risk related to radiation exposure may be considerable.

From the security or safeguards points of view, handling and loading of the fuel can be subject to specific rules and restrictions.

6.4.5. Documentation

In order to conduct this this exercise, the following documents can be given to the students:

(a) Background: neutron kinetics in subcritical state with a source;
(b) Schematic of the core configuration, including the position and characteristics of the fuel elements, control rods and neutron detectors;
(c) Step-by-step procedure to complete the task, including details to make the measurement (e.g., wait for equilibrium, average N) and a table to record the N and 1/N values for each state of the core;
(d) Graph paper or software application to plot the curve;
(e) Specific experimental related information and rules (such as: safety rods in their upper position before any change in the core reactivity, keep foreign matter out of the pool — ‘don’t lose your pen in the pool’, and others).

6.4.6. Questions to the students

Following the approach to criticality exercise, the following set of questions can be addressed to students:

(1) Give at least two examples of modifications of a reactor configuration that necessitate the completion of an approach to criticality before the reactor start-up.
(2) What is the role of the neutron source in the approach to criticality?
(3) What is the evolution of the neutron density when increasing the reactivity of the reactor between two successive subcritical states?
(4) Why does it take more and more time for neutron density to reach equilibrium when the reactor approaches criticality?
(5) What curve should be plotted to estimate the critical position of a control rod while withdrawing the rod step-by-step?
(6) Explain why it is important from a safety point of view to carry out an approach to criticality when the critical configuration of the reactor is not known. What could happen if reactivity is increased by too large a value?

Further evaluation of the impact of the experiment can be obtained through a deliverable such as a report or a presentation on the objectives, methodology and results obtained from these measurements, prepared by the students and delivered to the teacher.

For further information on criticality exercises, please consult the bibliography.

7. REACTOR KINETICS

7.1. BACKGROUND

Safe RR operation requires a comprehensive understanding of time-dependent reactor kinetics and/or dynamics[11]. Certain types of reactors have specific behaviour, depending on their design and operating modes. Research reactors are excellent tools for studying reactor behaviour because of their high flexibility of operation, exemplified in start-up, changes in reactor power or intentional shutdown actions. Exercises on reactor kinetics usually deal with reactor behaviour in its different states, i.e., subcritical, critical and supercritical. Usually, these exercises are easily performed at low power RRs, as the operation of such reactors can be specifically dedicated to educational activity and these exercises do not need specific experimental equipment. If applicable, the exercises can be performed with and without an external neutron source to study its influence on the reactor kinetics. Additionally, a study of the basic properties of delayed neutrons and their influence on reactor behaviour can be conducted.

7.2. THEORY

There are three categories of neutrons present in the reactor core: prompt neutrons, delayed neutrons (both resulting from fission reactions) and neutrons from an external source[12]. Prompt and delayed neutrons play different roles in reactor kinetics, with important consequences for the reactor safe control. We study at first the hypothetical case of a reactor with only prompt neutrons; in a second stage, we will include the delayed neutrons.
The behaviour of a hypothetical reactor as a finite multiplying system without an external neutron source and without delayed neutrons can be described by the following equation:

$$\frac{dn(t)}{dt} = \frac{k_{eff} - 1}{l} n(t) \tag{18}$$

where
$n(t)$: neutron density as a total number of neutrons at time t
l: prompt neutron lifetime
k_{eff} : effective multiplication coefficient[13]

This hypothetical reactor is in a critical state, with k_{eff} equal to one, if the neutron density remains constant. When the effective multiplication coefficient k_{eff} is larger than 1, the reactor is in a supercritical state, and when k_{eff} is smaller than 1, the reactor is subcritical. The total number of neutrons in the reactor[14] depends on k_{eff}, and can be obtained by integrating equation (18), assuming that $n(0) > 0$:

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[11] Both terms 'kinetics' and 'dynamics' are used for time-dependent reactor behaviour in various references, and both are often used as synonyms. In some references, the term 'kinetics' is used for time-dependent reactor behaviour without feedback, and 'dynamics' is used for time-dependent reactor behaviour with feedback. Reactor feedback experiments are described in Section 9.4 and 9.5.

[12] External neutrons or neutrons from an external neutron source are all neutrons that do not originate from the nuclear fuel and are usually produced through (α, n) or (γ, n) reactions.

[13] In this case the effective multiplication coefficient is only related to prompt neutrons, and delayed neutrons are neglected.

[14] Total number of neutrons in reactor corresponds to the number of fissions and thus to reactor power.

$$n(t) = n_0 e^{\frac{k_{eff}-1}{l}t} \tag{19}$$

From Eq. (20) it is evident that $n(t) = n_0$ when k_{eff} equals 1. When $k_{eff} < 1$, the numerator is negative, and $n(t)$ decreases exponentially. Lastly, when $k_{eff} > 1$, the numerator is positive, and $n(t)$ increases exponentially. These conditions are shown in Fig. 7. In this hypothetical case, the reactor power increases very rapidly when the reactor is in the supercritical state. For example, for $k_{eff} = 0.0001$ and prompt neutron lifetime[15] $\ell = 10^{-5}$ s, the number of neutrons, and thus reactor power, would increase e^{10} times in one second. Such a rapid increase in power could not be controlled, and would result in a criticality accident. This case, which assumes that k_{eff} applies only to prompt neutrons, is purely hypothetical, as was mentioned above.

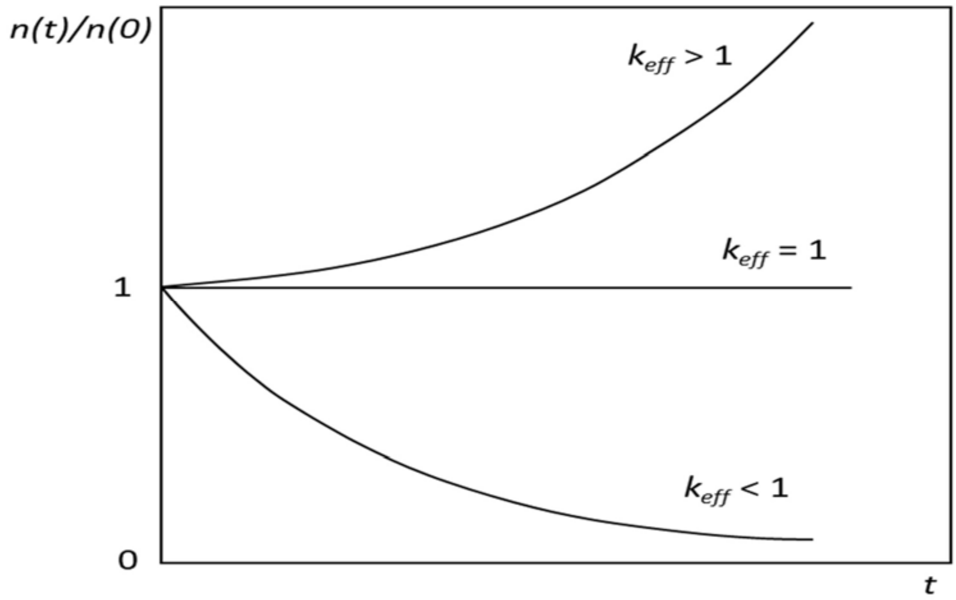

FIG. 7. Evolution of the number of neutrons for a reactor without an external neutron source. (reproduced from [16] with permission of the Czech Technical University in Prague, Czech Republic)

We now consider the case in which an external neutron source is in the reactor. Note that the definition of criticality depends only on k_{eff} and not on the external neutron source. The external neutron source term $S(t)$ is included in Eq. (18).

$$\frac{dn(t)}{dt} = \frac{k_{eff}-1}{l} n(t) + S(t) \tag{20}$$

For a time-independent neutron source, $S(t) = S_0$, and the total number of neutrons in the reactor can be obtained by solving equation (20) with $n_0 = 0$.

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[15] Prompt neutron lifetime for light water reactors is $l = 10^{-5}$–10^{-4} *s*.

$$n(t)=\frac{l\cdot S_0}{k_{eff}-1}\left(e^{\frac{k_{eff}-1}{l}\cdot t}-1\right) \tag{21}$$

However, the reactor behaviour is not obvious from Eq. (21) and therefore a closer analysis is required (see Fig. 8). First, in the case of $k_{eff} > 1$, the numerator in the exponent is positive, and $n(t)$ increases exponentially.

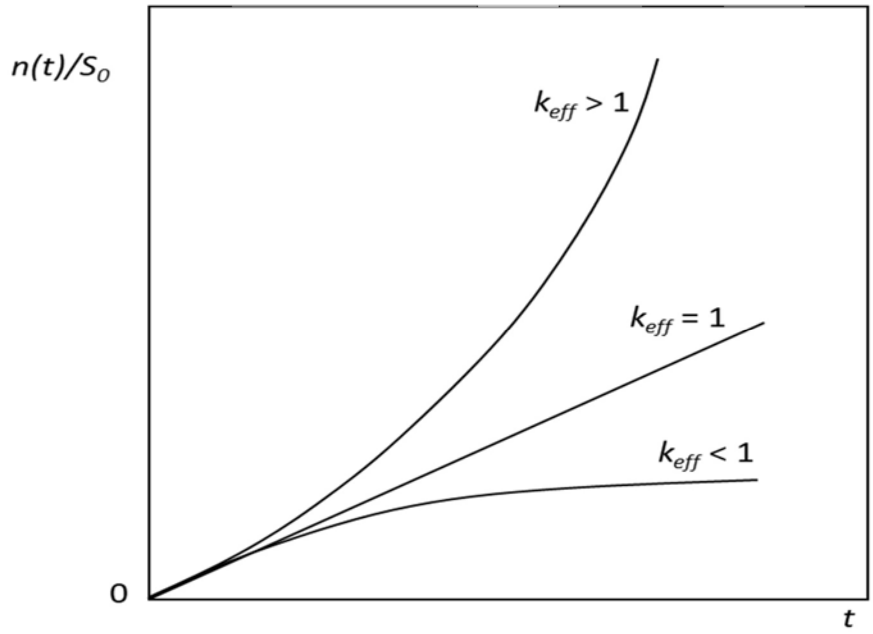


FIG. 8. Evolution of the number of neutrons for a reactor with an external neutron source.

Second, the subcritical reactor behaviour becomes clearer if Eq. (21) is rewritten in the form:

$$n(t)=\frac{l\cdot S_0}{1-k_{eff}}\left(1-e^{\frac{-(1-k_{eff})}{l}t}\right) \tag{22}$$

If $k_{eff} < 1$, the first term in Eq. (21) is positive and $n(t)$ increases at first, but as the exponential term declines over time, $n(t)$ stabilises to a time-independent value $n(\infty)$.

$$n(\infty)=\frac{l\cdot S_0}{1-k_{eff}} \tag{23}$$

Finally, when the reactor is critical, Equation (22) results in $n(t) = S_0\, t$, which means that in a critical reactor with an external neutron source, $n(t)$ increases linearly.

Figure 8 shows in a simple graphic the time evolution of the total number of neutrons in the subcritical, critical and supercritical states. In each case, the external neutron source changes the behaviour of the reactor when compared with a reactor without a neutron source. Exercises dedicated to reactor kinetics under the influence of an external source have to be performed at zero power, i.e., typically less than 0.1 W, so that the contribution of the source, which emits a finite number of neutrons per second, is not screened by the high number of neutrons produced by fission reactions.

Delayed neutrons strongly modify reactor behaviour, because they lead to much slower neutron density changes, which allows safe control of the reactor. In this case, k_{eff} applies to the sum of the prompt and delayed neutrons. The influence of the delayed neutrons on reactor behaviour cannot be neglected, except for large reactivity changes ($k_{eff} - 1 \geq \beta$), which would lead to a prompt critical reactor and a criticality accident as discussed above (see Fig. 7).

Delayed neutrons are produced by radioactive decay of certain fission fragments, the so-called precursors of delayed neutrons. For this reason, it is necessary to include delayed neutron production in kinetics equation Eq. (18) and to describe the time variation of the precursors' concentration. As precursors exhibit decay times that ranges typically from one-tenth of a second to hundreds of seconds, they are often classified in six groups, each group *i* having a mean radioactive decay time λi. In this case, reactor kinetics is described by a system of differential equations.

$$\frac{dn(t)}{dt} = \frac{k_{eff}\left(1-\beta_{eff}\right)-1}{l} n(t) + \sum_{i=1}^{6} \lambda_i c_i(t) \tag{24}$$

$$\frac{dc_i(t)}{dt} = \beta_{effi} \frac{k_{eff}}{l} n(t) - \lambda_i c_i(t) \quad i = 1, 2, ..., 6 \tag{25}$$

Eqs (24) and (25) can be rewritten using reactivity:

$$\frac{dn(t)}{dt} = \frac{\rho - \beta_{eff}}{\Lambda} n(t) + \sum_{i=1}^{6} \lambda_i c_i(t) \tag{26}$$

$$\frac{dc_i(t)}{dt} = \frac{\beta_{ieff}}{\Lambda} n(t) - \lambda_i c_i(t) \quad i = 1, 2, ..., 6 \tag{27}$$

where

$c_i(t)$: precursor's concentration for the i^{th} delayed neutron group

β_{eff}: effective delayed neutron fraction

β_{ieff}: effective delayed neutron fraction for the i^{th} delayed neutron group

Λ: mean neutron generation time ($\Lambda = \frac{l}{k_{eff}}$)

The above description of reactor kinetics behaviour assumes there is no spatial dependence of these parameters throughout the reactor core. This approach is called the point kinetic model of the reactor, and the equations (18) or (26) and (27) are called the one-point kinetic equation(s).

Equations (26) and (27) fully describe the time-dependent behaviour of a reactor operated at low power (typically less than 1 kW), i.e., a reactor without significant feedback effects, in all its operational states.

Various pulse, transient or periodical reactivity changes can be performed in order to study the details of reactor kinetics. These exercises can be considered advanced reactor kinetics exercises and are easily performed at RRs operating deliberately at low power, i.e., typically less than 1 kW. In addition, RRs can be equipped with specific devices, e.g., a fast moving absorbing device, to allow performance of an extended range of exercises.

Figure 9 gives examples of such exercises, showing the change in reactor power (upper curves) induced by a change in the position of a control rod (lower curves), which is representative of the reactivity changes. The figure is a print screen of the reactor console and intended to show the shape of the changes, not detailed numerical values. The left panel of the figure shows the typical response of the reactor for linear periodic changes in reactivity of ±10 cents. The right panel of the figure shows step periodical changes of ±12 cents.

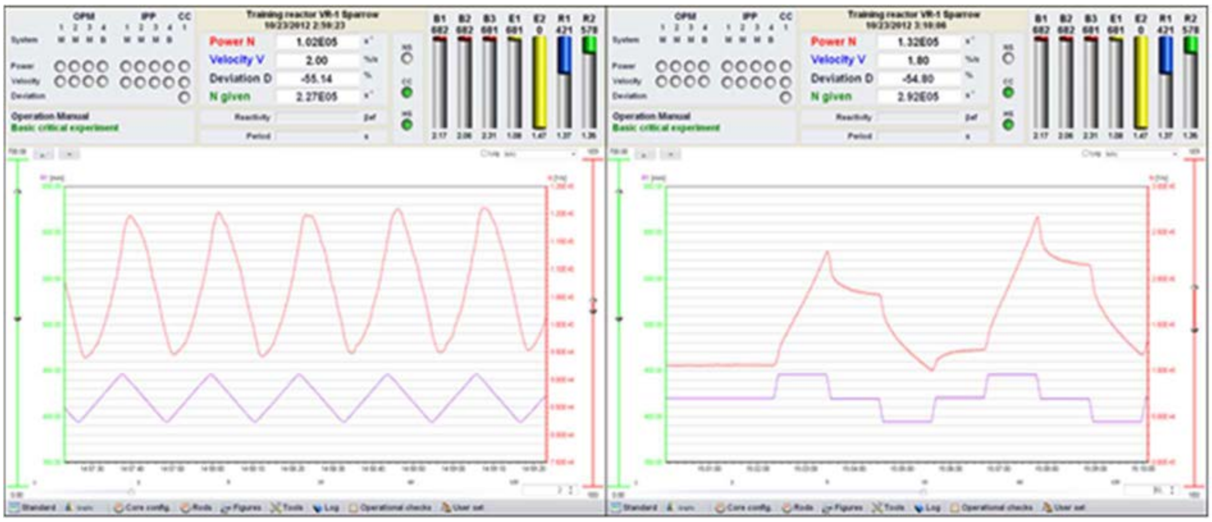


FIG. 9. Examples of typical reactor responses for the periodic reactivity changes. (courtesy of Czech Technical University in Prague, Czech Republic)

7.3. EDUCATIONAL ASPECTS

These reactor kinetics exercises belong to a group of reactor physics exercises whose general objective is to understand basic principles of reactor physics and their relation to safe operation of research or power reactors.

A broad spectrum of basic principles and concepts related to reactor kinetics can be addressed through these exercises. Learning objectives can include the understanding of:

(a) Neutron kinetics in subcritical, critical and supercritical states in the presence of an external neutron source;
(b) Neutron kinetics in subcritical, critical and supercritical states without an external neutron source;
(c) Respective contributions of prompt neutrons and delayed neutrons to neutron kinetics;
(d) Contribution of delayed neutrons from different groups of precursors, i.e., precursors with different radioactive decay times, to neutron kinetics;
(e) Need to maintain the reactor subcritical for the prompt neutrons, i.e., $\rho < \beta_{eff}$, so that the chain reaction is controlled by delayed neutrons;
(f) Safety issues related to an uncontrolled reactivity increase that can potentially lead to a criticality accident if $\rho > \beta_{eff}$;

(g) Importance of safety culture and an interrogative attitude while modifying reactor power: changes in reactivity should be performed with a clear idea of their resulting effect on neutron kinetics, and any deviation from expected behaviour should be identified, analysed and corrected.

These exercises are essential for students who are studying curricula related to reactor design, physics, operation and safety. They are highly suitable for students studying nuclear science as the major curriculum in all three study programmes — bachelor's, master's and doctoral. The level of the exercises can be basic, intermediate or advanced, depending on the level of knowledge of the students and the learning objectives.

These exercises are also suitable for students studying various major engineering curricula in master's and doctoral study programmes such as power engineering, mechanical engineering or electrical engineering in conjunction with a minor curriculum in nuclear engineering. The level of exercises in this case is usually basic or intermediate, depending on the level of knowledge of the students and the learning objectives.

For the basic and intermediate levels, the study of the reactor kinetics can be devoted to the study of a reactor's response in different states, i.e., with the reactor slightly subcritical, critical and slightly supercritical, both without and in the presence of an external neutron source. The main objective of the exercises is to show the behaviour of the reactor and give practical illustration of the equations of neutron kinetics. The duration of such an exercise can typically be 1–2 hours.

For the advanced level, the study of reactor kinetics can include the study of single or periodic transients to show the influence of prompt and delayed neutrons on reactor behaviour. Such exercises, which need a deeper analysis of reactor behaviour, are mostly suitable for nuclear engineering doctoral students and for students with a special interest in master's degree programmes, such as future reactor physicists or nuclear safety engineers. The duration of such exercises can typically be 2–3 hours. An extended mini-project can concern the comparative analysis of calculations conducted using the equations of neutron kinetics and experimental data recorded at the reactor. Such an exercise can be organised as a 1week mini-project for the students.

7.4. EXERCISES ON REACTOR KINETICS

7.4.1. Objective of the exercises

The objective of these exercises is to illustrate and provide a comprehensive understanding of the neutron kinetics that underpins the behaviour of the reactor at low power, i.e., with no feedbacks, which are addressed later in Sections 9.4 and 9.5.

A study of reactor kinetics can examine the different states of the reactor either with or without the presence of an external neutron source. In addition, such exercises can be conducted to study the major contribution of delayed neutrons to reactor control.

7.4.2. Equipment and conditions

Since performing limited changes in reactivity around criticality is a standard process in operating a reactor, no additional or specific equipment is generally needed to perform standard reactor kinetics exercises.

Some kind of neutron source to supply neutrons to be multiplied in the core and start up the reactor needs to be available. This can either be a removable source such as an Am–Be source placed in the core, or a source inherent to the core, for example, when gamma rays emitted from the fuel interact with beryllium elements through a (γ,n) reaction to produce neutrons. The reactor should be in a basic state for reactor operation at zero or low power, i.e., without any feedback effect (e.g. stable temperature and absence of poisons). One of the easiest way of modifying the state of the reactor is to move one of its control rods, using the calibration curve of the rod to adjust the change in reactivity to the planned value.

Standard neutron detection systems used for reactor operation and control can be used to follow the change in neutron density or related reactor power. A detection system capable of recording and plotting the time-dependent neutron density is advised to easily observe the reactor's response to reactivity changes.

To study and compare reactor kinetics without and with an external neutron source, the neutron source should be removable from the core. In some reactors this is not possible either because of technical or administrative reasons. In this case only a partial study of reactor kinetics can be conducted.

To perform advanced exercises, specific equipment or specific reactor features are needed. For example, for the study of reactor response to small reactivity transients, additional systems can be implemented in the core. These can consist of a tube inserted in the core, in which an absorber or fuel sample oscillates using a mechanical drive. Another example concerns the study of the pulse mode of TRIGA reactors. These reactors have been designed with a specific moderator (ZrH) that exhibits a fast and strong temperature feedback effect (see Section 2.4 and Ref. [17]). Such a reactor can be equipped with absorbing rod(s) that can be ejected from the core by the means of a pneumatic system. Ejection makes the reactor supercritical by prompt neutrons alone ($k_{eff} > \beta_{eff}$), and induces a very fast increase (within tens of milliseconds) of the power to thousands of MW. The temperature feedback effect ensures an immediate decrease of core reactivity within tens of milliseconds, and the reactor becomes strongly subcritical. The overall power transient has a typical duration of 30–50 ms. This operating sequence, referred as the pulse mode, can be used to observe and study prompt neutron supercritical excursions.

7.4.3. Methodology

The method to conduct exercises using one of the control rods to change reactor reactivity is given hereafter. Starting from criticality at low power P_0 (no feedback effect) with the control rod at its critical position (H_0) and the neutron source removed from the core, it is possible to perform the following steps:

(1) Observe the constant value of reactor power (i.e., as measured by the neutron detection system signal) related to the critical state ($k_{eff} = 1$), the reactor controlled either manually or through the automatic control system, when available;
(2) Observe closely small fluctuations in reactor power to demonstrate that the mean power is constant but undergoes some small fluctuations related to the statistical character of the fission reactions;
(3) Insert a rod by a few millimetres to make the reactor slightly subcritical ($k_{eff} < 1$), and observe the exponential decrease in power;

(4) Go back to criticality before withdrawing the control rod by a few millimetres to make the reactor slightly supercritical ($k_{eff} > 1$), and observe the exponential increase in reactor power;
(5) Withdraw a rod to an upper position to increase the power with a given doubling time, 30 s, e.g., and check the consistency between the expected (from the in-hour curve) and measured doubling time;
(6) Stabilize the reactor at a power 10 times larger than the previous critical state but without feedback effects. This shows that the critical position of the rod does not depend on reactor power, as long as there is no feedback effect;
(7) Insert a control rod to reduce reactor power back to its initial value P_0, and keep the reactor in a critical state;
(8) Insert the external neutron source in the core; one can observe a very slow and linear increase of reactor power due to the constant supply of neutrons from the source. To clearly observe this effect, the power should be very low, typically less than 100 mW;
(9) Insert the control rod to make the reactor slightly subcritical, and observe the exponential decrease in power;
(10) Bring the reactor back to criticality;
(11) Withdraw a control rod to make the reactor slightly supercritical, and observe the exponential increase in power.

An additional study of kinetics can include the observation of a prompt jump, when the reactor is brought to a supercritical state by a fast insertion of significant reactivity. The measured height of the prompt jump can be compared to a value calculated with the kinetics equations.

