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Considerations of Safety and Utilization of Subcritical Assemblies

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CONSIDERATIONS OF SAFETY AND UTILIZATION OF SUBCRITICAL ASSEMBLIES

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INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2021

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Printed by the IAEA in Austria
September 2021

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Considerations of safety and utilization of subcritical assemblies / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2021. | Series: IAEA TECDOC series, ISSN 1011–4289 ; no. 1976 | Includes bibliographical references.
Identifiers: IAEAL 21-01445 | ISBN 978-92-0-128021-3 (paperback : alk. paper) | ISBN 978-92-0-127921-7 (pdf)
Subjects: LCSH: Nuclear reactors — Safety measures. | Nuclear reactors — Utilization. | Accelerator-driven systems.

FOREWORD

Historically, subcritical assemblies (SCAs) have been used to perform pioneering experiments to support the design of nuclear reactors, to determine nuclear data and to validate various reactor physics codes and associated modelling tools. In recent years, there has been growing interest in SCAs from Member States. SCAs have diverse applications in nuclear research and training, including computational code benchmarking, cross-section measurement, detector calibration, the development of innovative measurement techniques and demonstration of autonomous reactor operation. This can be highlighted by the recent developments in scientific programmes in multiple Member States.

SCAs, in comparison to research reactors, have specific design features that are advantageous in terms of safety and utilization, such as subcriticality, low power and small source term. These facilities can thus be categorized as a low potential radiological hazard. The safety features also permit convenient access to experimental utilizations. The safety requirements and recommendations presented in IAEA safety standards for research reactors are applicable to SCAs. Given their low potential hazard, many of these requirements can be applied using a graded approach on a case by case basis.

This publication supplements the IAEA safety standards by providing practical information on safety in the design and operation of SCAs. It also provides information on and examples of utilizing SCAs for various types of research and training experiments. This publication is intended for operating organizations, regulatory bodies, technical support organizations and other organizations involved in the safety and utilization of SCA design and operation.

The IAEA appreciates the contributions of all involved in the drafting and review of this publication. The IAEA officers responsible for the publication were K. Sun, O. Dybach and A.M. Shokr of the Division of Nuclear Installation Safety and N. Pessoa Barradas of the Division of Physical and Chemical Sciences.

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1. INTRODUCTION

1.1. BACKGROUND

A wide variety of subcritical assembly (SCA) designs have been developed worldwide, often with specific objectives, including pioneer reactor physics experiments, training and education, cross-section measurement and computational code benchmark. The applications take advantage of the design safety features of SCAs, namely subcriticality, low power, and small source term, that categorize these facilities as a low potential hazard. In recent years, there has been growing interest in SCAs, which is highlighted by the developments that took place within the scientific programmes in multiple Members States. There is need of a technical supporting document to provide practical guidance and information for safety in the design and operation of SCAs in accordance with the requirements established in IAEA Safety Standards Series No. SSR-3, Safety of Research Reactors [1]. Also, the IAEA publication Nuclear Energy Series No. NP-T-5.3, Applications of Research Reactors [2] does not specifically address utilization of SCAs.

It is noted that in SSR-3, the safety requirements consider SCAs as a type of research reactor. This publication focuses on SCAs, particularly with respect to use of a graded approach, and makes a comparison between SCAs and other research reactors of higher potential hazards for the purpose of additional clarification.

1.2. OBJECTIVE

The objective of this publication is to provide practical guidance and information on safety and utilization of SCAs based on the IAEA safety standards and international good practices. This publication is intended for use by operating organizations, regulatory bodies, technical support organizations and other organizations involved in safety and utilization for SCA design and operation.

1.3. SCOPE

This publication focuses on safety and utilization considerations in design and operation of SCAs. It covers various types of designs and utilization programmes of these facilities. Examples of utilizing SCAs for various applications, including research and training, are also covered. Accelerator driven systems (ADS) and SCAs that utilize homogenous fuel are out of the scope of this publication.

The requirements, recommendations and guidance presented in IAEA safety standards for research reactors (see Refs [1, 3–18]) and other safety related publications (see Refs [19–21]) are applicable to SCAs. Due to their low potential hazard, however, a graded approach can be used to determine the most appropriate way to apply these requirements, recommendations and guidance. This publication takes into account the use of the graded approach in application of the safety requirements for SCAs, including with respect to the safety assessment, safety analysis report, operational limits and conditions, operating programme, safety of experimental devices, emergency planning, preparation for decommissioning, and the interface between safety and security.