The role of delayed neutrons can also be shown during a reactivity transient. Starting from a critical state, reactivity is successively decreased from 0 to -200 pcm by the insertion of one control rod and immediately increased from -200 to 0 pcm by withdrawal in a short period of time, i.e., approximately the mean precursor lifetime (~ 11 s). During the second part of the transient one can observe an increase of power related to the supply of delayed neutrons generated from the decay of precursors produced before the transient. This exercise is evidence of the important role of the precursors and associated delayed neutrons. It can be used to discuss the role of the delayed neutrons in maintaining the reactor critical. Indeed, when $k_{eff} = 1$, the reactor is subcritical by prompt neutrons, but the loss of prompt neutrons is compensated by the delayed neutrons supplied by the decay of the precursors. From Eq. (28) and with the assumption that there is only one group of precursors, when the reactor is critical, the neutron density n is linked to the precursor concentration c by the following equation showing that the level at which n can be stabilized depends on the quantity of the available precursors c.

$$n = \frac{\Lambda\lambda}{\beta_{eff}} c \tag{28}$$

Additional exercises can be conducted by performing periodic variations of reactivity to study the corresponding reactor response, as shown in Fig. 9. Such exercises can be used to indicate the important contribution of delayed neutrons to the total number. As can be seen on the curve in the right panel of Fig. 9, short period supercritical sequences increase the overall neutron density, but their duration is too short to increase the precursor concentration to a sufficient level to maintain a constant power when the reactor is brought back to criticality.

7.4.4. Safety considerations

The safety considerations given in Section 3.5.4 apply. This Section describes additional safety considerations specific to this exercise.

Changes in reactivity by means of control rod motion is a routine operation at a reactor, so standard operating procedures apply. These procedures usually include a limitation to the maximum value of reactivity that can be added. This limitation is often linked to standard and minimum values of doubling time (typically down to 20 s at low power reactors), that are used or authorized, respectively, at the facility.

When considering the implementation of additional equipment to perform neutron kinetics exercises, a comprehensive safety analysis should be conducted, referring to Section 3.1.2.7 and to Safety Standards No. SSG-24 [10] 'Safety in the Utilization and Modification of Research Reactors'. A limitation of the amplitude of reactivity change is usually applicable. Implementation of new equipment generally requires a review and authorization by either a safety committee or the regulatory body.

Standard operating procedures should also apply for the insertion and removal of an external neutron source. As indicated in Section 7.2, the external source does not modify reactivity, so that issues related to reactivity change cannot arise from the source.

From a radiation protection point of view, additional risk may be considered when performing exercises in which an external neutron source or devices are moved in or out of the core.

From a security or safeguards points of view, utilization of nuclear material to modify reactivity can be subject to specific rules and restrictions.

7.4.5. Documentation

In order to conduct this exercise, the following documents can be given to the students:

(a) Background: neutron kinetics, including prompt and delayed neutrons, without and with a neutron source;
(b) Schematic with the core configuration including the position and characteristics of the fuel elements, control rods, external neutron source, neutron detectors, specific equipment used for the exercises;
(c) Step by step procedure to complete the task;
(d) Reactor in-hour curve and calibration curve of the control rods to establish or to check for the consistency of the doubling time.

7.4.6. Questions to the students

Following the reactor kinetics exercises, the following set of questions can be addressed to the students:

(1) What are the conditions necessary to observe a linear increase of the neutron density on the reactor?
(2) When increasing the core reactivity rapidly, what are the two successive steps in neutron density increase? Give the name of the first step. How does the neutron density increase during the second step?

(3) The reactor is assumed to be operated at low power, i.e. with no feedback effects. Is the critical state of the reactor, i.e. the critical position of the control rod, dependent on reactor power? Explain your answer.

(4) Referring to kinetics equation Eq. (26) at criticality ($\rho = 0$), what is the variation (increase or decrease) of the population of prompt neutrons during time *dt*? Thus, what is the role of the delayed neutrons in maintaining the total neutron density constant at critical state?

(5) If the reactor power is increased by a factor 10 between two critical states, by what factor is the precursor concentration increased between these two states?

(6) Explain the reason for the inertia of the reactor related to neutron concentration changes when changing rapidly the core reactivity.

Further evaluation of the impact of the experiment can be obtained through a deliverable such as a report or a presentation on the objectives, methodology and results obtained from these measurements, prepared by the students and delivered to the teacher.

For further information on the reactor kinetics exercises, please consult the bibliography.

8. REACTIVITY CONTROL

8.1. BACKGROUND

Safe RR operation requires a comprehensive understanding of time-dependent reactor kinetics and/or dynamics[16] including its dependence on specific characteristics related to reactor design and reactor operating modes. Reactivity is the main parameter driving the reactor kinetics. Its absolute value and time variation have an immediate influence on reactor operation and a major impact on safety. Research reactors are excellent tools for studying reactor behaviour because of their high flexibility of operation and reactivity measurement exercises are very frequent and popular exercises at RRs.

Several operational reactor parameters related to the reactivity are routinely determined and checked. Those parameters include, for instance, maximal reactivity excess, control rod reactivity worth, shutdown margin, reactor sub-criticality, reactivity changes caused by insertion or removal of a fuel element and reactivity changes caused by an experimental device or sample inserted into or taken out from the core. Among those parameters, the most important are the control rod reactivity worth and reactivity calibration, the excess reactivity and the shutdown margin and the reactivity effect of devices in the core.

8.2. THEORY

The reactivity concept is representative of the reactor's deviation from criticality. The following simple consideration leads to the definition of reactivity. If the number of neutrons in the core at a given time is n_0, then in the next generation the number of neutron will be $n_0 k_{eff}$; where k_{eff} is the effective multiplication factor. The gain or loss in neutron population is $n_0 k_{eff} - n_0$. The reactivity ρ is the fractional change in neutron population per neutron generation, as follows:

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \tag{29}$$

If the reactor is in its subcritical state, $k_{eff} < 1$ and the reactivity is negative; if the reactor is in a supercritical state, $k_{eff} > 1$ and the reactivity is positive; and if the reactor is in the critical state, $k_{eff} = 1$ and the reactivity is $\rho = 0$.

The reactivity is a dimensionless parameter, as is evident from Eq. (29). During routine reactor operation, the real value of reactivity is a small number close to zero. It is not normally used as an operational parameter, and several reactivity unit's systems are used.

In order to keep a consistent dimensionless system of the reactivity units, [$\Delta k/k$] is often used. Alternative units for reactivity are [$\%\Delta k/k$] and [pcm] which are defined as:

$$1\ \% \frac{\Delta k}{k} = 0.01 \frac{\Delta k}{k} \qquad \text{and} \qquad 1\ pcm = 10^{-5} \frac{\Delta k}{k} \tag{30}$$

[16] Both terms 'kinetics' and 'dynamics' are used for time-dependent reactor behaviour in various references, and both are often used as synonyms. In some references, the term 'kinetics' is used for time-dependent reactor behaviour without feedback, and 'dynamics' is used for time-dependent reactor behaviour with feedback. Reactor feedback experiments are described in Sections 9.4 and 9.5.

Another widely used units of reactivity are the *effective delayed neutron fraction* [β_{eff}], the *dollar sign* [\$], and the *cent sign* [¢] and are defined as follows:

$$\rho = \frac{k_{eff} - 1}{\beta_{eff} k_{eff}} \quad [\beta_{eff}]\, or\, [\$] \qquad \rho = \frac{k_{eff} - 1}{100\beta_{eff} k_{eff}} \quad [¢] \tag{31}$$

From the previous definition, it is evident that units [β_{eff}] and [\$] are identical but depending on the country [β_{eff}] or [\$] is used. The advantage of the reactivity units defined in Eq. (31) is that for a reactor reaching prompt criticality the reactivity is close to 1 \$. A further advantage of using these units for reactivity measurements (such as source jerk or rod drop exercises) is that it gives the results directly in units β_{eff} and \$ and not in dimensionless reactivity values. There are some disadvantages when using these units, mainly due to the difficulties in measuring the reactor β_{eff} and to β_{eff} changes during the fuel cycle at reactors with a significant burnup. In principle, β_{eff} is different for each reactor.

Control rods, in their reactivity control function, are the main means to manage the state of the reactor, including to start-up and shut down the reactor and to modify the power level as the planed operation requires. In their safety function, control rods are used to perform a fast and safe shutdown of the reactor when needed. In order to fulfil all those functions, a RR is usually equipped with three types of control rods:

(a) *Safety control rods* are used as a reservoir of negative reactivity for the shutting down of a RR. The safety control rods are usually fully withdrawn from the core during normal operation and are ready to immediately drop down into the core resulting in a quick shutdown of a reactor. Generally, RRs have more than one safety control rod in order to comply with the redundancy principle required to the shutdown system on the basis of nuclear safety at the reactor. At some RRs, safety control rods can be used for other purposes, e.g. in TRIGA reactors one safety control rod, used for a power pulse experiment, is called a pulse rod;
(b) *Shimming control rods* are used for compensation of slow reactivity changes in a RR caused, among others, by fuel burnup, reactor poisoning or installing/removing of experimental devices into the core. In some RRs, shimming rods are also known as coarse rods, compensation rods or experimental rods. At some RRs, shimming control rods also function as safety control rods;
(c) *Regulating control rods* are used in normal operation for fine control of the reactor power, as well as for compensation of occasional minor reactivity changes. In the automatic power control mode, the regulating control rod serves to the reactor control system as the actuator to implement the desired control strategy, i.e. maintaining a constant power level.

Most RRs are equipped with control rods with the same construction and the same absorbing material (boron, cadmium, and hafnium are mostly used) and the only difference is their function, i.e. safety or shimming control rods. In the case of the regulating control rod, depending on the reactor design, it may be identical to the others or some characteristics may change, as dimension or material, in order to decrease its reactivity worth. Also, the relative position at which it is installed in the core may be used in order to manage the reactivity worth of the regulating control rod.

The amount of positive or negative reactivity inserted in the reactor core during the change of the control rod position (% of insertion in the core), basically depends on three factors:

(a) Rod location in the core (i.e. in the centre or at the core periphery);
(b) Amplitude of the rod position change (i.e. on Δh);
(c) Relative position to the core of the portion of control rod inserted or removed, i.e. in the bottom, centre or top part of the core.

Perturbation theory can be used to model the impact on reactivity of the change in the control rod position. In a 1-D approach, using cylindrical symmetry with respect to the direction of the rod movement, the relationship between reactivity and rod position can be expressed as:

$$\rho(x) = K\int_{-\frac{H}{2}}^{x} \cos^2\frac{\pi z}{H}dz = \frac{K}{2}\left(x + \frac{H}{2} + \frac{H}{2\pi}\sin\frac{2\pi x}{H}\right) \tag{32}$$

where
$\rho(x)$: reactivity worth of the control rod when located in the x position
x: distance between the control rod bottom end and the lower edge of the core
H: core height
K: constant

The constant K can be determined from the boundary conditions:

$$\rho\left[-\frac{H}{2}\right]=0 \quad \text{and} \quad \rho\left[\frac{H}{2}\right]=\rho_0 \tag{33}$$

where ρ_0 is the control rod reactivity worth. Using Eq. (33), Eq. (32) can be rewritten as:

$$\rho(x)=\left(\frac{x}{H}+\frac{1}{2}+\frac{1}{2\pi}\sin\frac{2\pi x}{H}\right)\rho_o \tag{34}$$

Eq. (34) shows that the relation between reactivity and control rod position does not have a linear behaviour, but a complex function giving rise to a so called *S-shape* curve.

The graphical representation of Eq. (34) for a typical RR is shown in Fig. 10 and is called the calibration curve of a control rod. From Eq. (34), it is evident that all control rods in the reactor exhibit, in this model, *S-shape calibration curves*. In a real reactor the *S-shape* curve can be slightly deformed due to inhomogeneities in the core, in particular due to the higher burnup in the lower half of the core. The control rod worth, ρ_0, depends on the amount and the type of absorbing material and also depends on a control rod location in the core. If the control rod is located in the centre of the core, then ρ_0 is generally larger than in the periphery of the core. In general, the control rod reactivity worth is very sensitive to thermal neutron flux, and thus, ρ_0 is larger when the rod is located in an area in the core with higher thermal neutron flux and it decreases when it is installed in an area with lower thermal neutron flux.

Figure 10 represents the integral form of the calibration curve. The differential form of the calibration curve is also often used to represent the reactivity worth of a control rod. Figure 11 is an example of this type of curve. Both integral and differential forms of calibration curves are widely used at low power RRs.

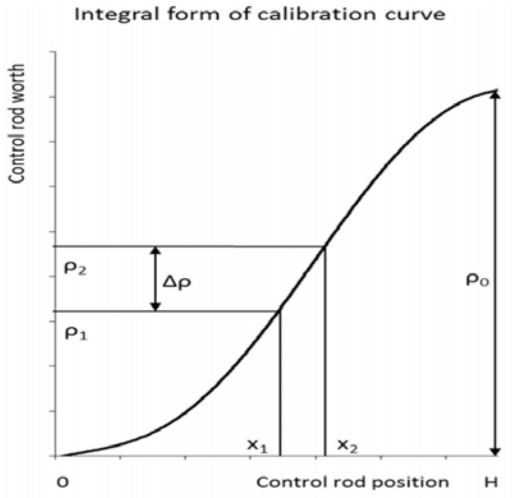


FIG. 10. Integral form of the control rod reactivity worth calibration curve. (reproduced from [16] with permission courtesy of the Czech Technical University in Prague, Czech Republic)

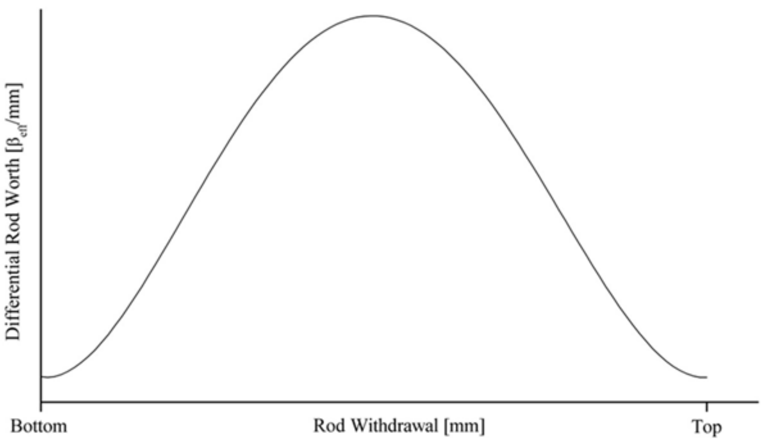


FIG. 11. Differential form of the control rod reactivity worth calibration curve. (reproduced from [16] with permission courtesy of the Czech Technical University in Prague, Czech Republic)

Determination of the operational core safety parameters of a RR is usually connected with the control rod calibration exercises. The main reason for carrying out these exercises is to operate and shut down the reactor within the safety limits provided in the OLCs, which are analysed in the SAR. The most important and mostly used parameters are the excess reactivity and the shutdown margin.

Any RR, for its operation and utilization, needs a way of inserting an extra reactivity, which is available in the core, to compensate effects related to reactor operation. This includes e.g. fuel burnup, xenon poisoning, insertion/extraction of samples into the core for irradiation, change in the position of neutron detectors. This extra reactivity, which is 'built in the core', is called excess reactivity. It is always kept limited and under strict control. If the excess reactivity is too small, some exercises or activities cannot be carried out because of a lack of excess reactivity available in the core to keep the reactor in its critical state. On the other hand, if the excess reactivity is large, there is a danger of a power trip or prompt criticality if something unexpected happens in the core. In order to prevent these situations, the maximal excess reactivity can be limited to the amount which does not allow for a reactor to become prompt critical.

The excess reactivity can be easily calculated from the following formula:

$$ER = \sum_{i=1}^{N} \int_{criticality}^{top} \frac{d\rho_i}{dx} dx \tag{35}$$

where

ER: *excess reactivity*

ρ_i: reactivity of *i*-th control rod in *x* position

N: number of control rods in the reactor core

The shutdown margin ensures that the reactor shutdown is safe and can be performed at all reactor states and under all conditions which can occur during the operation. The ability to safely shut down the reactor and to keep it in a shutdown state is basically given by the drop of all the control rods related to the shutdown system into the core which causes the fission chain reaction to stop.

The shutdown margin can be easily calculated from the following formula:

$$SDM = \sum_{i=1}^{N} \int_{bottom}^{criticality} \frac{d\rho_i}{dx} dx - \int_{room\ temperature}^{actual\ temperature} \frac{d\rho_M}{dt} dt - \int_{room\ temperature}^{actual\ temperature} \frac{d\rho_F}{dt} dt - \int_{0}^{actual\ power} \frac{d\rho_P}{dP} dP \tag{36}$$

where

SDM: shutdown margin

N: number of control rods to be inserted in the reactor core

ρ_i: reactivity of *i*-th control rod in *x* position

ρ_M: reactivity introduced in the core by the fuel temperature coefficient

ρ_F: reactivity introduced in the core by the moderator temperature coefficient

ρ_P: reactivity introduced in the core by the power coefficient, i.e. Xe

In many RRs a single failure (rod stuck) of the control rod with the largest rod worth ρ_{imax} during the shutdown is considered, and the shutdown margin is determined as:

$$SDM = \sum_{i=1}^{N} \int_{bottom}^{criticality} \frac{d\rho_i}{dx} dx - \int_{bottom}^{criticality} \frac{d\rho_{i\max}}{dx} dx - \int_{room\ temperature}^{actual\ temperature} \frac{d\rho_M}{dt} dt - \int_{room\ temperature}^{actual\ temperature} \frac{d\rho_F}{dt} dt - \int_{0}^{actual\ power} \frac{d\rho_P}{dP} dP \tag{37}$$

In a reactor at zero power, during normal operation, reactivity coefficients are negligible and thus, the three terms on the right hand side of the Eq. (37) are discarded and only the first negative term remains. Therefore, the shutdown margin for zero power RRs is determined as:

$$SDM = \sum_{i=1}^{N} \int_{bottom}^{criticality} \frac{d\rho_i}{dx} dx - \int_{bottom}^{criticality} \frac{d\rho_{i\max}}{dx} dx \tag{38}$$

Dozens of more or less accurate methods were developed to measure the reactivity. They can be classified from several points of view, i.e. according to the reactor state (critical, subcritical and supercritical), the type of approach (deterministic or statistic), the technique used (static or dynamic) among others. At low power RRs the following methods can be used: positive period (or the asymptotic period) method, source jerk method, rod drop method, control rod oscillator method, neutron source multiplication (or Greenspan) method, Rossi-α noise analysis method, Feynman-α noise analysis method, pulsed neutron source method, fuel–poison substitution method, criticality determination method and inverse rate method. In present days, various digital reactivity meters based on the inverse kinetics method are also used at RRs.

Because of the close connection between reactivity measurement and control rod exercises, almost all methods of measuring reactivity can be used for control rod calibration or for determination of the operational core safety parameters. Several other methods have been also developed specifically for control rod calibration at the low power RRs, including the doubling time, inverse rate, intercalibration (or control rod swap) and control rod insertion (or dynamic calibration) methods. The reactivity meters can be used for control rod calibration. In principle, any method of reactivity measurement to measure the shutdown margin and excess of reactivity can be used.

8.3. EDUCATIONAL ASPECTS

Reactivity related exercises belongs to the group of reactor physics exercises. The learning objective, information on the type of students and the level of the exercises are the same as described in Section 6.3.

Usually, no specific experimental instrumentation is needed for carrying out the reactivity exercises. A RR and its standard technology and an appropriate neutron detection system are necessary to carry out the exercise.

Some more elaborated reactivity exercises, such as Rossi-α noise analysis method, Feynman-α noise analysis method, pulsed–neutron source method or dynamic calibration control rod method are more suitable for nuclear engineering doctoral students and for students with a special interest in master's degree programmes, such as future reactor physicists or nuclear safety experts. In these cases, the level of the exercises is advanced. Usually specific reactor features or specific experimental instrumentation are needed for carrying out these advanced reactivity exercises.

Depending on the level of the exercise, the students should have a prior minimum background on the reactor principle, the functions of the moderator and reflector (see Section 9.2 for the moderation to fuel factor), the definition of the multiplication factor and reactivity (Sections 6.2 and 8.2) and the neutron flux distribution. The characteristics of the control rods and how to use their calibration curves should also be known. The background specific to each exercise should be summarised for each exercise. In the following, three types of exercises are described:

(a) Control rod calibration;
(b) Influence of core components to reactivity;
(c) Safety parameters related to core reactivity.

8.3.1. Control rod calibration

This exercise is appropriate for bachelor's, master's and doctoral programmes. This exercise is suitable for students studying nuclear engineering as the major curriculum as well as for students studying various major engineering curricula. The level of the exercise can be adjusted according to the student's background and pedagogical objectives. It goes from basic to intermediate level.

More advanced reactivity experiments (see Section 8.3) can also be carried out for nuclear engineering doctoral students and for students developing a specific expertise. This particular case is not covered within this guideline.

For the basic level, a method for the control rod reactivity calibration and reactivity worth measurement can be learned and understood. As an example, the method used to establish the calibration curve of a rod through the measurement of the doubling time for different position of the rod can be explained and used (see Section 8.4). Later on, the total reactivity worth of the rod can be measured by comparing two critical configurations of the rods, one with the rod fully inserted and the other one with the rod fully withdrawn. Duration of such an exercise is typically 1 to 2 hours.

For the intermediate level, starting from content of the basic level, additional exercises can be conducted. The methods for rod calibration can be explained and used (such as the swap , inverse rate and rod-drop methods). The *S-shape* of the calibration curve can be mathematically explained (see Eq. (34)). Finally, the rod drop technique can be used to measure the total reactivity worth of the control rod. Duration of each additional exercise is typically 30 min.

8.3.2. Influence of core components to reactivity

This exercise is appropriate for bachelor's, master's and doctoral programmes. This exercise is suitable for students studying nuclear engineering as the major curriculum as well as for students studying various major engineering curricula. The level of the exercise can be adjusted according to the student's background and pedagogical objectives. It goes from basic to intermediate level.

More advanced reactivity experiments can also be carried out for nuclear engineering doctoral students and for students developing a specific expertise. It can include for example the comparison between core calculations and practical measurements of the reactivity change. This particular case is not covered within this guideline.

The three exercises described in this guideline can be conducted for both basic and intermediate levels, since these experiences and their interpretation are relatively easy. When going from basic to advanced level it is possible to go more deeply into the theory or the safety issues related to the reactivity changes. The duration of each exercise is typically one to one and a half hour.

8.3.3. Safety parameters related to core reactivity

This exercise is appropriate for bachelor's, master's and doctoral programmes. As indicated in the monographic part, this exercise is suitable for students studying nuclear engineering as the major curriculum as well as for students studying various major engineering curricula. The level of the exercise is basic to intermediate level.

For the basic level, the ER and SDM can easily be calculated without temperature or poison feedback. Their compliance with the OLCs can be checked. Safety issues related to the ER and SDM can be discussed. Duration of such an exercise is typically one and a half hour.

For the intermediate level, similar exercise can be carried out, but in addition, the reactivity changes related to the temperature and poison effects and their impact on the ER and SDM can be determined and discussed. The overall duration of the exercise is typically three hours.

8.4. EXERCISES ON CONTROL ROD CALIBRATION

8.4.1. Objective of the exercise

The objective of this exercise is to characterise the control rods of the reactor, i.e. to establish the change in the core reactivity related to the change in the control rod position. From a practical point of view, according to the function of the rod, the level of information needed on this reactivity change is different. For safety rods, knowing the total reactivity worth is sufficient, while for shimming and regulating rods knowing the exact change in the core reactivity related to a change in the position of the road is necessary. The latter is referred to the reactivity calibration curve of the control rod.

8.4.2. Equipment and conditions

To establish the reactivity calibration curve or to measure the total reactivity worth of a control rod, the reactor should be in operation at low power, i.e. without any feedback effect (e.g. stable temperature and absence of poisons) modifying the reactor state. This will guaranty that the change in the core reactivity is only related to the rods and not to some additional effect. The in-hour curve of the reactor is necessary for the calibration of the control rod through the doubling time measurement technique.

The standard neutron detection systems used for the reactor control system can generally be used to follow the neutron density (or associated reactor power) and to measure the doubling time. Thus, no additional equipment is needed.

8.4.3. Methodology

This Section explains the way of carrying the exercise using three different methods. It can be completed by the use of other techniques.

8.4.3.1.Establishment of the calibration curve with doubling time measurement

For this exercise, the control rod to be calibrated is fully inserted in the core (position $z = 0$) and the reactor is set critical at a low power P_0 (no feedback effect). The rod is moved to a given position z_1 and the doubling time T_{d1} is measured during the reactor divergence. Using the in-hour curve the T_{d1} value is used to establish the reactivity of the core $\Delta\rho_1$ for the position z_1.

Before carrying out the next measurement, it is advised to lower the rod back to 0 and to make the reactor subcritical for a while in order to reduce the power back to P_0. This is a way to avoid significant increase of the power, and associated potential feedback, when conducting the successive divergences.

The rod is moved to a given position z_2 ($z_2 > z_1$) and doubling time T_{d2} ($T_{d2} < T_{d1}$) is measured during the reactor divergence. Using the in-hour curve the T_{d2} value is used to establish the reactivity $\Delta\rho_2$ of the core for the position z_2.

Such a technique can be used to plot the integral curve giving the efficiency of the rod, i.e. the measured $\Delta\rho$ value, as a function of the rod position z. However, in practice, the doubling time is limited to reasonable values, i.e. typically T_d greater than 20 s which correspond to reactivities lower than 150 pcm. Thus, since the total worth of a rod is generally larger than 500 pcm it is not possible to continue with the experiment withdrawing the rod up to its upper position without compensating the withdrawal of the rod to be calibrated with the insertion of another rod.

In practice, each time the doubling time becomes short (close to 20 s, for example), for a given $\Delta\rho'$ value, a new critical configuration of the rods will have to be reached by keeping the rod to be calibrated to its last position z' and by moving another rod down to find a new critical configuration.

This new critical state will be used to further extract the control rod to be calibrated and to measure the $\Delta\rho''$ value corresponding to the extraction of the rod from z' to a higher position z''. In this case, the overall reactivity change resulting from the withdrawal of the rod from 0 to z'' will be equal to $\Delta\rho' + \Delta\rho''$. Compensation of the withdrawal of the rod to be calibrated by another rod will have to be done as often as necessary to maintain the doubling time within safe and authorised values.

It has to be pointed out that the control rod used to compensate for the withdrawal of the rod to be characterised shouldn't modify significantly the worth of the rod to be calibrated. Indeed, the worth of a control rod depends not only on its characteristics but also on the configuration of the core. If a rod A is inserted close to the rod to be calibrated B, the rod A will reduce the reactivity worth of the rod B by reducing the local neutron flux where the rod B is inserted. This is known as the shadowing effect. It is then advised to compensate with a rod as far as possible of the rod to be calibrated. If necessary and possible, more than one rod can be used for the compensation.

Table 6 and Fig. 12 illustrate the resulting data and calibration curve that can be obtained by this technique. The shape of the curve, which is related to the neutron flux distribution in the core, can be discussed with the students.

TABLE 6. THE PRINCIPLE OF THE PLOT OF THE CALIBRATION CURVE BY THE DOUBLING TIME MEASUREMENT TECHNIQUE

Position of the rod to be calibrated	Position of the compensating rod	Measured doubling time	Change in core reactivity	Efficiency of the rod	Comment
0	z_A	∞	0	0	
z_1	z_A	T_{d1}	$\Delta\rho_1$ (0 to z_1)	$\Delta\rho_1$	
z_2	z_A	T_{d2}	$\Delta\rho_2$ (0 to z_2)	$\Delta\rho_2$	
z_3	z_A	T_{d3}	$\Delta\rho_3$ (0 to z_3)	$\Delta\rho_3$	Doubling time becoming short
z_3	$z_B < z_A$	∞	0	$\Delta\rho_3$	New critical state
z_4	z_B	T_{d4}	$\Delta\rho_4$ (z_3 to z_4)	$\Delta\rho_3+\Delta\rho_4$	
z_5	z_B	T_{d5}	$\Delta\rho_5$ (z_3 to z_5)	$\Delta\rho_3+\Delta\rho_5$	Doubling time becoming short
z_5	$z_C < z_B$	∞	0	$\Delta\rho_3+\Delta\rho_5$	New critical state
z_6	z_C	T_{d6}	$\Delta\rho_6$ (z_5 to z_6)	$\Delta\rho_3+\Delta\rho_5+\Delta\rho_6$	
…					

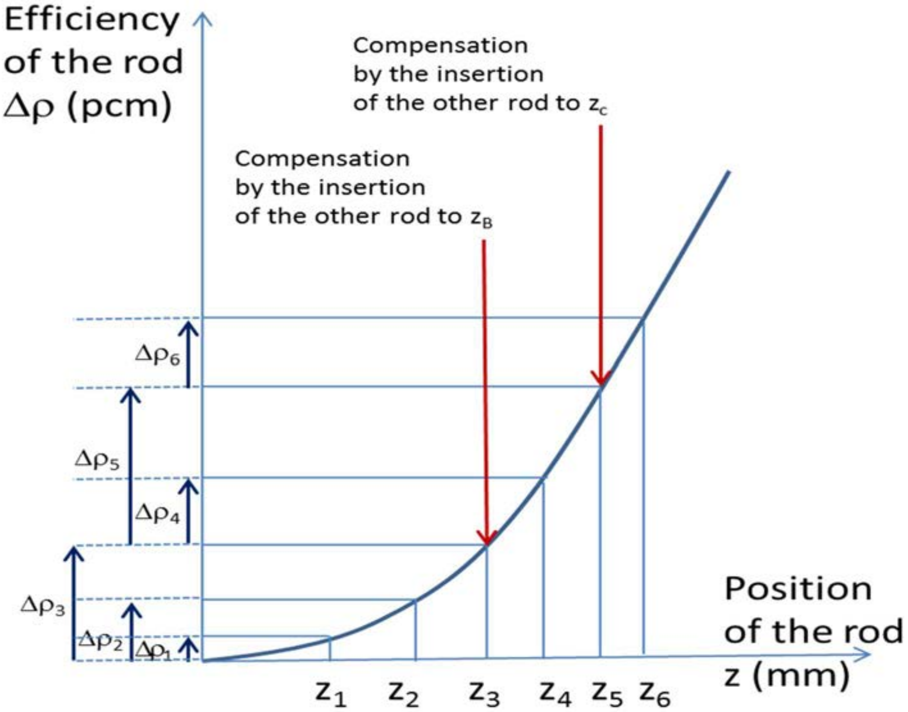


FIG. 12. The principle of the plot of the calibration curve by the doubling time measurement technique. (courtesy of the National Institute for Nuclear Science and Technology, CEA Saclay, France)

8.4.3.2. Total reactivity worth measurement by the comparison of two critical states

For this exercise, the control rod whose total worth is to be measured is successively placed in two different positions: fully inserted and fully extracted. For both conditions, the critical configuration of the rods is recorded. In practice, when achievable, a single control rod can be moved down to compensate for the extraction of the rod to be measured. Taking the calibration curves of the rods that were used to compensate for the extraction of the rod, and the differences between the two configurations, it is possible to establish the total worth of the rod being calibrated.

8.4.3.3. Control rod calibration by the inverse rate method

In this technique, the reactor is in a subcritical state during the whole calibration process. The inverse rate method is based on subcritical multiplication with a neutron source S (emission rate of the source) in the core. Starting from the equations of kinetics, when the neutron density n is stable in a subcritical state, k_{eff} can be written as:

$$k_{eff} = 1 - \frac{S}{n} \tag{39}$$

Taking into account the integral form of the calibration curve shown in Fig. 10 in Section 8.2, the reactivity change as a function of control rod position z can be expressed as:

$$\Delta\rho(z) = \rho_0 \frac{\rho(z) - \rho_{\downarrow}}{\rho_{\uparrow} - \rho_{\downarrow}} \tag{40}$$

where ρ_0 is the total worth of the control rod, while $\rho\uparrow$ and $\rho\downarrow$ are the core reactivities when the control rod is respectively fully extracted and fully inserted.

From the definition of ρ in Eq. (29) and Eq. (40), Eq. (40) can be expressed in the form:

$$\Delta\rho(z) = \rho_0 \frac{\frac{1}{n_{\downarrow}} - \frac{1}{n(z)}}{\frac{1}{n_{\downarrow}} - \frac{1}{n_{\uparrow}}} \frac{k_{eff\uparrow}}{k_{eff}(z)} \tag{41}$$

where *n(z)* is the neutron density at position *z* of the control rod, while $n\downarrow$ and $n\uparrow$ are the neutron densities when the control rod is respectively fully extracted and fully inserted.

Since the ratio $\frac{k_{eff\uparrow}}{k_{eff}(z)}$ is close to 1, the reactivity change can be determined from:

$$\Delta\rho(z) = \rho_0 \frac{\frac{1}{n_{\downarrow}} - \frac{1}{n(z)}}{\frac{1}{n_{\downarrow}} - \frac{1}{n_{\uparrow}}} \tag{42}$$

In practice, the counting rate given by neutron detectors, which is proportional to neutron density *n*, will be assumed to give the *n* values. The value of ρ_0, i.e. the total worth of the control rod being calibrated, has to be established by another technique such as the one described in the previous exercise.

The calibration is carried out according to the following procedure:

(a) Reactor is in its subcritical state (with neutron source in the core), with the control rod being calibrated fully inserted;
(b) Neutron detection system (from the control system or additional one) is used to measure counting rate corresponding to $n\downarrow$;
(c) Control rod is withdrawn step by step from the bottom to the top of the core with adequate increments (1/10 of the total motion of the rod for example);
(d) At each position of the rod, after the neutron density has reached the equilibrium, the counting rate $n(z)$ is recorded;
(e) When the rod is fully withdrawn, the $n\uparrow$ value is recorded;
(f) The calibration curve can then be plotted using Eq. (42), the value of ρ_0 and the values recorded for $n(z)$ at each position;
(g) Similarly to the first exercise, the shape of the curve, which is related to the neutron flux distribution in the core, can be discussed.

8.4.4. Safety considerations

The safety considerations given in Section 3.5.4 apply. This Section describes additional safety considerations specific to this exercise.

Usual procedures for reactor operation, such as those related to the change in the core reactivity through the motion of the control rods, are applied. Care should be taken in the control of the reactivity and in maintaining the doubling time to the specified values.

From a radiation protection point of view, this exercise should not bring additional risk compared to normal reactor operation.

8.4.5. Documentation

In order to conduct this exercise, the following documents can be given to the students:

(a) Background: neutron kinetics in subcritical state with a source, neutron flux distribution in the core;
(b) Schematic of the core with the control rods and neutron detectors;
(c) In-hour curve, calibration curve of the rod used for the compensation of the motion of the rod to be characterized;
(d) Step by step procedure to complete the task, including details on how to perform the measurement: wait for the neutron density to reach equilibrium, as in the exercise described in Section 6.4, take an average value of n;
(e) Graph paper or software application to plot the curve;
(f) Specific experimental related information and rules: rules related to the overall configuration of the rods may apply for example.

8.4.6. Questions to the students

A first evaluation of the impact of the experiment can be obtained using the following set of questions:

(1) Explain different functions of safety rods compared to shimming and regulating rods.
(2) Based on your answer to (1), what are the characteristics, relevant to its function, that are required to be known for the safety rods, the shimming rods and the regulating rods?
(3) Is it possible to establish the whole control rod calibration curve (with a total worth of 1000 pcm for example) by withdrawing it successively and measuring the corresponding doubling time? Explain why. What should be done to plot the whole curve?
(4) Explain the shape of calibration curve. Why it is not linear?
(5) When using one control rod for reactor regulation (manual or automatic) is it better to have this control rod in the middle part of the core or in the bottom (or top) part of the core? Explain why.
(6) Is the control rod reactivity worth depending only on the characteristics of the rod or is it also depending on the whole configuration of the core (position of other control rods for example)?

Further evaluation of the impact of the experiment can be obtained through a deliverable such as a report or a presentation on the objectives, methodology and results obtained from these measurements, prepared by the students and delivered to the teacher.

8.5. INFLUENCE OF CORE COMPONENTS TO REACTIVITY

8.5.1. Objective

As indicated in Section 8.2, reactivity measurement experiments are very frequent and popular experiments at RRs. Indeed, it is important to understand and to control the impact of the reactor parameters on the core reactivity. This guideline is dedicated to the study of the influence of three of these parameters, the fuel element position and characteristics, reflecting elements and insertion or removal of experimental devices. The safety issues related to these parameters are also of particular interest in this type of exercise.

The objective of this exercise is generally to characterise the reactivity of a reactor in its different states in order to establish reactivity change related to a change of a single reactor parameter. The reactivity change can then be discussed based on the theory.

8.5.2. Equipment and conditions

Carrying out the exercises described in this guideline implies the ability to unload, load or move fuel elements, reflecting elements and experimental devices in the core or its vicinity. To ensure that reactivity changes to be measured are only related to the manipulation of elements or devices, the reactor should be operated at low power, i.e. without any feedback effect e.g. stable temperature and absence of poisons) that could modify the reactor state.

The change in the reactivity between two reactor states is measured through the difference in the critical configuration (position) of control rods. The standard I&C systems can be used to check for the critical state, by means of the neutron measuring channels and to record the position of the control rods using the reactor control system.

To study the effect of fuel on reactivity, standard fuel elements are used. To study the reactivity effect of reflector, reflecting devices (graphite or beryllium) can be placed in the vicinity of the core. To study the effect of experimental devices on reactivity, such devices (containing uranium, graphite or cadmium among other options) can be introduced in the core or its vicinity.

8.5.3. Methodology

8.5.3.1.Effects on reactivity of the position of fuel elements in the core

For this exercise the state of a reactor is changed by unloading one fuel element from the core. Then the corresponding change in reactivity is measured. This is carried out by comparison of two critical states, before and after unloading the fuel element.

If achievable, the reactivity worth of fuel elements with similar characteristics (same quantity of fuel) and placed on different positions in the core should be measured. This allows to study the relevance of the position of the fuel element in the core, i.e. the neutron flux distribution, on the reactivity worth of the fuel element. In this exercise only one element is unloaded from the core at a time. It is reloaded into the core before the removal of the next fuel element.

Here is a proposed sequence for this exercise:

(1) Before carrying out the measurement, the critical configuration with all fuel elements in place is established (state 1). This is done by operating the reactor at low power (with no feedback effects).
(2) With the reactor in safe shutdown, one fuel element is unloaded from the core (state 2).
(3) The reactor is brought critical again using either the approach to criticality procedure or typical restart (if the core reactivity change is well controlled and known).
(4) The difference in reactivity between state 2 and 1 is determined from the control rod positions using the calibration curve(s) of the control rod(s) that was (were) moved between this two states.
(5) With the reactor in safe shutdown, the fuel element is reloaded into the core.

If reactivity worth measurement is conducted for each fuel element, then this sequence is repeated starting from state 2 through 5 and a table such as Table 7 can be completed.

TABLE 7. EXAMPLE TABLE OF CONTROL RODS POSITIONS TO MAKE THE REACTOR CRITICAL UNDER DIFFERENT CORE CONFIGURATIONS

State	Position Control rod 1 (mm)	Position Control rod *i* (mm)	Change in reactivity (pcm)
Initial state with all fuel elements			
Fuel in position 1 removed			
Fuel in position 2 removed			
Fuel in position 3 removed			
…			

It is important to recall the students that if precise information on the reactivity worth of the fuel element is not available, then the foreseen critical configuration of the control rods is also unknown; in that case, an approach to critical experiment is conducted after any change of the core configuration.

When possible, the position of only one control rod should be used to change the core reactivity. This will make the change in core reactivity easier to visualize and to calculate. To avoid perturbation of neutron flux distribution in the zone of the core where the fuel elements are moved (which will in turn modify the reactivity worth of the fuel element), it is advised to compensate for the change in reactivity moving only control rods which are placed far away from the area where the fuels elements are moved.

The reactivity change versus a position of fuel element can be plotted in order to study the impact of the fuel element, i.e. of the neutron flux distribution in the core, relative to the position in the core. A curve with a shape similar to those shown in Fig. 3 in Section 5.2 is expected. The shape of the curve can be explained according to theory and an additional study can be conducted at an advanced level by comparing the calculated shape of neutron flux distribution (using neutronics codes) to the measured experimental shape previously obtained.

If only fuel elements with different characteristics, such as uranium content, can be used, the combined influence of the uranium content and the fuel element location in the core will have to be taken into account. Finally, an alternative exercise could be carried out, loading at the same position elements with different values of the uranium content in order to study the effect of the uranium content on the core reactivity. A curve displaying the reactivity change versus the mass of uranium can be plotted to further analyse this effect.

8.5.3.2.Influence of reflecting elements or devices

In reactors equipped with reflecting elements (graphite, beryllium), the effect of he reflector on reactivity can be studied. Reflecting elements are usually placed at the periphery of the core to reduce loss of neutrons which are leaking out of the core. Unloading one or more of such reflecting elements will allow the observation and analysis of neutron reflection, the effect on the neutron flux density and its impact on the fuel element reactivity worth. Alternatively, reflecting devices can be introduced at the core periphery.
A procedure similar to that described to study the effect of the location of the fuel element in the core can be used. It will result in the determination of the change in reactivity for the removal or introduction of reflecting elements or devices. Comparison between measurement and calculations can also be conducted.

8.5.3.3.Influence of experimental devices

In a RR, experimental devices placed in the core or in its vicinity (channels) can significantly change the core reactivity. An experimental device can either result in an increase of reactivity (insertion of a reflecting material or uranium) or a decrease of reactivity (insertion of an absorbing material or an empty tube that will increase neutron leakage out of the core). Insertion of a reflecting material such as graphite or an absorbing material such as cadmium is common. Insertion of nuclear material is not possible at every RR due to safeguard issues and the facility OLCs. Insertion of an empty tube, equivalent to a beam tube, can result in radiation protection issues due to the removal of biological shielding and the potential increase on the dose rate on the vicinity of the tube.

Starting from the initial reference state different type of devices, i.e. reflecting or absorbing with different quantities of reflecting or absorbing material, can successively be inserted into the core. A procedure similar to that described to study the effect of the location of the fuel element in the core can be used. It will result in the determination of reactivity change due to removal or introduction of experimental devices. This experiment can be complemented with the comparison between the results of the measurement and those obtained from calculations using neutronics codes.

In some cases, the exercise can be conducted while reactor is in operation: the devices are removed one by one while the reactor is maintained in the critical state by modifying the position of the regulating rod. This can be done while the reactor is operated in automatic mode by slowly changing the reactor state. With this technique, reactivity change at each step should be limited typically to 100 pcm for safety reasons. This ensures long doubling times if the reactivity is not compensated.

8.5.4. Safety considerations

The safety considerations given in Section 3.5.4 apply. This Section describes additional safety considerations specific to this exercise.

Before implementing the experiments, conducting a safety analysis may be needed to check that the reactivity cannot vary in an uncontrolled way, both in normal and incidental conditions, as a result of the loading or unloading of the devices.

To reduce the time taken by the exercise the loading or unloading of the devices can be carried out while the reactor is in operation. In this case, a specific safety analysis should be conducted and the reactivity change should generally be limited to a value (typically 100 pcm) ensuring a reasonable value of the doubling time (typically 40 s) if the reactivity was suddenly increased due to incidental conditions. Such a situation can be encountered if a reflecting device is unintentionally dropped into the core while it is extracted from the core.

From a radiation protection point of view, device loading and unloading is usually carried by the reactor operating staff but, if this is allowed by the OLCs and procedures to be done by students, care should be taken to establish a detailed procedure and to ensure a monitoring and follow-up of the radiation protection issues which can be related to handling irradiated (i.e. activated) devices.

8.5.5. Documentation

In order to conduct this exercise, the following documents can be given to the students:

(a) Background: reactor principle including the role of the fuel, reflecting and absorbing devices on the core reactivity, neutron flux distribution in the core;
(b) Schematic of the core with the control rods, neutron detectors and devices to be loaded and unloaded;
(c) Calibration curve(s) of the rod(s) used for the compensation of the motion of the rod to be characterised;
(d) Step by step procedure to complete the task, including the characteristic of the devices that are loaded or unloaded, and the sequence for the device worth measurement;
(e) Graph paper or software application to plot the curves;
(f) Specific experimental related information and rules that may apply to the experiment, related to the limitations of the reactivity change for example.

8.5.6. Questions to the students

A first evaluation of the impact of the experiment can be obtained using the following set of questions:

(1) Is the worth (in pcm) of a fuel element only related to the characteristics of the fuel element or also to the general configuration of a core? Explain your answer. The same question can be applied to reflecting or absorbing devices.
(2) According to theory related to neutron flux distribution in the core, explain the shape of the curve presenting the relation of *reactivity change* versus *location in the core* obtained in the experiment to measure the fuel reactivity worth.
(3) Explain the function of the reflecting elements or devices. What is a positive impact on neutron flux distribution in the core? and the disadvantage when samples are irradiated around the core?
(4) Is it possible to load or unload a device in the core while reactor is in operation? Argue about the safety of the experiment according to the worth of the device. The analysis can be carried out for two different values of reactivity change: 50 and 200 pcm, evaluating normal and incidental conditions while performing the experiment.

Further evaluation of the impact of the experiment can be obtained through a deliverable such as a report or a presentation on the objectives, methodology and results obtained from these measurements, prepared by the students and delivered to the teacher.

8.6. SAFETY PARAMETERS RELATED TO CORE REACTIVITY

8.6.1. Objective

As described in Section 8.2, on the one hand, reactor operation needs the utilization of extra reactivity inserted by the withdrawal of the control rods. On the other hand, safe reactor shutdown needs the availability of sufficient negative reactivity to be inserted in the core to safely shut down the reactor when needed. These reactivity quantities are related, respectively, to the excess reactivity (*ER*) and shutdown margin (*SDM*)[17] of the reactor.

The objective of this exercise is to establish:

(a) Excess reactivity;
(b) Shutdown margin;
(c) Shutdown margin with one control rod stuck.

This exercise can be connected to the total control rod reactivity worth measurement and the control rod reactivity calibration exercises (guideline in Section 8.4) since the reactivity worth of each control rod is needed to establish the actual value of these parameters. After determining these parameters, it is possible to check if they comply with the values present on the OLCs. It is then advised to discuss the safety issues related to the values of the *ER* and *SDM* and to link them to the SAR.

Figure 13 illustrates the definition of *ER* and *SDM* considering only one control rod. The excess reactivity is defined in Eq. (35). *ER* corresponds to the extra reactivity that can be inserted by moving all the control rods from the reactor critical state position to their upper position.

The shutdown margin *SDM* is defined by Eq. (36). For zero power reactors, *SDM* corresponds to the negative reactivity inserted by the drop of all the control rods related to the shutdown system from their position at criticality to their bottom position. For high power reactors with temperature or poison effects, the change in core reactivity related to reactor shutdown needs to be considered (see Eq. (36)). This includes for example a positive reactivity change resulting from the decrease in temperature (both fuel and water) after reactor shutdown. In this case, the long term *SDM* will decrease due to this reactivity change.

Finally, in many RRs the OLCs prescribe that, in the event that the control rod with the largest reactivity worth is fails to drop resulting in the *SDM* with one rod stuck (single failure), the remaining control rods are still sufficient to bring the reactor to subcriticality. In this case, the reactivity related to the drop of this rod from its critical position is subtracted from the *SDM* (see Eq. (37)).

[17] Shutdown margin can refer either to the margin by which the reactor is subcritical when all control rods are inserted or to the margin by which the reactor would be shut down in the event of a SCRAM.

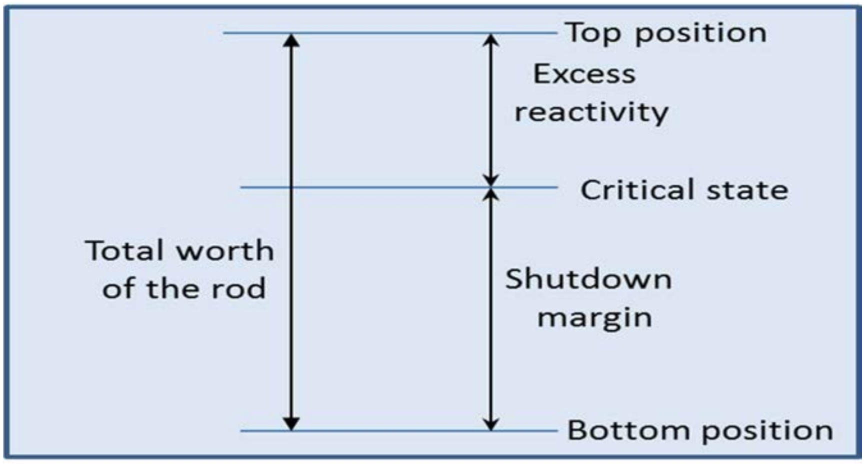


FIG. 13. Definition of the excess reactivity and shutdown margin.

8.6.2. Equipment and conditions

Determination of the three parameters requires the knowledge of the reactivity calibration curve or reactivity worth of the control rods.

A critical state of the reactor should be established with the standard operating conditions and using the standard I&C system. No additional equipment is needed.

With the reactor being operated at low power, i.e. with no feedback, the *SDM* (with or without control rod stuck) is easily determined. At high power, the change in the reactivity after reactor shutdown, which results from the feedback effects, will have to be taken into account for the establishment of the *SDM*.

8.6.3. Methodology

With the reactor in critical state, the position of each control rod for this state is recorded. The recorded positions are then used to establish the *ER* and *SDM*s. To be able to perform such a determination, the total reactivity worth of the control rods fully withdrawn or fully inserted and the control rods reactivity worth calibration curves are necessary as input data.

The following example illustrates how to determine the *ER* and *SDM*s. Let's assume that the reactor has five control rods of which two are the safety control rods. To simplify the calculations, we make the assumption that reactivity varies linearly with control rod position (percentage of extraction out of the core).

Table 8 gives the total worth of each control rod, the position of each control rod for the critical state, the reactivity change obtained for each control rod when it is withdrawn out of the core from its position at criticality to its top position ($\Delta\rho\uparrow$) and the reactivity change obtained for each control rod when it is inserted in the core from its position at criticality to its bottom position ($\Delta\rho\downarrow$).

TABLE 8. EXAMPLE OF CONTROL ROD WORTH AND POSITION AT CRITICALITY TO DETERMINE EXCESS REACTIVITY AND SHUTDOWN MARGIN

	Control Rod 1 (safety rod)	Control Rod 2 (safety rod)	Control Rod 3	Control Rod 4	Control Rod 5
Total control rod reactivity worth (pcm)	2000	2500	1700	1800	1500
Position at the critical state	Fully withdrawn	Fully withdrawn	Fully withdrawn	1/3 withdrawn	2/5 withdrawn
Reactivity change actual to top position $\Delta\rho\uparrow$(pcm)	0	0	0	1200	900
Reactivity change actual to bottom position $\Delta\rho\downarrow$ (pcm)	2000	2500	1700	600	600

From these values the *ER*, *SDM* and *SDM* with control rod 2 stuck (the one with the largest reactivity worth) can be calculated:

$$ER = \sum_{i=1}^{N} \int_{criticality}^{top} \frac{d\rho_i}{dx} dx = 2100 \text{ pcm} \tag{43}$$

$$SDM = \sum_{i=1}^{N} \int_{bottom}^{criticality} \frac{d\rho_i}{dx} dx = 7400 \text{ pcm}$$

SDM with one (CR2) rod stuck = 4900 pcm

These values should then be compared to the requirements prescribed in the OLCs regarding reactivity and discussed according to the corresponding safety issues related to reactor operation. Care should be taken since the excess reactivity is often larger than the beta of the reactor, leading to the risk of criticality accident. Concerning reactor shutdown, it is generally considered that after shutdown the negative reactivity in the core should be decreased to less than -2000 pcm, and values less than -5000 pcm are common.

When considering the operation at high power, the effect of a change in power and in turn in temperature of the core components has to be taken into account (see Section 9.2). The temperature effect will bring a negative contribution to the *SDM* value since a decrease in power will induce a decrease in temperature that will, in turn, increase the reactivity (negative temperature coefficient). This contribution can be calculated from temperature coefficient and the expected change in temperature following reactor shutdown.

Additional discussion can take place concerning the way reactivity insertion can be limited, such as speed limit for control rod withdrawal, limiting the automatic withdrawal of control rod (for example stopping the withdrawal after 15 s or requesting that the control rod can only be withdrawn deliberately by an operator action) or limiting the doubling time to a minimum value, i.e. first implementing an alarm, then inhibiting control rod withdrawal and finally the automatic shutdown (SCRAM) by the reactor protection system when the doubling time is below the set point. Ensuring reactor shutdown with a number of control rods can also be discussed. Using several control rods contributes to a large value of negative reactivity, to a more uniform distribution of absorbing material in the core, to enhanced safety since the probability for having more than one control rod stuck is very low. Considering the *SDM* with one control rod stuck is a way to ensure that the reactor can be safely shut down in such incidental condition and to demonstrate the compliance of the shutdown system with the single failure criterion.

8.6.4. Safety considerations

The safety considerations given in Section 3.5.4 apply. This Section describes additional safety considerations specific to this exercise.

Usual procedures for reactor operation are applied to reach a critical state that will be used for the determination of the excess reactivity and shutdown margin.

From a radiation protection point of view, this exercise does not bring additional risk compared to normal reactor operation.

8.6.5. Documentation

In order to conduct this exercise, the following documents can be given to the students:

(a) Background: reactor principle; core reactivity definition; criticality accident; function, worth and calibration curve of a control rod, limitation of the reactivity relative to safe operation, safety issues related to reactor shutdown;
(b) Schematic of the core with the control rods;
(c) Definition of the excess reactivity, shutdown margin and shutdown margin with one control rod stuck;
(d) Step by step procedure to complete the task, including details on the methodology to establish the ER and *SDM*s;
(e) Worth of the control rods that are fully extracted or inserted, calibration curves of the control rods in intermediate position at criticality;
(f) When feedbacks effects are to be taken into account, values of the reactivity coefficients and changes in the reactor parameters, such as temperature, should be known for *SDM*s determination.

8.6.6. Questions to the students

A first evaluation of the impact of the experiment can be obtained using the following set of questions:

(1) Provide a definition of the excess reactivity and shutdown margin.
(2) Why is it important to limit the excess reactivity? What type of severe accident could result from the insertion of a high reactivity?
(3) Why is it important to have a high value of the shutdown margin? What does that guaranty?
(4) What is the interest of relying on more than one control rod to shut down the reactor? What is the associated practical parameter related to this concept?

Further evaluation of the impact of the experiment can be obtained through a deliverable such as a report or a presentation on the objectives, methodology and results obtained from these measurements, prepared by the students and delivered to the teacher.

For further information on the reactivity control exercises, please consult the bibliography.

9. REACTOR DYNAMICS

9.1. BACKGROUND

The reactor kinetics exercises described in Section 8 are valid for zero power reactors only. This means that the reactor does not have significant feedback effects, i.e. that the reactor parameters such as temperature, pressure, burnup, among others, are constant or do not have an effect on reactivity.

Research reactors are excellent tools for studying the reactor feedback and reactivity coefficients. These exercises can easily be carried out at low power RRs because, at most reactors, these exercises can be done with the standard reactor equipment or they only require simple but specific experimental equipment.

Feedback exercises, which are carried out for students, usually cover the study of the temperature reactivity coefficients and void reactivity coefficients[18].

9.2. THEORY

In most RRs and all power reactors during the standard operation the parameters of the core vary, causing changes to the properties of the core and therefore affecting reactivity. Various reactivity coefficients are defined to make these changes easier to understand and to simplify the modelling of the dynamic and transient processes. They are defined as:

$$a_x^y = \frac{\partial \rho}{\partial x} \tag{44}$$

where

a_x^y: reactivity coefficient

ρ: reactivity

x: reactor parameter, e.g. temperature, power

y: particular part of the core, e.g. fuel or moderator

Using the definition of reactivity[19], Eq. (44) can be rewritten as:

$$a_x^y = \frac{1}{k_{eff}^2} \frac{\partial k_{eff}}{\partial x} \tag{45}$$

Usually k_{eff} is close to 1 and thus Eq. (45) can be written approximately as:

$$a_x^y = \frac{1}{k_{eff}} \frac{\partial k_{eff}}{\partial x} \tag{46}$$

The most important reactivity coefficients are fuel temperature coefficient, moderator temperature coefficient, void coefficient and power coefficient. In low power RRs where

---

[18] Power reactivity coefficients and xenon poisoning experiments are described in Section 9.2.

[19] Reactivity is defined in Section 8.2.

reactor power is very low and fuel burnup is negligible, the power coefficient is also negligible and only temperature and void coefficients are important.

Changes in core parameters directly cause reactivity changes; hence the reactivity coefficients work as a feedback. The basic requirement for the safety operation of the reactor is its dynamic stability, and one of the basic requirements for a stable system is a negative feedback of the system. Therefore, the combination of reactivity coefficients have to be negative to the make the reactor a stable system.

9.2.1. Fuel temperature reactivity coefficient

The fuel temperature reactivity coefficient can be defined using Eq. (44) and Eq. (46):

$$a_T^F = \frac{\partial \rho}{\partial T} \qquad \alpha_T^F = \frac{1}{k_{eff}} \frac{\partial k_{eff}}{\partial T} \tag{47}$$

where

a_T^F: fuel temperature reactivity coefficient
ρ: reactivity
k_{eff}: effective multiplication factor
T: fuel temperature

The fuel temperature reactivity coefficient is based on Doppler effect, i.e. broadening of the resonance capture cross section. Increasing the fuel temperature in the reactor causes a broadening of the resonance capture cross section and thus decreases the resonance escape probability. Doppler effect depends on fuel enrichment, i.e. the $^{235}U/^{238}U$ ratio, because a similar resonance broadening effect can also be found in ^{235}U. For the majority of RRs and all power reactors, this effect is negative, as well as the fuel temperature reactivity coefficient. For high enriched fuel used in some RRs, the Doppler effect is positive and the fuel temperature reactivity coefficient is positive as well.

9.2.2. Moderator temperature reactivity coefficient

The moderator temperature reactivity coefficient can be defined using Eq. (43) and Eq. (46) as:

$$a_T^M = \frac{\partial \rho}{\partial T} \qquad \alpha_T^M = \frac{1}{k_{eff}} \frac{\partial k_{eff}}{\partial T} \tag{48}$$

where

a_T^M: moderator temperature reactivity coefficient
ρ: reactivity
k_{eff}: effective multiplication factor
T: moderator temperature

The moderator temperature coefficient could be positive or negative and depends on the moderator-to-fuel ratio[20]. If the ratio is lower than the 'optimal value', the core is 'undermoderated' and the coefficient is negative. In this case, the decrease in the moderator density will mainly decrease the moderation in the core, causing a decrease in reactivity

---

[20] It is also called the water-to-uranium ratio in light water reactors.

(negative reactivity effect). If the ratio is higher than the 'optimal value' the core is 'overmoderated' or 'over absorbed' and the coefficient is positive. In this case, the decrease in the moderator density will mainly cause a decrease of the neutron absorption in the core. This causes an increase in the reactivity (positive reactivity effect). These two effects are shown in Fig. 14.

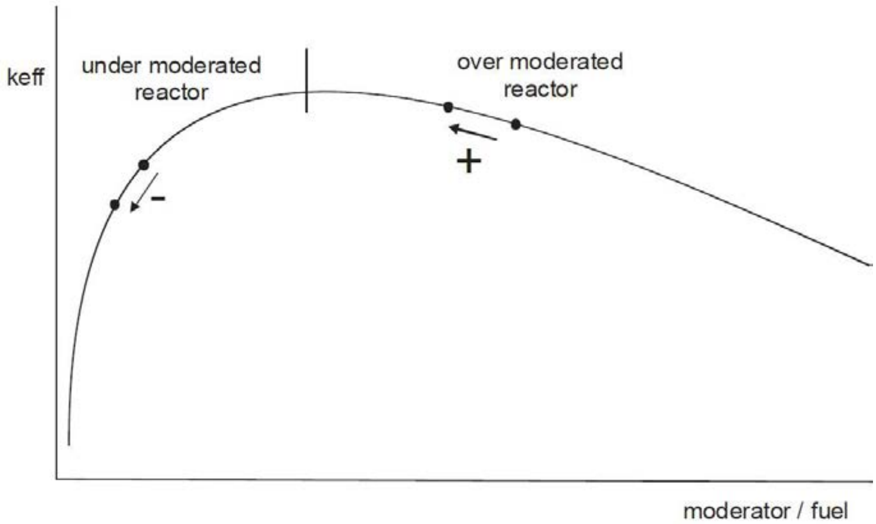


FIG. 14. Moderator-to-fuel ratio in a reactor. (reproduced from [16] with permission courtesy of the Czech Technical University in Prague, Czech Republic)

A more detailed explanation of Fig. 14 is evident from Fig. 15 where a simplified homogeneous model of the VR-1 reactor was established.

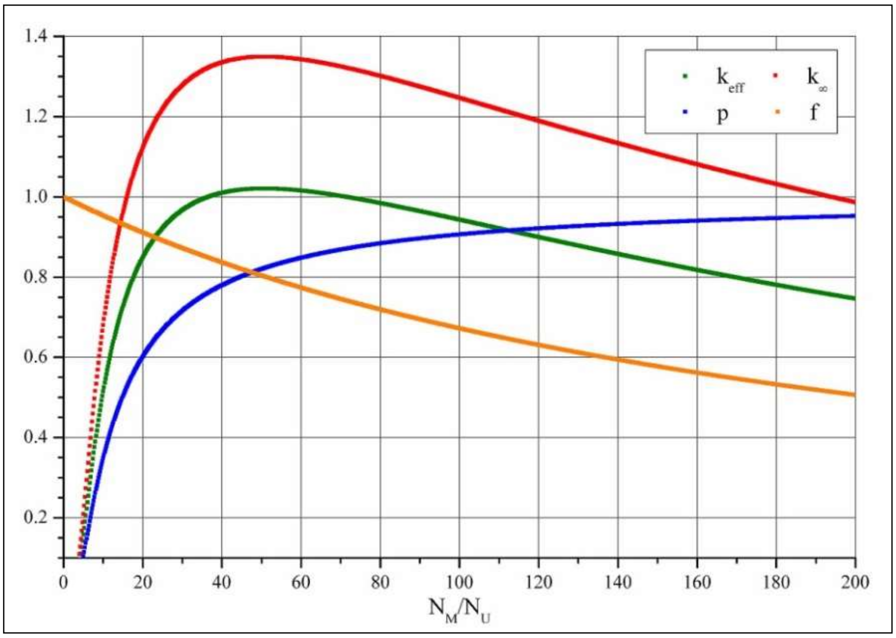


FIG. 15. Moderator-to-fuel ratio reactor (reproduced from [16] with permission courtesy of the Czech Technical University in Prague, Czech Republic).

All four factors of k_∞, together with k_{eff}, were calculated for various moderator-to-fuel ratios, i.e. N_M/N_U, where N_M is the number of moderator atoms and N_U is the number of uranium atoms. The fast fission factor ε and thermal fission factor η are only slightly affected by the moderator-to-fuel ratio, but the thermal utilization factor f and resonance escape probability p are strongly dependent on the moderator-to-fuel ratio. If the amount of moderator in the core increases, (i.e. N_M/N_U increases) the neutron leakage decreases. Neutron absorption in the moderator increases and causes a decrease in the thermal utilization factor f. Having insufficient moderator in the core (i.e. N_M/N_U decreases) causes an increase in slowing down time and results in a greater loss of neutrons by resonance absorption p. This also causes an increase in neutron leakage. Because the moderator-to-fuel ratio affects thermal utilization factor and the resonance escape probability, it also affects k_∞ and k_{eff}. As shown in Fig. 14, there is an optimum point above which increasing the moderator-to-fuel ratio decreases k_{eff} due to the dominance of the decreasing thermal utilization factor. Below this point, a decrease in the moderator-to-fuel ratio decreases k_{eff} due to the dominance of the increased resonance absorption in the fuel.

9.2.3. Void reactivity coefficient

The void reactivity coefficient can be defined using Eq. (44) and Eq. (46) as:

$$a_V^M = \frac{\partial \rho}{\partial V} \qquad a_V^M = \frac{1}{k_{eff}} \frac{\partial k_{eff}}{\partial V} \tag{49}$$

where

a_V^M: void reactivity coefficient
ρ: reactivity
k_{eff}: effective multiplication factor
V: void fraction.

The void reactivity coefficient is usually defined for the reactors which use liquid moderator or coolant, including RRs and NPPs. It is very important in the operation of all light water reactors where steam creates voids in the reactor core. This displaces a certain amount of the moderator from the core, thus affecting reactivity. Void reactivity coefficient acts in a similar way to the moderator temperature coefficient and its behaviour is similar to that of the moderator temperature coefficient. If the reactor is undermoderated, the void reactivity coefficient is negative, and in the case of an overmoderated reactor, the void reactivity coefficient is positive. Figure 14 and Fig. 15 can be used for explaining the nature of a void reactivity coefficient in a way similar as for a moderator temperature coefficient. The void reactivity coefficient is used for a case where the moderator or coolant changes state from liquid to gaseous, i.e. boiling occurs in the core[21].

9.2.4. Long term reactivity feedback effects

The long term reactivity feedback effects are related to the power reactivity coefficient that can be defined in the same way as the temperature and void reactivity coefficients:

$$a_P = \frac{\partial \rho}{\partial P} \qquad a_P = \frac{1}{k_{eff}} \frac{\partial k_{eff}}{\partial P} \tag{50}$$

where

a_P: power reactivity coefficient

[21] The terms ‘steam reactivity coefficient’ or ‘bubble reactivity coefficient’ are used in some of the sources in the Bibliography.

ρ: reactivity
k_{eff}: effective multiplication factor
P: reactor power

The power reactivity coefficient is closely connected to both temperature effects, because with increasing the reactor power there is usually a correspondent increase in both the fuel and moderator temperature.

Xenon poisoning is caused by the extremely large ^{135}Xe absorption of thermal neutrons[22]. More than 95% of ^{135}Xe is produced from the radioactive decay of ^{135}I produced by ^{235}U fission according to the chain in the scheme (S1). Additionally, approximately 5 % of ^{135}Xe is produced directly by the fission of ^{235}U. ^{135}Xe can be disappearing from the core either through the capture of neutrons or through radioactive decay according to scheme (S1):

$$^{135}Te \xrightarrow{\beta^-(19\,s)} {}^{135}I \xrightarrow{\beta^-(6.6\,h)} {}^{135}Xe \xrightarrow{\beta^-(9.1\,h)} {}^{135}Cs \xrightarrow{\beta^-(2.3\,mil.\,years)} {}^{135}Ba \qquad \text{(S1)}$$

The concentration of ^{135}I and ^{135}Xe can be easily expressed by the following equations:

$$\frac{dN_I(t)}{dt} = -\lambda_I N_I(t) + y_I \Sigma_f \phi(t) \qquad (51)$$

$$\frac{dN_{xe}(t)}{dt} = \lambda_I N_I(t) + y_{xe} \Sigma_f \phi(t) - \lambda_I N_{xe}(t) - \sigma_{xe} N_{xe}(t)\phi(t) \qquad (52)$$

where
$N_I(t)$, $N_{xe}(t)$: ^{135}I or ^{135}Xe concentration
λ_I, λ_{xe}: ^{135}I or ^{135}Xe decay constant
y_I, y_{xe}: ^{135}I or ^{135}Xe fission yield
σ_I, σ_{xe}: ^{135}I or ^{135}Xe microscopic cross section for absorption
Σ_f: macroscopic cross section for fission
$\varphi(t)$: neutron flux

Because of the small ^{135}Xe fission yield in comparison to ^{135}I fission yield (approximately 5% only) the direct origin of ^{135}Xe is often neglected and Eq. (52) can be written as follows:

$$\frac{dN_{xe}(t)}{dt} = \lambda_I N_I(t) - \lambda_{xe} N_{xe}(t) - \sigma_{xe} N_{xe}(t)\phi(t) \qquad (53)$$

Performing an analytical solution of both Eq. (51) and Eq. (53) for steady state neutron flux $\varphi(t) = f = constant$, we get:

$$N_I(t) = [N_I^o + y_I \Sigma_f \phi \frac{1}{\lambda_I}(e^{\lambda_I t} - 1)]e^{-\lambda_I t} \qquad (54)$$

---

[22] Absorption of thermal neutrons in one xenon nucleus is equal to the absorption of thermal neutrons in approximately five thousand ^{235}U nuclei.

$$N_{xe}(t)=(\lambda_I N_I^o - y_I \Sigma_f \phi)(e^{-\lambda_I t} - e^{-(\sigma_{xe}\phi+\lambda_{Xe})t})(\frac{1}{\sigma_{xe}\phi+\lambda_{xe}-\lambda_I})+$$
$$+(N_{xe}^o+\frac{y_I \Sigma_f \phi}{\sigma_{xe}\phi+\lambda_{xe}})e^{-(\sigma_{xe}\phi+\lambda_{Xe}t)}+\frac{y_I \Sigma_f \phi}{\sigma_{xe}\phi+\lambda_{xe}} \tag{55}$$

Xenon and iodine equilibrium is achieved when a neutron flux is in a steady state for a sufficient time, i.e. t→∞, and Eq. (54) and Eq. (55) can be expressed in a simpler form:

$$N_I^\infty = \frac{y_I \Sigma_f}{\lambda_I}\phi \qquad N_{xe}^\infty = \frac{y_I \Sigma_f \phi}{\lambda_{xe}+\sigma_{xe}\phi} \tag{56}$$

After the reactor shutdown, the absorption of ^{135}Xe dramatically decreases and its production from ^{135}I precursor remains. The xenon poisoning peak is created as shown in Fig. 16.

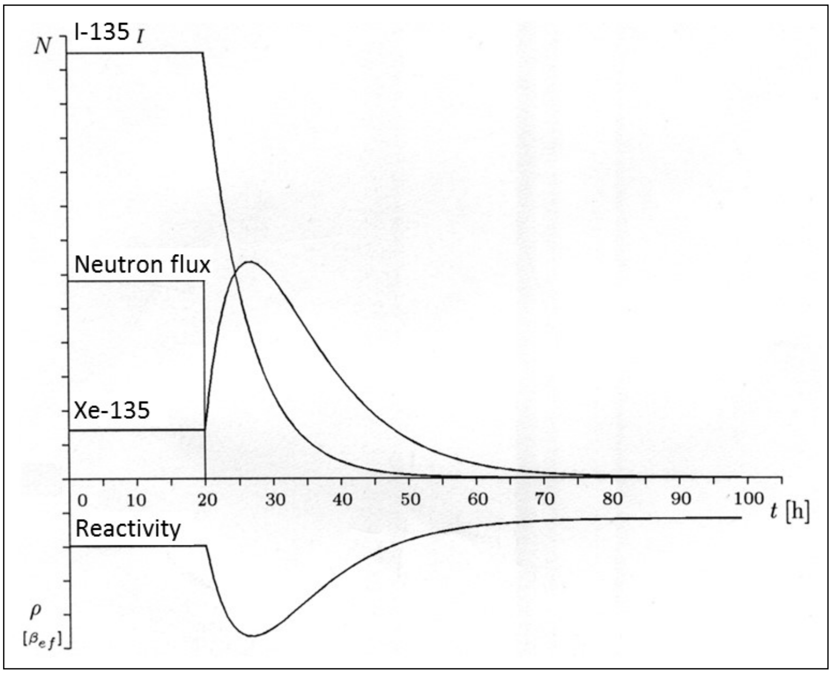


FIG. 16. The xenon poisoning peak after reactor shutdown. (Courtesy of the Czech Technical University in Prague, Czech Republic)

When a RR with sufficient power is analysed in the long-term perspective of few weeks, months or years, fuel burnup, i.e. decrease of nuclear fuel, in the reactor core can be observed. The measurable effect of burnup strongly depends on the reactor power and time of investigation. In the case of low fuel burnup in the reactor core the effect can be described through the basic processes displayed as shown in the scheme (S2).

$$
{}^{235}U + n \begin{array}{l} \nearrow \textit{fission} \\ \searrow {}^{236}U \end{array}
\qquad
{}^{238}U + n \longrightarrow {}^{239}U \xrightarrow{\beta^-} {}^{239}Np \xrightarrow{\beta^-} {}^{239}Pu
$$

$$
{}^{239}Pu + n \begin{array}{l} \nearrow \textit{fission} \\ \searrow {}^{240}Pu \end{array}
\qquad
{}^{240}Pu + n \longrightarrow {}^{241}Pu
\qquad
{}^{241}Pu + n \begin{array}{l} \nearrow \textit{fission} \\ \searrow {}^{242}Pu \end{array}
\tag{S2}
$$

The concentration of ^{238}U in the core (with low enriched fuel) is higher than the ^{235}U concentration. In the case of low burnup[23] in the first assumption concentration of ^{238}U can be observed as constant. The only relevant process related to ^{238}U is production of ^{239}Pu through resonance absorption. Because of very low absorption cross sections of ^{236}U, ^{239}U and ^{239}Np, absorption on these isotopes can be neglected. The half-life of ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu and ^{241}Pu is longer than a few weeks or months and thus can also be neglected. Due to low burnup, ^{242}Pu can be assumed as the final isotope which can be produced and because of low yield of neutrons from ^{241}Pu fission its contribution to production of ^{239}Pu also can be neglected. Based on these assumptions the following system of Eq. (57) can be formulated which describe the simplified model of fuel burnup in the core of RRs:

$$
\begin{aligned}
&\frac{dN_5(t)}{dt} = -\varphi\sigma_5^f N_5(t) \qquad\qquad \frac{dN_8(t)}{dt} = 0 \\
&\frac{dN_9(t)}{dt} = \varphi\sigma_8 N_8(t) \ + \varepsilon(1-p)(\nu_5\sigma_5^f N_5(t) + \nu_9\sigma_9^f N_9(t))\varphi - \varphi\sigma_9 N_9(t) \\
&\frac{dN_0(t)}{dt} = \varphi\sigma_9 N_9(t) \ - \varphi\sigma_9^f N_9(t) - \varphi\sigma_0 N_0\ (t) \\
&\frac{dN_1(t)}{dt} = \varphi\sigma_0 N_0(t) \ - \varphi\sigma_1 N_1(t)
\end{aligned}
\tag{57}
$$

where

$\varphi(t)$: neutron flux

$N_5(t)$, $N_8(t)$, $N_9(t)$, $N_0(t)$ $N_1(t)$, $N_2(t)$: ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, concentrations

σ^f_5 and σ^f_9: fission cross section of ^{235}U and ^{239}Pu

σ_8, σ_9, σ_0, σ_1: absorption macroscopic cross section of ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu

ν_5 and ν_5: number of neutrons produced per fission of ^{235}U and ^{239}Pu

ε: fast fission factor

p: resonance escape probability

Time t in the Eq. (57) can be replaced by new parameter named effective time z which is more appropriate to describe the changes in the fuel because it also considers neutron flux, i.e. reactor power (during the same time, in reactor with higher power the fuel burns more than in reactors with lower power). Effective time z defined in Eq. (58) is very close to fuel parameter named burnup B, but burnup B is a more complex parameter where more effects should be included.

$$
dz = \varphi(t)\,dt \tag{58}
$$

---

[23] Comparing to nuclear power plants.

Eq. (57) can be solved analytically or numerically using Eq. (58). The analytical solution for ^{235}U and ^{239}Pu is shown in Eq. (59), where constants C_1, C_2, C_3 and C_4 are shown in Eq. (60):

$$\begin{aligned} N_5(z) &= C_1 N_5^0\, e^{-z} \\ N_9(z) &= C_3(1-e^{-C_2 z})N_8^0 + C_4(e^{-z} - e^{C_2 z})N_5^0 \end{aligned} \tag{59}$$

$$C_1 = \sigma_5;\quad C_2 = \frac{\sigma_9}{\sigma_5} - \varepsilon(1-p)\nu_9 \frac{\sigma_9^f}{\sigma_5};\quad C_3 = \frac{\sigma_8}{\sigma_5 C_2};\ C_4 = \frac{\varepsilon(1-p)\nu_5 \dfrac{\sigma_5^f}{\sigma_5}}{C_2 - 1} \tag{60}$$

9.3. EDUCATIONAL ASPECTS

The exercises presented in the next Sections belong to the group of reactor physics exercises. The learning objectives, information on the type of students and the level of the exercises are the same to those described in Section 6.3.

9.3.1. Exercises on reactor dynamics

Usually no specific experimental instrumentation is needed for carrying out the temperature reactivity coefficient exercises for RRs with power typically above 50 kW. A RR and its standard equipment and instrumentation (including the measurement of the water temperature) are enough. Additional information about the water temperature at a fuel plate position and the fuel temperature would allow a more detailed study of the effect.

For the void reactivity coefficient exercises, specific reactor features or specific experimental instrumentation are usually needed in order to introduce in the core coolant/moderator a void fraction.

Prior to the study of the reactor dynamics and feedback effects students should be familiar with the reactor kinetics, i.e. without feedbacks. The background needed for this exercise is given in Section 9.2. It includes the knowledge of the different effects modifying the core reactivity (Doppler, moderator density change and void effects) that includes their origin and their impact on the reactivity: increase or decrease, kinetics of the phenomenon, order of magnitude of reactivity change associated to the temperature coefficient. The control rod calibration curve to determine the core reactivity changes needs to be available.

These exercises are appropriate for bachelor's, master's and doctoral programmes. They are suitable for students studying nuclear engineering as the major curriculum as well as for students studying various major engineering curricula. The level of the exercise usually goes from intermediate to advanced level.

For the intermediate level, the overall temperature feedback effect (Doppler + moderator density change) can be observed and the corresponding reactivity coefficient can be estimated. The time dependent changes in core reactivity, which is instantaneous for the Doppler Effect but exhibits time delay for the density effect, can also be evidenced and discussed. The self-stabilization of the reactor power through the temperature effect can be shown. The void effect, which can be emulated by the injection of gas bubbles from the bottom of the core or the introduction of aluminium pins transparent to neutrons in comparison to water, can also be observed and discussed. Finally, the safety issues related to these effects can be analysed. Total duration of such an exercise is typically 2–3 hours.

For the advanced level, additional exercises and studies can be carried out. The measurement of both fuel and water time dependent temperature can allow the determination of each reactivity coefficient. Deeper analysis of the reactivity effect and their safety related issues can be analysed, including incidental conditions where the core reactivity can increase (if water temperature decrease due to cold water injection for example). Depending on the content, additional duration is expected to range from 1–3 hours.

9.3.2. Exercises on long term reactivity feedback effects

For carrying out studies of long term core reactivity feedback effects, usually no specific experimental instrumentation is needed. only the RR itself and its standard technology, together with an appropriate neutron detector system, are necessary. The study of the effect of xenon in reactor operation can be studied, but, contrary to other reactor physics exercises, it requires much more time (typically 30 to 100 hours) which makes it difficult to integrate it in real time in an educational programme. It is rather expected that recorded data can be given to students to understand the xenon poisoning effect. Such an exercise can be carried out in RRs with power above 100 kW where xenon effect is measurable. Study of fuel burnup requires a RR with a minimum power of 1 MW, but power levels of 5–10 MW are more appropriate. Again, such an exercise requires time, and previously recorded data can be used instead.

Prior to the study of the long term reactivity effects, the students should be familiar with the background given in Section 9.2. It includes: (1) the knowledge of the processes for xenon production and removal as well as their kinetics; (2) the knowledge of the fission and capture processes that controls ^{235}U, ^{238}U, ^{239}Pu and ^{241}Pu concentrations, as well as their kinetics. It has to be completed by knowledge of the safety issues related to the change in xenon concentration (including the definition of the ‘xenon peak’) that has a strong impact on the reactivity of the core.

The learning objective of the exercise is to understand the long term effects at the reactor. Core poisoning has an impact on reactor operation and safety when fast changes in power are applied. The fuel burnup is compensated by the control rods and limits the fuel life and reactor cycle duration on high power reactors.

This exercise is appropriate for bachelor’s, master’s and doctoral programmes. This exercise is suitable for students in nuclear engineering as the major curriculum as well as for students in various major engineering curricula. The level of the exercise can be adjusted according to the student’s background and pedagogical objectives. Study of the long term reactivity feedback effects usually corresponds to intermediate or advanced level.

For the intermediate level, it is suggested to study the reactivity change related to xenon poisoning that can be further completed by the analysis of the safety issues related to reactor start-up after reactor shutdown. Duration of a complete exercise needs to operate the reactor for a few dozens of hours to evidence the buildup of the poison concentration. However, the exercise can be conducted over 3 to 6 hours with the use of recorded data and a complementary software application.

For the *advanced level*, the study of the reactivity change related to fuel burnup can be added to the exercise. This exercise can be carried out on high power reactors that are run by cycles of a few weeks. It is advised to use data recorded during one cycle and analyse them according to core calculations carried out with a benchmarked code. Illustration of the effect can be shown and discussed in 2–3 hours. A detailed study related to core calculations can be done in the frame of a micro project for the students.

9.4. EXERCISES ON REACTOR DYNAMICS

9.4.1. Objective

The learning objective of the exercises is to understand the behaviour of RRs at higher power, i.e. with feedbacks, in order to support safe reactor operation. This guideline describes exercises related to the temperature effect and the void effects. The theoretical basis is described in Section 9.2.

Temperature effects include the Doppler effect occurring within the fuel and the water density change effect. Both can be studied when the reactor power is increased above a few dozens of kW. The void effect can be reproduced by the injection of argon gas, in the form of small bubble from the bottom of the core or introducing samples of aluminium replacing moderator in the core. In both cases, the feedback effect will result in a reactivity change. The latter can be calculated through the change in the position of the regulating rod used to maintain reactor critical. The calibration curve of the regulating rod is then used to evaluate the reactivity change according to position variation.

9.4.2. Equipment and conditions

To study the temperature and void effect the reactor should be initially in a state where no feedback is observed, i.e. typically at low power. This state will serve as a reference state to study the reactivity change. Then, the exercise will be carried out by modifying the reactor state and studying the resulting reactivity change.

To perform the temperature coefficient exercises in RRs with power above a few dozens of kW, no specific experimental instrumentation is needed. Following the change in critical position of one regulating rod will give information about reactivity change. Sensors to measure the water and/or fuel temperature can give additional information and would allow precise study of the Doppler and moderator density change effects.

For a low power RR (typically less than a few dozens of kW) exhibiting little temperature effect, it is possible to artificially change the temperature of coolant and thus the temperature of the moderator and the fuel. For example, a water tank equipped with a heater can be used to raise the temperature that water that will then be introduced in the core through the loop of the primary coolant circuit or through a specific loop, if available, that introduces the water directly to the core.

To perform the void exercise, the reactor should be equipped with specific devices. One way of simulating the void effect is to inject gas bubbles from the bottom of the core. The flow rate of the gas can be changed using a regulating valve. Another option to emulate the void effect is by inserting small aluminium devices (pins for example) in the core. Replacing the corresponding volume of water by aluminium, which has lower neutron absorption cross section and is a much less efficient moderator than water, is equivalent to reducing the volume of moderator as void will do.

9.4.3. Methodology

9.4.3.1.Temperature effect

In this exercise, a reference state is achieved with the reactor critical at low power. Then the core temperature is changed by increasing a reactor power to a level indicating significant temperature feedback effect, i.e. typically above a few dozens of kW.

Initially, the reactor power is increased from a low power level to a level stabilized by the automatic power control mode. The temperature will increase, and the behaviour of the reactor related to the temperature effect can be followed up. In most of the cases, it is possible to distinguish between the two main temperature effects, the Doppler effect taking place in the fuel and the water density change effect. Both effects contribute to a reactivity decrease when the temperature is increased (for safety reasons the reactor is undermoderated).

Figure 17 gives an example of the changes on the values of the reactor parameters (power, water temperature and position of the regulating rod) during a temperature exercise carried out while the reactor is operated in natural convection, i.e. without forced convection cooling, in order to reach higher values of temperature.

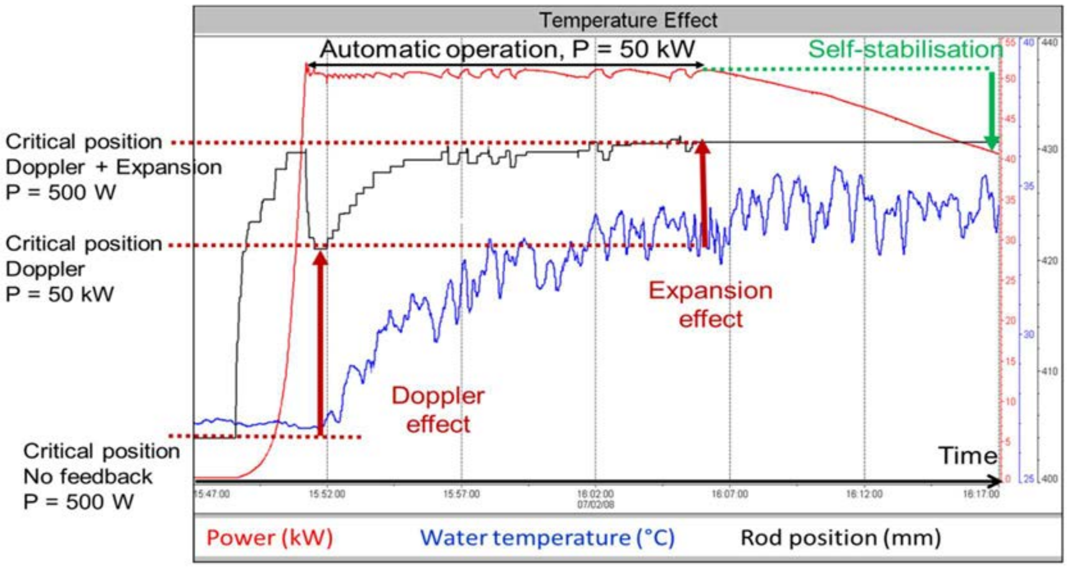


FIG. 17. Example of a curve obtained for the study of the temperature effect. (Courtesy of the National Institute for Nuclear Science, CEA Saclay, France)

In the example illustrated in Fig. 17, the reactor power is increased from 500 W to 50 kW. When the reactor power reaches 50 kW, the water temperature has not been increased yet and the critical position of control rod has changed from 405–423 mm. This corresponds to reactivity change of about 120 pcm (according to the reactivity calibration curve of the control rod) that can be attributed to Doppler effect. The reactor power is then kept constant at 50 kW with the automatic power control system and the critical position of the control rod changes in parallel to the water temperature. The percentage of the control rod withdrawn compensates the decrease in core reactivity attributed to moderator expansion effect (change in density). This exercise shows that the Doppler and moderator expansion effects are taking place on different time scales. Doppler effect is instantaneous since the fuel temperature changes immediately when power is increased. Water expansion takes time since it results from the temperature increase in the water due to heat exchange from fuel to water and because the volume of water is much larger than the volume of fuel.

Establishment of natural convection after a certain period of time can be observed by utilizing a camera installed in the pool and showing the top of the core.

Both Doppler and moderator expansion effects give a negative contribution to reactivity. Thus, the overall temperature coefficient is negative, i.e. the reactor is undermoderated. The recorded values of the critical position of the control rod at 500 W and 50 kW and its change, can be used to determine the overall temperature coefficient. In the case of the example, the overall temperature coefficient equals to –17 pcm/°C.

This exercise is also used to show the self-stabilization of the reactor when switching from automatic to manual control (at t = 16:02:30). The reactor power decreases by itself due to the continuing increase in water temperature.

In such an exercise, if the reactor is equipped with sensors measuring both the fuel and water temperatures, it is possible to determine both the fuel and water reactivity coefficients.

Additionally, such an exercise can be used to emphasize the safety issues related to the temperature feedback effect. On the one hand it is important to design a undermoderated reactor that, in normal operation, exhibits a negative reactivity coefficient. This ensures the self-stabilization of the reactor observed in Fig. 17. On the other hand, potential incidental condition may arise if the water temperature is decreased. Indeed, in the example given above a decrease of 20 degrees of water temperature would result in an increase of reactivity by 340 pcm. This would in turn result in a fast increase of power (doubling time typically less than 3 s) that would lead to an automatic reactor shutdown. As a practical illustration of the design of the reactor protection system of RRs, in the frame of this exercise and if the reactor OLCs allows it, switching the reactor from natural to forced convection, will automatically trip the shutdown system of the reactor by means of the reactor protection system to stop the uncontrolled reactivity increase event.

9.4.3.2. Void effect

The void effect is studied at low power with no perturbation from the other feedback effects such as temperature effect. From a practical point of view, this exercise should be carried out after the reactor has been shutdown for a sufficient period of time, in order to guarantee that there is no other feedback effect present in the core from previous operation. At the beginning of the exercise the reactor is maintained in a critical state. The position of the regulating rod is recorded, z_{ref}.

The injection of gas bubbles or the replacement of water by aluminium devices can be used to emulate void increase. Starting from the initial state with no void feedback effect, the quantity of void can be increased step by step, changing the gas flow or a number of aluminium devices.

During this exercise the reactor can be maintained in a critical state while the parameter, i.e. void fraction introduced in the core, is changed and the regulating rod is used to compensate for the change in the reactivity. This can be done while the reactor is operated in automatic mode by changing slowly the reactor parameter if this does not induce safety risks and the OLCs allows this operation. To do so, reactivity changes should typically be limited to less than 100 pcm. This will limit the corresponding doubling time to reasonable values (>40 s) if for any reason the reactivity change was not compensated automatically or manually, or if the reactivity was increased suddenly (for example if system runs out of gas).

For each critical state *i*, the critical position of the regulating rod, z_i, is recorded. Using the reactivity calibration curve of the regulating rod and z_i, the reactivity change in the core $\Delta\rho_i$ is determined. A table similar to Table 9 can be filled during the exercise.

TABLE 9. CRITICAL POSITION OF THE REGULATING ROD z_i AND REACTIVITY CHANGE $\Delta\rho_i$ PER REACTOR STATE

State	Position of the regulating rod	Reactivity change
Reference	z_{ref}	0
State 1	z_1	$\Delta\rho_1$
State 2	z_2	$\Delta\rho_2$
State i	z_i	$\Delta\rho_i$

The reactivity change versus gas flow rate or number of aluminium devices inserted in the core can be plotted to establish general trend of the reactivity change in the core related to the void effect: reactivity decreases when the void fraction increases.

This behaviour, as well as the safety issues related to the void effect when the reactor is undermoderated, can be discussed with reference to Fig. 14.

Since the emulation of the void effect is usually done locally in one region of the core, it is generally difficult to establish an exact value of the void coefficient in such an exercise.

9.4.4. Safety considerations

The safety considerations given in Section 3.5.4 apply. This Section describes additional safety considerations specific to this exercise.

Usual operating procedures for reactor operation at power resulting in temperature effect are applied. Before implementing the exercises, a safety analysis should be conducted to check that the reactivity cannot vary in an uncontrolled manner in normal and incidental conditions resulting in unsafe reactor operation. For example, in an undermoderated reactor, any sudden decrease in the water temperature should be avoided, i.e. no switch from natural to forced convection while the reactor is in operation should be allowed. Also, reactivity change related to the void effect should be strictly limited to values typically not larger than 100 pcm, giving rise to reasonably slow variations of the neutron density (doubling time of the order of 40 seconds).

From a radiation protection point of view, this exercise should not bring additional risk compared to normal reactor operation. For the void exercise, risks arising from the activation of the gas injected in the core or from the manipulation of the activated aluminium devices should be considered.

9.4.5. Documentation

In order to conduct this exercise, the following documents can be given to the students:

(a) Background: reactor kinetics and dynamics, moderation factor, Doppler and water expansion effects, temperature and void coefficients;

(b) Schematic of the core showing the system implemented for gas injection or the insertion of aluminium devices;
(c) Calibration curve of the rod used for the compensation of the feedback effects;
(d) Step by step procedure to complete the task;
(e) Graph paper or software application to plot the curve;
(f) Specific experimental related information and rules: rules related to the limitation of the change in the reactivity for the void exercise for example.

9.4.6. Evaluation

A first evaluation of the impact of the exercise can be obtained using the following set of questions:

(1) What is the first temperature feedback effect taking place when the reactor power is increased? In which component of the reactor does it take place?
(2) When the power is maintained constant through the automatic power control system, reactivity slowly decreases before it reaches a reactivity equilibrium implying a new different position for the regulation rod critical position. To which effect is this related this behaviour? What is the reason for the inertia of the system to reach a new stable reactivity value?
(3) Are the corresponding temperature coefficients negative or positive? What does this mean in relation to the moderation factor of the reactor? Is this situation good from the safety point of view?
(4) Explain what would happen if reactor is switched from natural to forced convection when reactor has reached an equilibrium in temperature? What is the safety issue related to this action?
(5) From a safety point of view, what should be the maximum value of the reactivity change related to the void effect during the experiment? Why?

Further evaluation of the impact of the experiment can be obtained through a deliverable such as a report or a presentation on the objectives, methodology and results obtained from these measurements, prepared by the students and delivered to the teacher.

9.5. EXERCISE ON LONG TERM REACTIVITY FEEDBACK EFFECTS

9.5.1. Objective

As indicated in Section 9.2, the study of long term reactivity feedback effects is closely related to reactor dynamics experiments. Long terms effect can be caused by xenon poisoning (and other poisons) and fuel burnup.

Carrying out exercises on the long term reactivity feedback effect needs reactor operation at high power, typically above 100 kW, and for longer periods of time, typically for dozens of hours for the poisons and weeks for the fuel burnup. Thus conducting an exercise on long term reactivity feedbacks effects needs either specific organization or the use of recorded data.

The objective of this exercise is to characterize the long term reactivity feedbacks effects related to reactor operation at high power. It includes the study of the change in the critical state of the reactor, i.e. position of the control rods, as a function of time. The change in the reactivity can then be analysed and related to the poisoning effect or fuel burnup. Finally, the safety issues related to xenon poisoning can be discussed.

9.5.2. Equipment and conditions

Usually no specific experimental instrumentation is needed for carrying out studies of long term reactivity feedback effects.

Xenon influence on core reactivity can be studied in RRs with a power above few hundreds kW. In contrast with other exercises, which can be carried out in 1–3 hours, the study of xenon effect requires longer periods, i.e. typically 30 to 100 hours. This makes it impossible to complete the exercise in real time. Also, due to fuel burnup issues, it is not worth to operate the reactor at high power for such a long period of time just to carry out the xenon experiment. Thus it is advised to carry out the experiment once in order to record the data that will be later used to illustrate the xenon effect. If the RR has to be used for other purposes than education and if proper planning can be organized, it is also possible to bring the students to the facility at certain stages of the experiment so they can collect by themselves the data needed. It is, for example, possible to bring the students for the last 3 hours of the experiment. They can measure the reactor data needed for these last three hours while they are given the data recorded over the previous 30–100 hours.

In addition to the experimental measurements carried out on the reactor, the study of core poisoning (xenon and samarium for example) can easily be completed by computed calculations. Such calculations give a lot of flexibility in studying the poisoning of the core resulting from changes in the reactor power.

Study of fuel burnup requires a RR with a high power, i.e. a minimum power of 1 MW and preferably above 5 MW. This experiment also requires time, typically a few weeks and thus it is preferable to use previously recorded data.

In practice, the poisoning and fuel burnup effects will be evidenced by the change in the critical configuration of the reactor, i.e. the control rod positions for the critical state, during the reactor operation cycle. These variations in the position of the control rods will then be evaluated, using the reactivity calibration curves of the control rods, to determine the change in the core reactivity.

9.5.3. Methodology

9.5.3.1.Study of the poisoning effect

Poisoning effect in a reactor is not only related to xenon, but it has a major contribution to core reactivity when the reactor is operated at high power.

Indeed, change in the poison concentration is not directly measured in the reactor in this exercise. But poisoning effect is evidenced through the corresponding change in the core reactivity. The later one is compensated by the change in the position of the regulating rod(s) to maintain reactor in critical state.

In practice, large changes in the power level also induces significant changes in the temperature of reactor components which will in turn reflect in significant variations in the core reactivity through the temperature feedback effects. These changes will have to be taken into account in the analysis of the experimental results. This is possible to be done, i.e. to separate in the experimental results the impact of the temperature feedback effects, because they occur in a time scale (instantaneous for Doppler and in the hour range for water expansion) which is shorter than for the poisoning effect.

Figure 16 shows the expected change in xenon concentration when starting up the reactor, operating it for 20 hours at high power before reducing the power to zero to observe the xenon peak. From an experimental point of view, an increase in xenon concentration will induce a decrease of the core reactivity that is compensated by the motion of the regulating rod that is moved up to compensate for this reactivity decrease. Reducing the power to a very low level (10 W for example) instead of directly shutting down the reactor gives the opportunity to maintain the reactor critical and to follow the variation of the core reactivity over time.

The following experimental procedure can be applied to study this xenon effect:

(a) In its initial state, reactor should be critical at a very low power P_0 with no feedback effects from previous and actual operation. The critical configuration is used as the reference state for the experiment;
(b) Reactor power is increased up to the nominal power or a significant power (typically few hundreds of kW and above) P_1;
(c) Reactor is maintained at P_1 for 30 hours. The position of a regulating rod, and water temperature in the primary circuit (temperature outlet, T_{outlet}), are regularly recorded to identify the time when thermal equilibrium is reached. This configuration corresponds to a second reference state from which the poisoning effect buildup plays a major role in the change in the control rods positions;
(d) At $t = 30$ hours, the power is decreased down to P_0. From the point of view of xenon removal by neutron capture, a strong decrease in the neutron density is experimented, by typically more than 10^4 times, and is almost equivalent to a reactor shutdown. However, operating the reactor at low power allows for the critical state to be maintained and the corresponding critical position of the regulating rod to be recorded as a function of time;
(e) In order to analyse the main part of the xenon peak, the reactor should be maintained at P_0 for at least 48 hours. The position of the regulating rod and the water temperature in the primary circuit needs to be regularly recorded to identify the time when new thermal equilibrium is reached. This corresponds to a third reference state from which the successive buildup (due to the decay of iodine to xenon) and decrease (due to xenon radioactive decay) of the xenon concentration are observed.

Table 10 can be used as an example of how to record data during the exercise. Using the calibration curve(s) of the regulating rod(s), the change in the position of the rod(s) can be used to establish the change in core reactivity related to the poisoning effect. The students can then plot a curve showing the change in core reactivity versus time. This curve can then be analysed taking into account the core reactivity change.

TABLE 10. EXAMPLE OF TABLE TO BE FULFILLED DURING THE XENON POISONING EXERCISE

Date and time	Start:		End:		
Time (hour)	Reactor power P (W)	Water temperature T_{outlet} (°C)	Regulating rod position n_1 Z_1 (mm)	Regulating rod position n_i Z_i (mm)	Reactivity change (pcm or $)
t_0	P_0	T initial	Z reference 1	Z reference 1	0
...					
t_1	P_1				
...					
t_2		T equilibrium	Z reference 2	Z reference 2	
...					
t_3 (30 h)	P_0				
…					
t_4		T equilibrium	Z reference 3	Z reference 3	
…					
t_5 (80 h)	End of experiment				

9.5.3.2.Study of the fuel burnup effect

At medium power reactors, fuel burnup effect takes place over years. Such reactors are usually not operated in a continuous way before the maximum burnup is reached. For such facilities, the exercise is generally limited to the use of reactor parameters recorded at different time intervals. For a given configuration (core and experimental devices) and state (power, temperature, poisoning) of the reactor, data regarding the energy produced by the reactor, by summing different sequences of reactor operation, and corresponding change in the control rods position can be established. Using the reactivity calibration curve of the control rods, the corresponding reactivity change can be calculated. From these data, the influence of the fuel burnup can be evidenced and analysed according to theory.

At high power reactors, which are often operated at their nominal power for a cycle of few weeks, the evolution of the position of the control rods as a function of time during the cycle will evidence the fuel burnup effect in a similar way.

9.5.4. Safety considerations

The safety considerations given in Section 3.5.4 apply. This Section describes additional safety considerations specific to this exercise.

Usual procedures for reactor operation, such as those related to continuous operation at high power and to xenon effect (no restart after a reactor SCRAM), are applied.

From a radiation protection point of view, this experiment should not bring additional risk compared to normal reactor operation.

9.5.5. Documentation

In order to conduct this exercise, the following documents can be given to the students:

(a) Background: long term effects related to power reactivity coefficient, xenon poisoning and fuel burnup;
(b) Calibration curve(s) of the rod(s) used for reactor regulation and/or compensation of fuel burnup;
(c) Step by step procedure to complete the task and questions;
(d) Data previously recorded on the reactor since the duration of the poisoning or fuel burnup experiment (dozens of hours to years) exceed the standard duration of a session on the reactor (3 hours);
(e) Graph paper or software application to plot the curve;
(f) Specific experimental information and rules related to conditions for reactor restart after a reactor SCRAM is interesting in the light of the study of the xenon poisoning for example.

9.5.6. Evaluation

A first evaluation of the impact of the exercise can be obtained using the following set of questions:

(1) What is the impact of xenon poisoning on the reactor operation and what are the actions needed to maintain reactor at constant power level?
(2) What is typical time scale of xenon poisoning effect? Justify the experimental results with the time constant for xenon production from iodine radioactive decay and removal through radioactive decay.
(3) How long, after reactor start-up, does the xenon concentration reach equilibrium?
(4) How long, after reactor shutdown, does the ^{135}Xe concentration reach its peak value? What is the expected peak value of negative reactivity?
(5) Establish simple rules for safe reactor start-up after a SCRAM.
(6) Explain the effect of the fuel burnup on reactor operation. From an operational point of view, what limits the lifetime of the fuel?

Further evaluation of the impact of the experiment can be obtained through a deliverable such as a report or a presentation on the objectives, methodology and results obtained from these measurements, prepared by the students and delivered to the teacher.

For further information on the long term effects exercises, please consult the bibliography.

10. REACTOR POWER CALIBRATION

10.1. BACKGROUND

Reactor power calibration is closely related to safe operation of a RR. A well-established calibration between the neutron signals measured by the I&C and the thermal power is essential, in particular for RRs exhibiting significant moderator and coolant temperature increase during standard operation. Reactor power calibration together with the calibration of the neutron detection systems is carried out periodically in all RRs. It is typically performed at least once a year or after any significant change in the reactor configuration, such as refuelling or after a change in the position of the detectors.

10.2. THEORY

Fission of ^{235}U produces a large amount of energy, but not all the released energy can be further recovered in a reactor as a source of heat. Energy released from ^{235}U fission can be determined from mass balance before and after fission and its equivalent to energy. For example, if ^{235}U splits into two fission products ^{95}X and ^{139}Y with production of two neutrons as is shown in the scheme (S3):

$$ {}^{235}_{92}U + {}^{1}_{0}n \rightarrow {}^{95}X + {}^{139}Y + 2\,{}^{1}_{0}n \qquad \text{(S3)} $$

Before the fission process total mass of ^{235}U and incident neutron can be determined as a sum of 235.124 amu (^{235}U) and 1.008665 amu (^{1}n) which is equal to 236.132665 amu, where amu is atomic mass unit. After the fission, total mass of fission products and two released neutrons can be determined as a sum of 94.945 amu (^{95}X), 38.955 amu (^{139}Y) and 2.01733 amu (2^{1}n) which is equal to 235.91733 amu. Therefore, mass loss after fission is 0.215335 amu, which is equivalent to 200.818 MeV. Other types of ^{235}U fission produce similar mass loss with average released energy of 200 MeV.

The distribution of released and recoverable energy from ^{235}U fission is shown in Table 11.

TABLE 11. RELEASED AND RECOVERABLE ENERGY FROM ^{235}U FISSION [18]

Form of energy	Released energy (MeV)	Recoverable energy (MeV)
Kinetic energy fission fragments	168	168
Fission products decay:		
- Beta rays	8	8
- Gamma rays	7	7
- Neutrinos	12	—
Prompt gamma rays	7	7
Fission neutrons (kinetic energy)	5	5
Capture of gamma rays	—	3–12
TOTAL	207	198–207

It is evident that not all released energy can be further recovered. The energy of neutrinos cannot be recovered, but due to additional capture of gamma rays in the reactor core, the recoverable energy is almost same as the released energy. Considering a recoverable energy of 200 MeV per ^{235}U fission, the recovery of 1 MW thermal power needs about 3.125x10^{16} fissions/s according to Eq. (61):

$$P = 1\,MW \times \frac{10^6\,joules}{MWs^{-1}} \times \frac{fission}{200\,MeV} \times \frac{MeV}{1.60\times10^{-13}\,joule} = 3.125\times10^{16}\,\frac{fissions}{s} \quad (61)$$

The relation between thermal power and neutron flux can be determined from the following equation:

$$P(t) = \Sigma_f E_r V \varphi(t) \quad (62)$$

where

P: thermal power of a reactor
Σ_f: macroscopic fission cross section
E_r: recoverable energy from ^{235}U fission
V volume of the reactor core
$\varphi(t)$: time dependent neutron flux

When the reactor is in steady state, Eq. (62) can be rewritten as follows:

$$\overline{P} = \overline{\Sigma}_f E_r V \overline{\varphi} \quad (63)$$

where

$\overline{P}$: average thermal power in steady state reactor
$\overline{\Sigma}_f$: average macroscopic fission cross section in a steady state
$\overline{\varphi}$: average neutron flux in steady state reactor

Equation (63) can be reduced to its simple form as follows:

$$\overline{P} = K\overline{\varphi} \quad (64)$$

The measurement of relation between neutron flux and thermal power, or reactor power calibration can be carried out by several methods, but two of them are most frequently used: the calorimetric method and the heat balance method.

10.2.1.1. The calorimetric method

The calorimetric method of reactor power calibration is based on the temperature evolution of the water in reactor pool or tank during standard reactor operation. This method is similar to that applied when determining a conventional non-nuclear heat equivalent in calorimetric system with electrical heater. When operating a RR at a constant power P, the nuclear heat produced through fission results in a temperature increase dT during time interval dt, such as:

$$P = K_1 \frac{dT}{dt} \quad (65)$$

where K_1 is the heat capacity constant. In real exercise at a reactor the differentiation expression in Eq. (65) is replaced by discrete increments in temperature and time, ΔT and Δt respectively, and Eq. (64) is used in the form:

$$P = K_1 \frac{\Delta T}{\Delta t} \tag{66}$$

The heat capacity constant K_1 can be determined from the following equation:

$$K_1 = \rho_w V_w c_{Pw} \tag{67}$$

where

ρ_w: water density

V_w: water volume in reactor pool (or tank)

c_{Pw}: specific heat capacity of water.

An example of reactor power calibration exercise at a 100 kW TRIGA is shown in Fig. 18. It shows that the water temperature increases linearly with time when the reactor is operated at a constant power. The slope of the trace in the graph is the experimental determination of K_1 in Eq. (67).

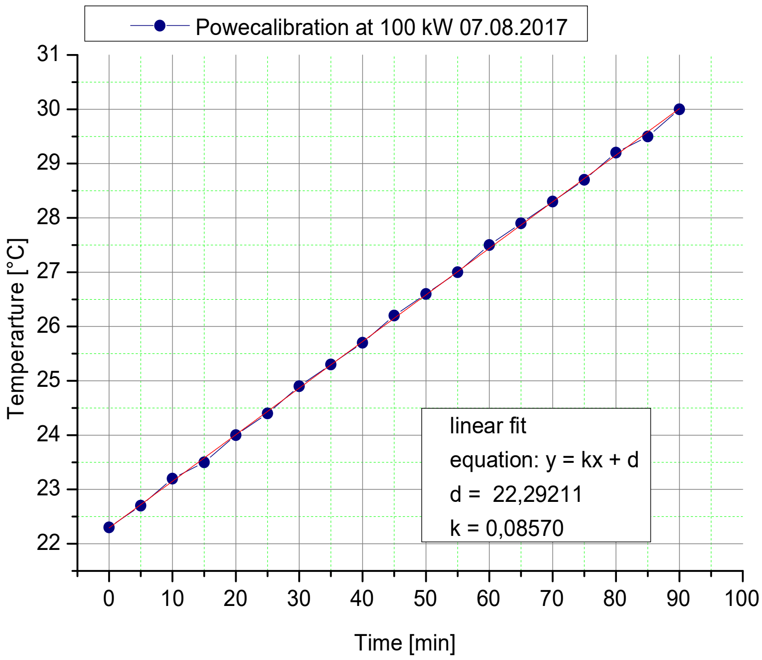


FIG. 18. An example of reactor power calibration exercise at 100 kW. (Courtesy of Atominstitut, Technical University, Vienna, Austria)

10.2.1.2. The heat balance method

The heat balance method of reactor power calibration is mainly used for RRs with thermal power above 1 MW. The method is based on measurement of heat which is removed from the reactor pool (or tank) through the reactor primary cooling system. If the water temperature in a pool (or tank) is close to the air temperature in the reactor hall and to the temperature of the reactor hall ground, thermal balance, i.e. thermal power P of a RR, is given by:

$$P = \dot{m}\, c_{Pw} (T_{OUT} - T_{IN}) \tag{68}$$

where

$\dot{m}$: water flow rate of in the cooling loop

c_{Pw}: specific heat capacity of water

T_{IN}, T_{OUT}: temperature at the inlet to the pool (or tank) and outlet from the pool (or tank) respectively

10.3. EDUCATIONAL ASPECTS

This exercise belongs to the group of reactor physics exercises. The learning objectives and information on suitability for different groups of students are similar to those described in Section 6.3.

Reactor power calibration exercises can be carried out at RRs with power above 50 kW, where increasing the temperatures of moderator and coolant are measurable during reactor standard operation. Usually no specific experimental instrumentation is needed for carrying out reactor power calibration exercises for RRs. A reactor and its standard technology and appropriate neutron detectors, set of thermometers and flow rate meter are necessary to carry out this exercise.

This exercise is appropriate for bachelor's, master's and doctoral programmes. This exercise is suitable for students studying nuclear engineering as the major curriculum as well as for students studying various major engineering curricula. The level of the exercise can be adjusted according to the student's background and pedagogical objectives. It goes from basic to intermediate level.

The two standard methods for power calibration, i.e. calorimetric and heat balance methods, described in Section 10.2 and in this guideline in Section 10.4 are mainly basic level exercises. Duration of the calorimetric method is typically 1 to 1.5 hours. Duration of the heat balance method is typically 2 to 3 hours since it needs to reach equilibrium for the temperature of the core and water circuit.

10.4. EXERCISE ON REACTOR POWER CALIBRATION

10.4.1. Objective

As indicated in Section 10.1, reactor power calibration is an important aspect for the safe operation of nuclear reactors. Indeed, the measurement of the reactor power by means of neutron detectors requires periodic calibration to determine the relationship between the counting rate or the current measured by each neutron detection system and the thermal reactor power.

The objective of this exercise is to establish the thermal reactor power that can be later used to check and to adjust the settings of the neutron detection systems. Additionally, the importance of a proper setting of the neutron detection systems can be discussed in relation to the reactor protection system and the reactivity insertion events. For example, this system activates the reactor shutdown system if the reactor power exceeds the upper threshold allowed for safe reactor operation, i.e. the nominal power plus a fixed margin usually in the 10–20% range.

10.4.2. Equipment and conditions

In order to carry out the power calibration a measurable increase in the moderator and coolant temperatures should be obtained when operating the reactor in a steady state at the nominal power or at a lower power level.

In fact, the K constant in Eq. (64) obtained from the experiment includes the effect of the heat capacity constant and some loss of energy through the system boundaries (pool/tank wall and surface). In actual reactor power calibration, where the pool/tank boundaries are not totally insulated, the K value used in calculating the reactor power may not be the same as given in Eq. (67) by the product of water density, water volume in the pool/tank and specific heat of pool/tank water.

For the calorimetric method, the reactor should be equipped with one or more thermometers to measure the coolant temperature.

For the heat balance technique, the reactor should be equipped with thermometers measuring the coolant temperature at the core inlet and core outlet, as well as with a flow rate meter. Neutron detection systems are also used to control and maintain the reactor power at constant level during the exercises. Reactors are usually equipped with such measuring systems, thus in most cases, no specific additional equipment is needed for this exercise. Nevertheless, additional thermometers may be installed for the exercise in order to have more precise or multiple measurements. The installation of a stirrer in the pool can significantly improve the homogeneity of the pool temperature distribution allowing a more accurate measurement.

10.4.3. Methodology

The calorimetric and heat balance methods described here are applicable to water cooled reactors.

10.4.3.1. Power calibration by the calorimetric method

In this exercise, the reactor power is determined by measuring the increase in water temperature over a given period of time. We assume that the core is in a container (tank or pool) which is filled with water.

Whenever this can be achieved, the experimental conditions should minimize the heat loss from the water container to the surroundings (concrete shielding, secondary pool). To do so, the initial water temperature should be brought to the temperature of its surroundings. The water level in the container should be set to a standard value that corresponds to the standard volume of water for which the heat capacity of the RR has been established.

The exercise is carried out with the water cooling system switched off, i.e. in natural convection. In order to get an accurate measurement of the change in water temperature, which is representative of the global change in water temperature and not of the local change in the vicinity of the thermometers, an appropriate stirrer can be installed in the water container.

The reactor power is increased and stabilised to a given power level. This level should ensure significant water heating, since the accuracy of the calibration depends on the amplitude of the change in water temperature. Thus, at medium power reactors, it is common to carry out the exercise at the nominal reactor power.

The exercise is carried out recording the increase in water temperature T for an interval that ensures a proper determination of the temperature-rise rate $\Delta T/\Delta t$, i.e. typically for one hour. Table 12 or similar can be used to gather the data needed for the experiment. It assumes that three thermometers are installed in different positions of the water container. $\Delta T/\Delta t$ can be determined either through visual fitting on the plot of the average temperature versus time, or using a software to process the data. From a pedagogical point of view, it is advisable that the students plot the curve on a graph paper step by step during the exercise.

From the $\Delta T/\Delta t$ value the students can establish the reactor power and check this measured value with the value given by the reactor I&C systems. The exercise should include questioning about the need for periodic calibration of the I&C system, so that an accurate value of the power is displayed by the system. This usually results in a proper setting of the calibration factors relating the signals given by the neutron detectors and the power. The uncertainty related to the power measurement also needs to be established and discussed.

TABLE 12. EXAMPLE OF A TABLE TO BE FULFILLED FOR THE CALORIMETRIC EXERCISE

DATE:	Time exp. started:		Time exp. ended:	
Water Level				
Cooling System	Off			
Time (min)	T_1(°C)	T_2(°C)	T_3(°C)	Average T(°C)
0				
2				
4				
6				
8				
…				
60				

To complete, the calorimetric exercise, the heat capacity of a reactor may also be measured experimentally. This can be done in two ways:

(a) Heaters with a known power (a few dozen of kW typically) may be installed in the pool/tank to establish the heat capacity through the measurement of the resulting $\Delta T/\Delta t$. For this additional exercise, the reactor should be shut down;
(b) Once the power calibration through the heat balance method is done, the reactor can be set to a different power to establish its heat capacity.

10.4.3.2. Power calibration by the heat balance method

In this exercise, the reactor is operated at a constant power in forced convection, i.e. with a constant cooling water flow rate in cooling systems. When thermal equilibrium is reached, the difference in water temperature at the inlet and outlet of the core is used to establish the reactor power according to Eq. (68).

We assume that the core is in a container (tank or pool) which is filled with water. As for the previous exercise, the experimental conditions should minimize the heat loss from the water container to the surroundings (concrete shielding, secondary pool).

The exercise is carried out with the water cooling system switched on, i.e. in forced convection, with the nominal flow rate. Thermometers measuring the inlet and outlet temperatures should be located at positions ensuring the most accurate measurement of the temperature.

The reactor power is increased and stabilised to a given power level that ensures a significant difference in water temperature between the inlet and outlet, since the accuracy of the calibration depends on the amplitude of the change in water temperature. Thus, at medium power reactors, it is common to carry out the exercise at the nominal reactor power.

Although the determination of the power relies only on the measurement of the inlet and outlet temperature when the reactor has reached thermal equilibrium, recording the evolution of the parameters during the establishment of the equilibrium will make the exercise more dynamic. Thermal equilibrium usually takes between one to two hours to be reached on most reactors. Table 13 gives an example of the table that can be used by the students to gather the data necessary for the exercise. When equilibrium is reached, the value of ΔT and the flow rate are used to establish the reactor power. As with the previous method, the exercise can include discussion about the need for periodic calibration of the I&C system as well as the uncertainty related to this calibration.

TABLE 13. EXAMPLE OF A TABLE THAT CAN BE FULFILLED FOR THE HEAT BALANCE EXERCISE

DATE:	Time exp. started:		Time exp. ended:
Water flow rate			
Cooling System	On		
Time (min)	T_{inlet} (ºC)	T_{outlet} (ºC)	ΔT (ºC)
0			
5			
10			
15			
20			
…			
120			

10.4.4. Safety considerations

The safety considerations given in Section 3.5.4 apply. This Section describes additional safety considerations specific to this exercise.

Usual procedures for reactor operation up to the nominal power are applied. Reactor operation in natural convection for the calorimetric experiment may be subject to some limitation in the operating power. This is related to the maximum temperature of the fuel or cooling water.

When installing additional devices, such as thermometers and stirrer, care should be taken to potential core reactivity induced changes, to the risk of foreign objects falling on the core, and the risk related neutron activation and contamination of the devices.

From a radiation protection point of view, additional risks related to device activation and contamination should also be taken into account.

10.4.5. Documentation

In order to conduct this exercise, the following documents can be given to the students:

(a) Background: neutron and thermal power of a reactor, thermal-hydraulic related to the calorimetric and heat balance methods;
(b) Schematic of the core and pool showing the position of the thermometers, heater and stirrer, according to the exercises;
(c) Step by step procedure to complete the task, including a description of the installation of additional equipment and the tables to be fulfilled with the recorded data. For the calorimetric experiment, the heat capacity of the reactor should be known;
(d) Graph paper or software application to plot the curves;
(e) Specific experimental related information and rules: rules related to the installation of the devices may apply for example.

10.4.6. Question to the students

A first evaluation of the impact of the exercise can be obtained using the following set of questions:

(1) Briefly describe the two methods (calorimetric and heat balance) that can be used for power calibration.
(2) Discuss the uncertainties related to the measurement of thermal power.
(3) What type of experiments can be carried out to establish the heat capacity of a reactor?
(4) Why is it important to carry out power calibration periodically which allows correct setting of the calibration factor that relates the signal of the neutron detection systems and the actual reactor power?
(5) What could be the consequences of a wrong setting of this calibration factor on the protection system of the reactor?

Further evaluation of the impact of the experiment can be obtained through a deliverable such as a report or a presentation on the objectives, methodology and results obtained from these measurements, prepared by the students and delivered to the teacher.

For further information on the reactor power calibration exercises, please consult the bibliography.

11. NEUTRON ACTIVATION ANALYSIS

11.1. BACKGROUND

Neutron activation analysis is a non-destructive analytical technique based on the measurement of characteristic radiation from radionuclides formed directly or indirectly by neutron irradiation of a material. This technique is widely used to determine qualitatively and quantitatively the composition of materials, including the measurement of impurities or trace elements.

NAA is widely used in archaeology, geology, biomedicine, earth sciences, industrial products analysis, nutrition research, health sciences and forensic science, among other areas. The technique allows the analysis of a wide range of objects such as archaeological artefacts, soils, minerals, rocks, atmospheric aerosols, dust in ice cores, tree rings, herbs, plants, human or animal hair, nails, skin and many others. Samples with mass from the microgram to the kilogram range ban me measured.

NAA plays also a significant role in neutron science, where various types of samples such as foils or wires are used for basic or applied research for example for the measurement of nuclear data for various nuclear reactions. In reactor physics NAA is widely used for neutron flux mapping in the cores of RRs (see Section 5.2).

11.2. THEORY

In principle any neutron source with sufficient neutron emission rate or neutron flux such as neutron generators or accelerators, various radioactive neutron sources as well as RRs, can be used for NAA. When a sample of a material under study is loaded into a RR irradiation port, neutrons from the core interact with stable isotopes in the sample and may produce unstable isotopes. These unstable isotopes may decay through gamma ray emission that, after unloading the sample from the core, are measured in a spectrometry system to identify gamma ray lines, which are specific to the radioactive isotopes present in the sample. Through the analysis of the gamma ray lines identified, the original stable isotopes present in the sample are determined, both qualitatively and quantitatively. A schematic arrangement of the main processes in NAA is shown in Fig. 19.

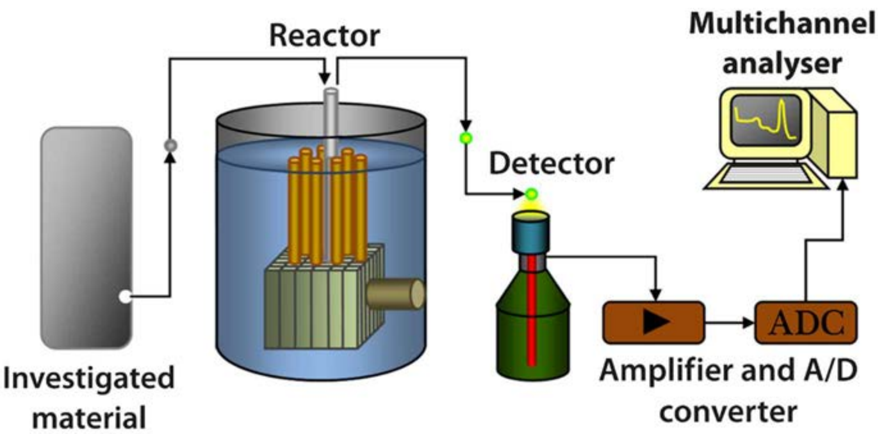


FIG. 19. Schematic arrangement of the main processes in neutron activation analysis. (Courtesy of the Czech Technical University in Prague, Czech Republic)

At the moment of measuring, the activity of each radionuclide can be calculated by the following equation:

$$A(t_M) = \Phi \sigma N_0 (1 - e^{-\lambda t_R}) e^{-\lambda t_M} \tag{69}$$

where

$A(t_M)$: activity, function of time
Φ: neutron flux
σ: absorption cross section of the initial isotope
N_0: number of initial isotopes
λ: radioactive decay constant of the resulting isotope
t_R: irradiation time
t_M: time interval between end of irradiation and beginning of measurement, i.e. $t_M = 0$ when the sample irradiation is stopped

The analysis of the gamma spectrum allows a qualitative determination of the composition of the sample. A radioactive nuclide may be identified from the gamma spectrum using two different types of information:

(1) Energies of the full energy gamma peaks produced by activation;
(2) Half-life of the full energy peak(s) produced by activation.

It is common to make several measurements of the same irradiated sample: shortly after irradiation, a few days and finally a few weeks after, to determine the short, medium and long lived radionuclides, respectively. For pedagogical purposes, the measurement should be done directly after irradiation; in that case, sensitivity is highest for the short lived isotopes.

Starting from the spectrum gathered from the samples (which may be of a complicated composition) it is necessary to search, manually or using software tools, for the isotope exhibiting the right combination of the full energy peak and half-life. For pedagogical purposes, samples with simple composition should be used in order to illustrate the technique in a simple way. Specific application examples can also be used to illustrate the technique, such as the search for Zn in cosmetics or Mn in steel.

A quantitative determination of the composition of complex samples needs a strong theoretical background and experience, so such a determination is usually not conducted as exercises for educational purposes except in the frame of academic or research projects.

11.3. EDUCATIONAL ASPECTS

Exercises related to the application of NAA in various fields are highly suitable for students studying neutron science, neutron applications or nuclear analytical techniques as their major curriculum in all three study programmes — bachelor’s, master’s and doctoral.

The level of exercises can be basic, intermediate or advanced and depends on the knowledge level of the students. The NAA exercises are also suitable for students studying various fields such as archaeology, biology, earth or environmental sciences as the major curriculum in conjunction with minor curriculum in use of nuclear analytical techniques in master’s and doctoral degree programmes. The level of the exercises can be adjusted according to the student’s background and to the pedagogical objectives.

For the basic level, the exercise can be conducted for students with basic knowledge on neutron interaction with matter, radioactivity and radiation detection. With these students, the neutron flux mapping can be carried out as a particular use of the NAA technique. Additional knowledge on reactor physics can then be used to analyse the results obtained about the neutron flux distribution in the core of the reactor. Duration of such an exercise is 1 to 2 hours.

For the intermediate level, the exercise can be conducted for students with additional basic knowledge in gamma spectrometry. In this case samples can be activated at the reactor in order to study the gamma spectrum and establish the composition of simple samples that will be chosen according to the final pedagogic objective: general purpose or specific application such as environmental studies. Duration of such an exercise is typically 3 hours.

For the advanced level, specific applications can be studied. Additional knowledge is needed for a comprehensive analysis of the results of the NAA. Exercises can be conducted in one day, as a demonstration, or in the form of a micro project.

Specific experimental instrumentation is needed for carrying out exercises based on application of the NAA technique at RRs. The reactor is used as the source of neutrons for irradiation, i.e. sample activation. An appropriate gamma ray spectrometry system is needed for NAA. Typical gamma ray spectrometry system consists of a germanium semiconductor detector, a preamplifier, a high voltage power supply, an analogue-digital converter, a multichannel analyser, a computer and software ensuring the gamma ray spectrum analysis and interpretation.

Research reactors are excellent tools for education related to the basic principles of NAA as well as for obtaining practical experience related to this analytical technique and its applications. NAA can be applied using RRs with nominal power of few kW and above. The neutron flux will be a limitation to apply this technique for specific elements. While at low power i.e. low neutron flux, some elements cannot be measured, from a power of approximately 1 MW and above, almost all the elements can be studied.

11.4. EXERCISE ON NEUTRON ACTIVATION ANALYSIS

11.4.1. Objective of the exercise

The objective of the exercise is to learn and to understand the application of NAA in the frame of various nuclear analytical techniques and its use in different fields of science and technology. A clear understanding of basic principles of NAA, its advantages and limitations is a prerequisite to those who are willing to use this powerful technique in archaeology, geology, biomedicine, earth sciences, forensic sciences or others.

11.4.2. Equipment and conditions

NAA can be typically performed on RRs with power above 1 kW due to sample activation limitations. Increasing the reactor power allows to extend the number of elements that can be studied. At 1 MW and above, most elements can be studied.

The reactor should be equipped with a location where the sample can be placed for irradiation. A proper knowledge of neutron flux at this position is important in order to be able to obtain quantitative information through NAA.

In the frame of education, when this is possible, it is pedagogically interesting to have the students involved in sample preparation, loading and unloading it in the reactor and in the measurement process. In addition to these activities, also radiation protection aspects related to the calculation of the foreseen dose rate of the samples and its actual measured value after irradiation can be included.

The irradiation channel or beam port can be itself equipped with a pneumatic transfer system that allows a fast loading and unloading of the sample from the irradiation location. Depending on the maximum speed of the transfer system, it may be possible to search for isotopes with very short decay period (typically less than 1 min). Usually for teaching purposes, isotopes with period of a few minutes are chosen because their decay can easily be studied during the duration of a half a day exercise; and also because using a limited quantity of material and irradiation time, the sample activity can be limited to the values compatible with radiation protection requirements related to the radiation dose rate.

A wide variety of elements and materials can be used in this exercise. Table 14 shows the decay periods and gamma ray energy of some candidate elements to be used as specimen in the elaboration of exercises for an educational programme on NAA. These candidates exhibit decay periods on the order of a few minutes. The quantity of material to be irradiated is generally limited from milligrams to a few grams due to radiation protection requirements.

TABLE 14. USUAL CANDIDATE ELEMENTS FOR THE ELABORATION OF A NAA EXERCISE

Element	Isotope	Period (min)	Energy of the gamma (MeV)
Magnesium	^{27}Mg	9.46	844
Aluminum	^{28}Al	2.32	1779
Titanium	^{51}Ti	5.79	320
Vanadium	^{52}V	3.75	1434
Cooper	^{66}Co	5.10	1039
Silver	^{110m}Ag	0.4	658
Tin	^{125m}Sn	9.5	331
Platinum	^{199}Pt	31	543

For the sample characterization, two different types of instrumentation can be used. For neutron flux mapping, where it is necessary to quantitatively characterize radiation emitted by the sample, this can be done with a precise counting system.

To carry out a comprehensive NAA experiment and identify the isotope(s) present in a sample, it is necessary to use a gamma ray spectrometry system.

11.4.3. Methodology

For the preparation of the exercise, licensed staff is involved in order to guarantee that the experiment protocol incudes the procedures and precautions needed to be conducted in a safe way.

NAA sample preparation, irradiation and measurement are typically carried out according to the following steps:

(a) Set up the gamma ray spectrometry system for the experiment. Standard sources can be used to calibrate the system;
(b) Before performing the exercise, acquire a spectrum of the sample to characterize its initial state of activity;
(c) If possible, i.e. prior approximate knowledge of the sample materials, estimate on the basis of sample mass, neutron flux and spectrum and desired activity the irradiation time needed.
(d) Irradiate, for a pre-determined period of time, the test sample in an appropriate and well-characterised irradiation position. Record the time at which the irradiation started and finished;
(e) Wait the time needed for the sample to decay to level of activity that allows its manipulation and measurement;
(f) Put the test sample in the measurement system. Standard radiation protection procedures are applied when manipulating the irradiated sample and transferring the sample to the measuring system;
(g) Acquire the spectrum of the irradiated sample. Record the starting time and the acquisition duration. Acquisition can be repeated at different period intervals to help identifying the isotopes according to their half-life.

The spectrum is then analysed based on the channel numbers of the full energy peaks and other useful features, such as single escape peaks or Compton edges, as well as the net integral for each full energy peak. The variation with time of the net integral of each peak can be used to determine the half-life of the isotope responsible for the peak.

Using the energy of the peak and the corresponding half-life the radioactive isotopes present in the sample can be identified using different sources, for example references [19–22]. Specific processing software can also be used to identify the isotopes corresponding to each peak.

Then, it is possible to go backwards starting from each radioactive isotope to identify the original isotope(s) before sample activation. This is done using tables giving the natural isotopic abundance and activation cross sections [23, 24].

Additional investigations can concern the error calculation and analysis for both parameters, i.e. half-life and energy.

11.4.4. Safety considerations

The safety considerations given in Section 3.5.4 apply. This Section describes additional safety considerations specific to this exercise.
Special care is needed to ensure safe handling of the irradiated samples. From a radiation protection point of view, additional risk due to sample handling after irradiation as well as during the sample characterization should be considered. It may be necessary to leave some minimum time interval between sample irradiation and sample removal out of the irradiation system. This is very dependent on the sample characteristics.

From a security or safeguard points of view, handling and irradiating nuclear material such as samples containing uranium can be subject to specific rules and restrictions.

The risks associated with this frequent utilization of RRs are generally low. A loss of the irradiation capsule barriers could result in the following:

(a) Radioactive contamination of operating staff and experimenters as well as risk of contamination in the analysis laboratory (in case of loss of integrity of irradiated samples);
(b) Risk of exposure to radiation in case of blockage of the rabbit containing irradiation capsules in a pneumatic or hydraulic tube;
(c) Risk of contamination of internal structures of the reactor in case of an excessive heating of the irradiated samples leading to a loss of their integrity and the potential destruction of the capsules.

The provisions that have to be implemented and the actions to be taken to prevent the above mentioned risks should be indicated in the OLCs and operating procedures. The list of materials forbidden for irradiation in the reactor (such as the mercury because of its corrosive properties) should also be included in the OLCs.

11.4.5. Documentation

In order to conduct this exercise, the following documents can be given to the students:

(a) Background: neutron activation and gamma spectrometry;
(b) Description of the reactor and its irradiation capabilities: core irradiation, beam port, rabbit system;
(c) Step by step procedure for sample irradiation, including radiation protection issues;
(d) Characteristics of the neutron flux in the location of sample irradiation;
(e) Step by step procedure for the setting up and calibration of the detection system, for sample measurement as well as for the analysis of the spectra, that can include the use of an interpretation software;
(f) Tables with the properties of nuclides for the analysis of the spectra.

11.4.6. Questions to the students

A brief evaluation of the understanding of the technique can be obtained using the two following questions:

(1) Explain the principles of the NAA technique and give an example of information that can be gained from this technique for sample characterization.
(2) Explain the isotopes identification procedure and it basis.
(3) Why it is interesting to carry out the sample characterization after different decay times?
(4) What type of isotopes can be measured if we let the sample decaying for a long time?

Further evaluation of the impact of the experiment can be obtained through a deliverable such as a report or a presentation on the objectives, methodology and results obtained from these measurements, prepared by the students and delivered to the teacher.

For further information on NAA, please consult the bibliography.

12. OTHER APPLICATIONS

12.1. BACKGROUND

Various neutron applications widely used at RRs such as neutron radiography, radioisotope production, nuclear and radiochemistry, neutron transmutation, geochronology, among others, have big potential for education of students studying neutron sciences and applications. Students who use these neutron applications in various fields of science and technology, e.g. archaeology, geology, paleogeology, biology, earth or environmental sciences, agriculture, industrial applications, nuclear chemistry, radiochemistry, and others can also be educated at RRs.

Apart from NAA, which is widely used in education, the other neutron applications are sparsely used in educational programmes at the bachelor's level. On the other hand, master and doctoral students who develop their dissertation in one of the applications described in this Section often use a RR for that purpose.

Applications described in this Section can be used to develop exercises related to neutron application that can then be integrated in an academic curriculum. Educational exercises can be developed either as a way to present the experimental techniques and their potential applications, or as a way to perform applied studies or developments with the use of these experimental techniques.

The principles and requirements of each of the applications covered in this Section are only briefly described here. The reader is referred to reference [6] for more details on each application.

12.2. THEORY

12.2.1. Neutron radiography

Neutron radiography, also called neutron imaging, is a non-destructive technique for studying the inner structure and composition of artefacts and samples. Neutron radiography is widely used in various industrial applications and materials research in investigation of alloys, welds, engine turbine blades, electronic components, explosives, hydrogen fuel cells, nuclear fuel and structure of fuel pellets, and many others. It is also used in cultural heritage, archaeology, palaeontology, earth and environmental studies, biology as well as in various other research and scientific applications.

The basic principle of neutron radiography is similar to that of X ray radiography. The neutron imaging technique is based on the absorption and scattering of a neutron beam as it passes through an object. The inner structure of the object can be revealed on a film or on a digital picture from a digital processing system due to different absorption and scattering characteristics of the different inner structures of the object.

Three-dimensional neutron imaging is called neutron tomography. In this case, similarly as in medical X ray computed tomography, the studied object is fixed in a rotation plate and turned around in small steps; a three-dimensional image is created by algorithmic reconstruction from the set of neutron radiographs taken at each step. Fig. 20 shows the schematics of a neutron radiography system.

For scientific and industrial purposes neutron radiography is generally used at RRs with nominal power of several hundred kW and above, where the intensity of neutron beam is greater than approximately 10^5 $cm^{-2}s^{-1}$. Nevertheless, for pedagogical purposes, lower power reactors with neutron fluxes in irradiation field in the range of 10^4 $cm^{-2}s^{-1}$ can also be used to demonstrate the technique and its advantages over X ray radiography.

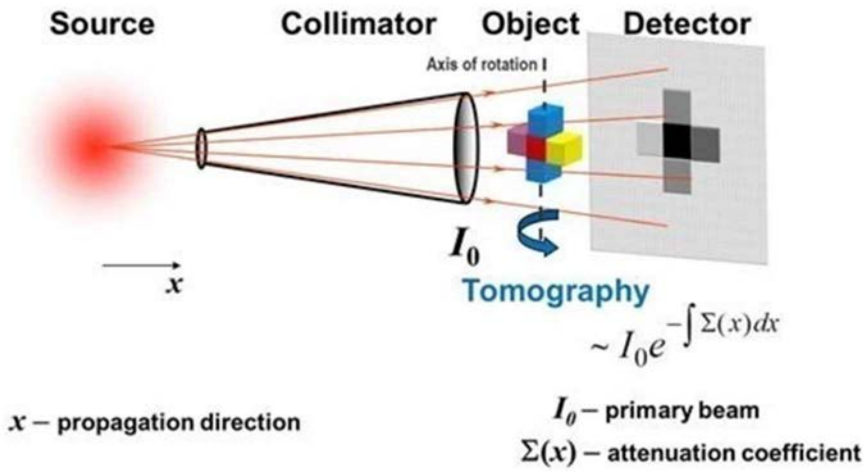


FIG. 20. Schematic arrangement of the main processes in neutron radiography. (courtesy of N. Kardjilov, HZB, Germany)

Because the nature of the interaction with matter of neutrons and X rays is very different, the two techniques are highly complementary. Neutrons are electrically neutral and can easily penetrate deeply into most materials which allows investigating large sized objects. Neutrons interact strongly with hydrogen and other light elements, and neutron radiography is very appropriate for investigation of objects which contain hydrogen, e.g. all organic materials and many technologically important (see Fig. 21). On the other hand, neutrons are weakly absorbed by most heavy elements such as iron and lead, which makes neutron imaging ideal to study the inner structure of artefacts and components with light materials encased beneath metal layers.

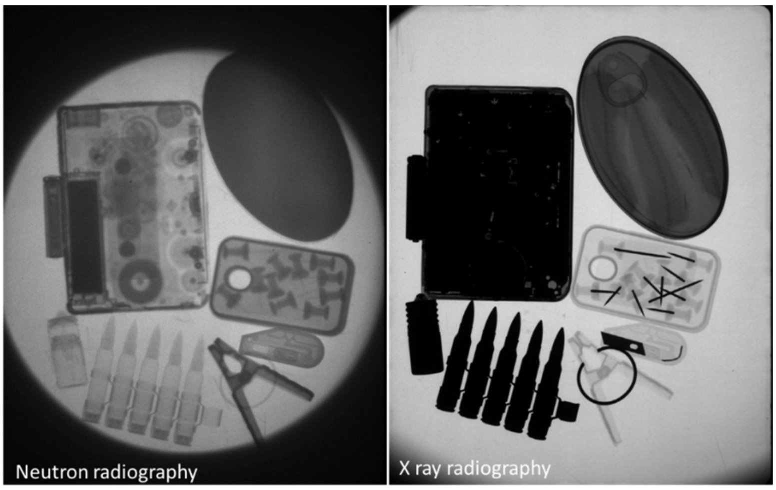


FIG. 21. Comparison of pictures obtained with neutron (left) and X ray (right) radiography. Neutron radiography can be used to investigate objects with hydrogen containing materials. (courtesy of Kindai University, Osaka, Japan)

Advanced neutron imaging techniques using cold neutrons together with energy and polarization selective techniques make possible to visualize the inner crystalline, strain and magnetic structure of samples. Dynamic and real time radiography is also possible.

For further information on neutron radiography, please consult the bibliography.

12.2.2. Radioisotope production and radiotracers analyses

Radioisotopes play a very important role in everyday life. Radioisotopes are used in a wide range of applications in industry, medicine, agriculture, E&T as well as R&D. Radioisotopes in industry are mainly used as a source of gamma radiation in non-destructive testing of welding of pipelines, tubes or various technological parts of industrial processes. In medicine radioisotopes are used for diagnostics, therapy and also for medical tools sterilization. In agriculture radioisotopes are used, for example, for seeds sterilization. Radioisotopes with suitable half-life and suitable specific activity are used in E&T as well as in R&D for various irradiation or calibration exercises.

In all these applications, radiotracers are also commonly used, for example in industrial engineering processing, wastewater purification systems, oil well interconnections and geothermal power characterization. The advantage of radiotracers is their enabling of non-invasive studies of both steady state and dynamic systems in equilibrium situations and for transport and exchange phenomena to provide information on the chemical and physical status of elements.

The most common radioisotopes used for various applications are ^{60}Co and ^{192}Ir which produce strong gamma radiation suitable for non-destructive testing as well as for sterilization of medical tools or seeds. The other radioisotopes typically used in industry are ^{24}Na, ^{47}Sc, ^{82}Br, ^{140}La or ^{203}Hg.

The most common diagnostic radioisotope in medicine is ^{99m}Tc; other radioisotopes typically used in medicine are ^{32}P, ^{89}Sr, ^{90}Y, ^{125}I, ^{131}I, ^{177}Lu, ^{153}Sm, ^{166}Ho, ^{169}Er, ^{177}Lu, ^{186}Re or ^{188}Re.
In RRs, the radioisotopes are produced mainly through (n, γ), (n, p) or fission reactions. Examples of typical reactions used for radioisotope production are shown in Table 15.

TABLE 15. TYPICAL REACTIONS USED FOR RADIOISOTOPE PRODUCTION

Reaction	Example
(n, γ)	${}^{59}_{27}Co + {}^{1}_{0}n \rightarrow {}^{60}_{27}Co + \gamma$ ${}^{191}_{77}Ir + {}^{1}_{0}n \rightarrow {}^{192}_{77}Ir + \gamma$ ${}^{98}_{42}Mo + {}^{1}_{0}n \rightarrow {}^{99}_{42}Mo + \gamma$
(n, γ) & β decay	${}^{130}_{52}Te + {}^{1}_{0}n \rightarrow {}^{131*}_{52}Te + \gamma$ and ${}^{131*}_{52}Te \rightarrow {}^{131}_{53}I + \beta^-$ ${}^{176}_{70}Yb + {}^{1}_{0}n \rightarrow {}^{177*}_{70}Yb + \gamma$ and ${}^{177*}_{70}Yb \rightarrow {}^{177}_{71}Lu + \beta^-$
(n, p)	${}^{58}_{28}Ni + {}^{1}_{0}n \rightarrow {}^{58}_{27}Co + {}^{1}_{1}p$ ${}^{32}_{16}S + {}^{1}_{0}n \rightarrow {}^{32}_{15}P + {}^{1}_{1}p$
Cascade of reactions	${}^{186}_{74}W + {}^{1}_{0}n \rightarrow {}^{187}_{74}W + \gamma$ ${}^{187}_{74}W + {}^{1}_{0}n \rightarrow {}^{188}_{74}W + \gamma$ ${}^{188}_{74}W \rightarrow {}^{188}_{72}Re + \beta^-$
Fission	Short lived fission products ${}^{99}_{42}Mo$, ${}^{131}_{53}I$ Long lived fission products ${}^{137}_{55}Cs$, ${}^{147}_{61}Pm$, ${}^{90}_{38}Sr$

Low power RRs with low neutron flux ($<10^{12}$ $cm^{-2}s^{-1}$), are usually used for production of short lived radioisotopes such as ^{24}Na, ^{38}Cl, ^{56}Mn, ^{64}Cu, ^{41}Ar or ^{198}Au, which can be produced during a one day shift. Medium power RRs with medium neutron flux (10^{12}–10^{14} $cm^{-2}s^{-1}$), which are usually operated in longer cycles, can also produce ^{35}S, ^{51}Cr, ^{60}Co, ^{90}Y, ^{99}Mo, ^{125}I, ^{131}I, ^{133}Xe, ^{153}Sm, ^{169}Yb, ^{177}Lu, ^{186}Re or ^{192}Ir. And finally high power reactors with high neutron flux ($>10^{14}$ $cm^{-2}s^{-1}$) are suitable for production of ^{75}Se, ^{89}Sr, ^{117m}Sn, and $^{188}W/^{188}Re$ or ^{252}Cf.

For further information on radioisotopes production, please consult the bibliography.

12.2.3. Nuclear chemistry and radiochemistry

Research reactors can be used as sources of neutrons and gamma rays in applications of nuclear chemistry and radiochemistry for industry, biology, agriculture, medicine, archaeology, earth sciences, nutrition projects, health projects or forensic science. Various methods of nuclear chemistry and radiochemistry are used to chemically process the samples, specimens, materials or studied objects before or after irradiation by neutrons and gamma. In RRs, nuclear chemistry and radiochemistry are connected with applications such as NAA, radioisotope production, radiotracers analysis or R&D.

12.2.4. Neutron transmutation

Transmutation is a technique where neutrons or gamma rays are used to change the properties of materials in order to create new ones with specific desired properties. Research reactors are essential tools for two transmutation processes based on neutron irradiation of materials — neutron transmutation doping and gemstone coloration.

Neutron transmutation doping, also known as silicon doping, creates impurities in pure silicon ingots during irradiation by thermal neutrons. It is based on the (n, γ) reaction on ^{30}Si followed by beta decay of the unstable ^{31}Si to stable ^{31}P, which is a dopant in silicon as shown in the scheme (S4):

$$^{30}_{14}si + ^{1}_{0}n \rightarrow ^{31}_{14}Si + \gamma \quad \text{and} \quad ^{31}_{14}Si \rightarrow ^{31}_{15}P + \beta^- \qquad \text{(S4)}$$

Other materials such as germanium, gallium arsenide, gallium nitride, gallium phosphide, indium phosphide or indium selenide are also suitable target materials for neutron transmutation doping, but silicon is by far the most commonly used material.

Gemstone coloration is based on the change of colour that some gemstones undergo after irradiation, making them more attractive and more valuable. Irradiation can be performed using fast neutrons or high energy gamma rays. Topaz is one of the main gemstone irradiated in RRs, undergoing change in colour from white transparent to blue. After the irradiation, containers with gemstones are transferred to a storage facility until their activity decreases to an acceptable limit which is in general strictly regulated. In some countries this practice is banned. The time of storage depends on gemstones characteristics, but typically it takes several months.

For further information on neutron transmutations, please consult the bibliography.

12.2.5. Geochronology

Geochronology is for the science of studying the age of geological objects such as rocks, minerals, sediments or fossils. Research reactors are essential tools for two geochronology methods which can be used for absolute measurement of the age of samples; these are argon dating, also known as potassium–argon dating, and fission track dating. These two methods allow determination of age of geological samples from the very young (2000 years old, which allows also the study of archaeological samples) to the very old (up to 4.5 billion years). Geochronology is a very specific application of RRs and is performed in only few of them in the world.

For further information on geochronology, please consult the bibliography.

12.2.6. Neutron beam experiments

Neutron beams experiments bring neutrons out of the reactor tank or pool to the place where the target (or sample) is located. In this case, various horizontal channels, also known as beam ports, are used. Beam ports go through the pool or tank and the reactor shielding out of the reactor block and are usually equipped with shutters, which allow to open and close the ports in requested time during experiments or irradiation.

Usually, all neutron applications which use beam ports are called beam experiments, for example neutron imaging or boron neutron capture therapy. In some scientific communities, the term neutron beam experiments refer to experiments dedicated to studies of material structure, which is a relevant area of application of neutron beams. Clear understanding of the properties of materials and interactions between the various components of materials are essential for effective use of current materials in new conditions or future successful use of new materials which are under development. Material structure investigation can describe ordering relations between atoms or molecules and between electronic and magnetic moments as well as dynamic characteristics of atoms and molecules; or phase correlations between the motions of neighbouring atoms. Neutron beam experiments are closely related to science and technology, for instance condensed matter physics and chemistry, nanotechnology, polymer science, life science, sustainable energy research, sensors and smart materials, biotechnology, spintronics, engineering and archaeology, among other applications.

Neutron beam experiments related to investigation of material structure can study small scale structures of metals and alloys, inorganic compounds and ceramics. In this case various diffraction methods such as powder diffraction, single crystal diffraction, residual stress measurement, crystallographic texture measurement, diffuse neutron scattering measurement or liquid and amorphous materials diffraction are used. The second group of experiments which can use neutron diffraction are related to large scale structures studies of polymers, surfactants, micelles, macromolecules such as biological molecules, and multi-layered solids. In this case typically small angle neutron scattering, ultra-small angle neutron scattering, neutron reflectometry or quasi-Laue diffraction for biology are used. The third group of experiments are related to studies of atomic and molecular dynamics where triple axis spectrometry, time-of-flight spectrometry, backscattering spectrometry or neutron spin echo are used.

Medium and high flux RRs are typically used for neutron beam experiments as a source of neutrons. State-of-the-art facilities are usually found in high performance neutron sources based on a high flux RR or on a neutron spallation source. These often function as user facilities open to national or international research communities. The research done in such cases can be part of PhD programmes.

For further information on neutron beam experiments, please consult the bibliography.

12.3. EDUCATIONAL ASPECTS

Sections 12.2.1 to 12.3 gives an overview of various applications or research fields in which neutrons emitted by the reactor are used either for matter modification or characterization. If these applications are performed in the frame of an industrial application, service or a research activity, the reactor hosting them needs a minimum operating power and some specific equipment, which in some cases has very high installation cost and requires a very high level of expertise. Requirements on the development time and cost and staff required for operation for each technique are given in ref. [6].

Generally, educational activity alone does not justify investing in an equipment to carry out such an exercise, usually a few times per year, in the frame of educational programmes. Thus, when developing an educational programme that includes the study of one of these applications, one should consider two approaches.

A first approach is to get in contact with a facility that is applying the technique on a regular basis. According to the equipment available and to the regular activity performed at the reactor, an exercise or set of exercises can be developed to cover the planned pedagogical objectives. With this approach, traveling to a facility located far from the student's institution and even abroad may be needed.

A second approach is to develop an illustrative exercise that shows the principle of the technique using the available reactor capabilities. In this case the exercise will have to be developed according to reactor characteristics and the existing equipment.

In both approaches, the exercise to be developed strongly depends on reactor characteristics and equipment. As a result, this guideline provides mainly very basic indications and gives some examples of exercises that have been developed at RRs.

The exercises related to the applications described in this Section are highly suitable for students studying neutron sciences and applications as the major curriculum in all three study programmes — bachelor's, master's and doctoral. The level of the exercises can be basic, intermediate or advanced and depends on the level of knowledge of the students.

The exercises are also suitable for students studying various fields of human activities such as archaeology, geology, paleogeology, biology, earth or environmental sciences, agriculture, industrial applications, nuclear chemistry or radiochemistry as the major curriculum in conjunction with the minor curriculum in use of nuclear analytical techniques. The level of exercises in this case is usually basic or intermediate and depends on the level of knowledge of the students.

For the basic and intermediate level, the objectives of the exercise are generally to show the basic and main principles of the application. It can be carried out in the form of a simplified exercise at a reactor not fully suitable for the application or at a reactor carrying out the application on a regular basis. Typical duration of such an exercise can range from 3 to 6 hours.

For the advanced level, the objective of the exercise is to study in detail the application. In this case, the exercise should be carried out on a facility involved in this application on a regular basis. The exercise may have a duration of six hours and more, long exercise should be developed in the form of micro projects by the students, especially in the case of master's or doctoral programmes.

12.4. EXERCISES ON OTHER APPLICATIONS

12.4.1. Objective

The learning objective of exercises related to neutron radiography, nuclear chemistry and radiochemistry, neutron transmutation, geochronology and neutron beam experiments is to understand the basic principles of these neutron-based techniques and their applications. A clear understanding of these exercises provides the necessary basis for students who are studying curricula related to neutron applications. The relation of these neutron applications to various curricula is showed in Section 2.6.

12.4.2. Equipment and conditions

For each application, general aspects related to the equipment needed and particular conditions are given in Sections 12.2.1 to 12.3.

When developing an illustrative exercise, it is necessary to take into account the characteristics and equipment of the reactor and to carry out a basic evaluation of its capabilities for perform the exercise. Additional equipment or modification of some devices may be needed to reach the proper conditions for the exercise. This includes e.g. beam port adaptation, implementation of a system for sample irradiation, additional shielding and installation of specific detection devices.

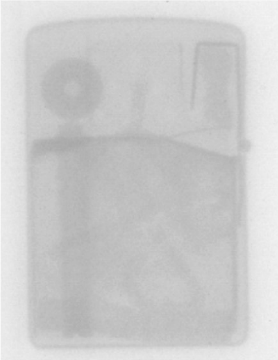
Operating conditions of the reactor and its equipment will have to be defined within the specifications of OLCs.

12.4.3. Methodology

Taking into account the fact that the exercises related to applications of neutron irradiation are broad and very dependent on reactor characteristics and available equipment, this Section provides description of only a few examples from the wide variety of exercises that have been developed at many RRs.

12.4.3.1. Neutron radiography at a low-power reactor (10 W)

Although neutron radiography facilities are generally installed only at medium and high flux RRs, for educational purposes the feasibility of the implementation of a neutron radiography demonstration facility was investigated at the AGN-201K reactor, from the Reactor Research and Education Centre at the Kyung Hee University in Republic of Korea [25]. The AGN-201K reactor has a nominal power of 10 W. The thermal flux level at the exit of beam line from thermal column was investigated using MCNP code. The use of a collimator was evaluated in order to check the feasibility to use the beam for neutron radiography. With enough confirmation on feasibility, a neutron-sensitive image plate and collimator was installed. Fig. 22 presents three shots obtained with the system installed at AGN-201K.



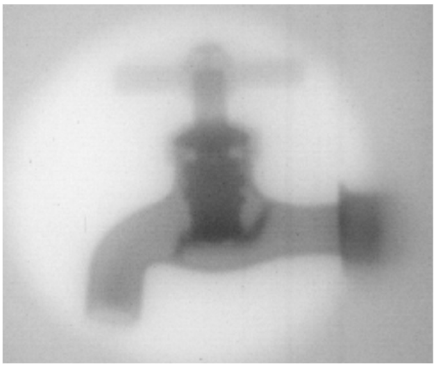


FIG. 22. Images of Neutron Radiography from left to right: lighter, lock pad and faucet. (Reproduced from Ref. [25] with permission courtesy of the European Nuclear Society)

12.4.3.2. Neutron radiography at a high-power (10 MW) instrumented facility

In contrast to the previous example, this exercise related to neutron radiography can also be carried out at a high power RR where neutron radiography is a regular application. This is the case of the Budapest Research Reactor, a 10 MW, light water cooled and light water moderated, tank type reactor, which is equipped with a cold neutron source feeding the Neutron Optics and Radiography for Material Analysis (NORMA) instrument. The NORMA instrument is based on an optical system connected to a digital camera whose principle is shown in Fig. 23.

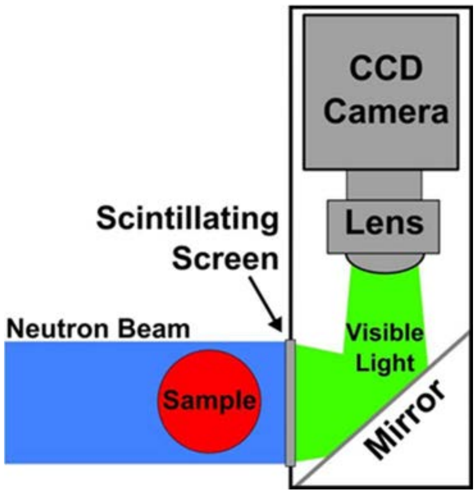


FIG. 23. Schematic of an optical system connected to a digital camera used for neutron radiography. (Courtesy of the Nuclear Analysis and Radiography Department, Centre for Energy Research, Hungary)

After transmission through the sample, neutrons interact with the scintillator, such as *$^6LiF/ZnS(Cu)$*, where they are transformed into a light signal which is processed and finally recorded by light sensitive pixelated camera system.

This facility is used to perform a regular experiment at the BRR facility. The exercise purpose is to:

(a) Study the NORMA imaging facility;
(b) Analyse the time evolution of water absorption process in a pH paper soaked in water by dynamic radiography;
(c) Study and understand beam inhomogeneity and camera noise, and correct them;

(d) Make a movie from the image frames;
(e) Study 3D image processing with an appropriate software.

12.4.3.3. Time of flight spectroscopy and neutron powder diffraction experiments at FRM II reactor

This Section presents two examples of educational experiments that are performed at the FRM II reactor in Munich, Germany. Both educational experiments are carried out with low energy neutrons whose energy and related wavelengths match the dimension of the lattice structure of the sample to be characterized.

In the *time of flight spectroscopy experiment*, thermal neutrons are slowed down to 25 K in a cold neutron source, which is basically a tank containing liquid D_2. With neutron energy matching the energy of atomic motions, quasi-elastic scattering of neutrons with the sample to be analysed can be characterised by time of flight spectroscopy. The amount of energy transferred and neutron scattering angle are the characteristics of a sample. The advantage of this technique is that a huge range of momentums and energy transfers can be captured simultaneously. The experiment carried out at the FRM II reactor is used to study mechanism of molecular self-diffusion, i.e. internal motions of molecules and long-range diffusion processes, using n-alkanes or salt solutions.

In the *neutron powder diffraction experiment*, interference phenomena resulting from coherent elastic scattering of neutron waves with crystalline matter are studied. The facility is equipped with a constant-wavelength high-resolution neutron powder diffractometer whose main components are neutron source, monochromator, sample and detector as shown in Fig. 24.

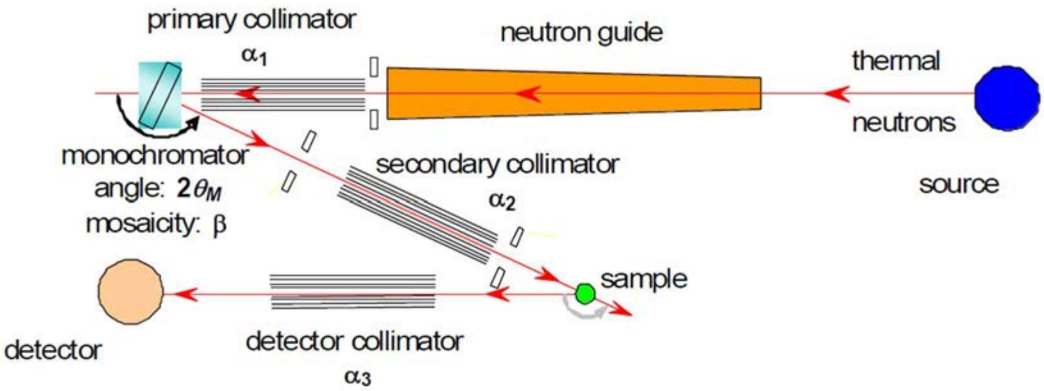


FIG. 24. Schematic of a system used for neutron powder diffraction. (Courtesy of Research Neutron Source Heinz Maier-Leibnitz (FRM II), Technical University of Munich, Germany)

The experiment carried out in the FRM II reactor is related to study of phase and structure analysis of lead titanate, $PbTiO_3$, at various temperatures. The electromechanical properties of lead zirconate titanate, $PbZr_{1-x}Ti_xO_3$, can be understood by their phase transformation behaviour that can be analysed by neutron powder diffraction technique. At high temperatures, $PbZr_{1-x}Ti_xO_3$ exhibit the perovskite structure with simple cubic symmetry resulting in paraelectric behaviour. During cooling, titanium-rich samples undergo a phase transition to a tetragonal phase resulting in ferroelectric behaviour. Zirconium rich samples undergo a phase transition towards a rhombohedral phase. When the Zr/Ti ratios are close to so called morphotropic phase boundary between rhombohedral and tetragonal phase, material shows the highest piezoelectric response, which is its most interesting technological application.

The experiment investigates the temperature-dependent phase transformation behaviour of a $PbZr_{1-x}Ti_xO_3$ sample with a composition on the tetragonal side. Diffraction patterns at different temperature steps between room temperature and 600°C are collected. These experimental conditions are provided by the use of a vacuum high-temperature furnace. The structural changes at different temperatures are investigated by an analysis of the lattice parameters. Based on the experimental data, the relations between structural changes and corresponding physical properties can be discussed.

12.4.4. Safety considerations

The safety considerations given in Section 3.5.4 apply. This Section describes additional safety considerations specific to this exercise.

The safety considerations for specific types of RR utilization are presented below, for teaching purpose or as useful elements for understanding the risks associated with the reactor utilization.

12.4.4.1. Neutron radiography

One risk related to neutron radiography is the potential degradation of the water barrier containing the reactor (pool or tank) because of the damaging of the beam tube. This could result in significant drainage of water leading to uncovering of reactor core and to significant degradation of the fuel in case of high power RRs. Such accident could result in important radiological consequences to the operating staff and the environment. Neutron radiography of explosive materials, which is performed in some RRs, is an example that presents this risk.

12.4.4.2. Radioisotope production

The risks associated with radioisotopes production in RRs includes the risk of contamination and exposure to radiation of operating staff as well as the risk of radioactive releases to the environment in case of a loss of integrity of the irradiated targets, which could occur during irradiations or during handling operations after irradiations. A safety report should be prepared before the first irradiation and should be reviewed by the reactor safety committee and, if relevant, submitted to the regulatory body for review and approval. In a subsequent step, routine irradiations for radioisotopes production which are within the authorized envelope conditions, could be authorized internally by the operating organization. Effective implementation of the provisions to prevent the above mentioned risks should be implemented timely prior to the start of the production.

12.4.4.3. Neutron transmutation

The risks associated with this application is the contamination of operating staff and visitors. Incidents occurred in some facilities where an operator took by hand, without any prior radiological control and without protection, small debris of irradiated silicon left on a table in reactor hall by the operator who was in charge of performing silicon doping operations. Such event highlights the importance of strict application of radiation protection rules and the importance of communication about operational and safety issues between operating staff and experimenters, including students performing exercises.

12.4.4.4. *Neutron beam experiments*

The risks associated with such experiments are mainly related to possibility accidental exposure to radiation of experimenters in working zones near the beam tubes (performing handling operations or setting of their experimental devices) while the neutron beam is present i.e. beam shutter is opened. Such incidents occurred in different facilities and their causes were mainly related to:

(a) Non-application or violations of operating procedures and radiation protection rules;
(b) Deficiencies in the existing provisions for access control in the working zones when the neutron beam is present.

12.4.5. Documentation

In order to conduct this exercise, the following documents can be given to the students:

(a) Background: description of the technique and the basic aspects of the application to be used;
(b) Characteristics of the reactor and equipment related to the exercise;
(c) Schematic of the core and equipment related to the exercise;
(d) Nuclear and radiological safety instructions related to the concerned experimental area;
(e) Specific information related to the exercise and rules to be applied; step by step procedure to complete the task.

12.4.6. Questions to the students

A first evaluation of the impact of the exercises can be obtained using the following set of questions (when applicable):

(a) Briefly, explain the basic principle of the technique to be used and its application;
(b) Explain the main characteristics of the material that is produced and how it is produced;
(c) Briefly, explain the main information that can be gathered from the samples through the application of this specific technique;
(d) Give the limitations of the technique, such as: minimum neutron flux to be used, typical equipment needed, type of sample to be used or characterised, type and quantity of isotope produced, type of information given, and others.

Further evaluation of the impact of the experiment can be obtained through a deliverable such as a report or a presentation on the objectives, methodology and results obtained from these measurements, prepared by the students and delivered to the teacher.

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ANNEX CONTENTS OF THE SUPPLEMENTARY ELECTRONIC FILES

The supplementary material available on-line (https://nucleus.iaea.org/sites/connect/RRIHpublic/CompendiumDB) is a compilation of information and educational material provided by 40 facilities in 31 Member States: Algeria, Argentina, Austria, Belgium, Brazil, Canada, Czech Republic, Finland, France, Germany, Ghana, Hungary, India, Indonesia, Iran, Italy, Jamaica, Japan, Republic of Korea, Malaysia, Mexico, Morocco, Netherlands, Pakistan, Portugal, Russian Federation, Slovenia, Switzerland, Thailand, United States of America, Viet Nam.

This material was collected following an IAEA Technical Meeting on “Research Reactors Utilization for Higher Education Programmes” (June 2014) and a Training workshop on the Compendium on Research Reactor Utilization for Higher Education Programmes (2017). The organizations contributed their material on a voluntary basis to be shared with the community. The material compiled in the supplementary material provides complementary resources for the development of reactor exercises and their protocols.

The different types of documents available on the supplementary material are described below:

A-1. Facility description form presenting the main characteristics of each research reactor that provided educational material for the compendium.

This form provides in a standardized format basic information on the facility, a description of the reactor for educational purposes, features specific to education, a list of exercises performed at the facility, methods of integration of the reactor exercises into curricula and the national programme, as well as additional information relevant to education.

A-2. Collection of protocols for reactor exercises, which were provided by facilities on a voluntary basis.

These protocols are classified according to the following topics:
(a) Technical tour to the research reactor;
(b) Safety aspects of reactor operation;
(c) Reactor start up and operation;
(d) Neutron detection;
(e) Neutron flux mapping;
(f) Criticality experiment;
(g) Reactor kinetics;
(h) Influence of control rods on reactivity;
(i) Influence of core components on reactivity;
(j) Safety parameters related to core reactivity;
(k) Reactor dynamics;
(l) Long term effects;
(m) Reactor power calibration;
(n) Neutron activation analysis;
(o) Applications of neutron irradiation.

The collection of protocols includes two types of documents:

(1) A cover page for each individual reactor exercise that facilities shared with the Compendium. This document provides in a standardized format basic information on the conduct of the exercise at the reactor. The template of this document is included in the supplementary material available on-line.
(2) Extensive protocols for reactor exercises that facilities willed to share in the frame of the Compendium. These protocols have been provided either in their original format, i.e. protocols provided to the students, or in a standardized format.

A standardized format developed by the IAEA contains the following fields:

(a) Introduction;
(b) Purpose;
(c) Safety measures;
(d) Instruments and materials;
(e) Experimental procedures;
(f) Main parameter(s) measured;
(g) Typical data recorded;
(h) Data analysis, assumption and equations;
(i) Pre-knowledge required from students;
(j) Results;
(k) Conclusions.

Its template, which includes a brief description of the expected content of each field, is included in the supplementary material available on-line.

Original protocols provide good examples of documents distributed to students for the conduct and analysis of exercises at RRs. The content of these original protocols is not always comprehensive regarding the fields listed in the standardized format as additional fields may be covered in separate lectures, manuals and books, or be presented verbally. Also, it is noteworthy that students are usually expected to record, analyse and provide conclusions to the exercises with adequate guidance but without being given any indication on expected experimental results.

Standardized protocols provide good resource materials for professors or reactor operating organizations for the development of reactor exercises. Such documents can be used as a resource material to develop protocols to be distributed to students. Assuming that students are expected to record, analyse and provide conclusions to the exercises with adequate guidance, it is advised that such protocols should integrate the following fields: Introduction, Purpose, Pre-knowledge required from students, Safety measures, Instruments and materials, Experimental procedures, Main parameter(s) measured, Guidance to data recording and analysis and Conclusions. Also, the questions to the students given in the last paragraph of each exercise's guidelines in the Compendium text can be integrated with the protocols. Indeed, to take full advantage of the reactor exercise, students are expected to deliver a report, and eventually make a presentation, describing the work done, analysis of the experimental results and conclusions.

A-3. Search engine to filter the protocols according to the research reactor and country or according to the topic of the exercise.

LIST OF ABBREVIATIONS

AGN	Aerojet General Nucleonics
ALARA	As low as reasonably achievable
ARGONAUT	Argonne's Nuclear Assembly for University Training
E&T	Education and training
ER	Excess Reactivity
FWHM	Full width at half maximum
HP	High power
HPGe	High Purity Germanium
I&C	Instrumentation and control
IRT	Standardized Research Reactor (from the Russian Исследовательский Реактор Типовой)
IT	Information technology
LP	Low power
MCNP	Monte Carlo N-Particle
MNSR	Miniature Neutron Source Reactor
MS	Member State
NAA	Neutron activation analysis
NORMA	Neutron Optics and Radiography for Material Analysis
NP	Nominal power
NPP	Nuclear power plant
OLC	Operational limits and conditions
PIE	Postulated initiating event
PSA	Probabilistic safety assessment
R&D	Research and development
RR	Research reactor
RRDB	Research Reactor Database
SAR	Safety analysis report
SCRAM	Safety Control-Rod Axe-Man
SDM	Shutdown margin
SLOWPOKE	Safe LOW-POwer Kritical Experiment
STEM	Science, technology, engineering and mathematics
SUR	Siemens-Unterrichtsreaktor
TLD	Thermoluminescent dosimeter
TRIGA	Training, research, isotopes, general atomic
TTL	Transistor–transistor logic
WWR	Water-Water Reactor

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