With respect to utilization of SCAs, this publication covers strategic planning aspects [22] and different specific applications and experiments [2, 23]. Practical examples from the experience of Member States of applications and utilization of these facilities are provided.

1.4. STRUCTURE

This publication consists of five sections and nine annexes. Section 2 presents general safety and utilization provisions applicable to SCAs. Section 3 describes the safety considerations and Section 4 covers their utilization. Section 5 provides conclusions.

Annex I presents the status of SCAs worldwide based on the IAEA's Research Reactor Database (RRDB). Annex II provides an example of regulations for SCAs in the Russian Federation. Annex III provides examples of passive design safety features of an SCA to be constructed in Ukraine. Annex IV discusses the safety considerations for a TRIGA-fuelled SCA in the Philippines. Annex V presents an example of the application of the graded approach to safety requirements in Italy. Annex VI describes a subcritical graphite pile facility in the USA, along with details of its utilization for educational and research purposes. Annex VII describes an SCA in Canada and its experimental programme. Annex VIII provides an example of using an SCA for training and education in nuclear engineering in Algeria. Annex IX presents a generic methodology for fissile materials loading in SCAs.

2. GENERAL PROVISIONS FOR THE SAFETY AND ULTILIZATION OF SUBCRITICAL ASSEMBILES

2.1. DESIGN CHARACTERISTICS OF SUBCRITICAL ASSEMBLIES

An SCA contains a mass of fissile materials that is typically insufficient to self-sustain a fission chain reaction under normal operation states and design basis accidents. Its operation requires external neutron source(s). Few SCAs are designed to have the potential to reach criticality for accommodating more flexible utilization needs, with subcriticality ensured by engineering and administrative provisions. Most often, an SCA is a lattice of fuel rods or fuel assemblies in a neutron moderator such as water or graphite. Subcritical assemblies may use neutron reflectors, depending on the design and utilization considerations. Some SCAs, designed and operated with fast neutron spectrum, are used for reactor physics benchmarking purposes.

Some important aspects have direct relevance to safety considerations for SCAs, as follows:

- Subcriticality. Para 6.145 of SSR-3 [1] states “The design and construction of the core of a subcritical assembly shall ensure that criticality cannot be reached for any core configuration (fuel, reflector and neutron source, if any), temperatures, moderation and reflection circumstances.” This applies for all operational states and in design basis accidents. SCAs thus cannot sustain a chain reaction without the presence of external neutron source(s). Once the neutron source(s) is removed, the rate of fission reaction and, accordingly, the thermal power decrease exponentially, thus ensuring effective control of reactivity and safe shutdown conditions.

- Low power. Most SCAs are operated at “zero” power (normally in the range of few milliwatts, or at most, a few watts), where heat generation is insignificant. In addition, the low level of fuel burnup suggests that the decay heat is negligible. Most SCAs thus do not require forced coolant circulation for removal of heat generated in the core or from the spent fuel, as applicable. The low power leads to a correspondingly low neutron flux, which practically limits the range of possible applications of most SCAs.

- Small source term. Most SCAs use fresh fuel (or at most, slightly depleted or irradiated fuel). Consequently, the source term to be considered for a radioactive release in the event of an accident is small. Combined with their subcriticality and low power design features, these facilities are categorized as a low potential hazard, with no radiological consequences beyond the building of the facility during normal operation and design basis accidents.

Subcritical assemblies have been built for a variety of uses, the most common being education and training. Subcritical assemblies are also used in distinct research activities, including nuclear data benchmarking to support the development of new reactor concepts. To conduct experimental programmes and to meet technical and regulatory requirements imposed by operating a nuclear installation, SCAs need to be technically supported by specific services, laboratories and ancillary facilities, such as for nuclear material storage, gamma spectroscopy analysis, radiation protection and waste management.

The worldwide status of SCAs is presented in Annex I. The data are derived from the IAEA's Research Reactor Database (RRDB) [24], a database containing technical information on research reactors, including critical and subcritical assemblies in Member States. An example of national regulations for SCAs is provided in Annex II. Examples of passive safety features of an SCA are provided in Annex III.

2.2. EXTERNAL NEUTRON SOURCES FOR SUBCRITICAL ASSEMBLIES

When an external neutron source is introduced into an SCA, the emitted neutrons start to be multiplied in the fuelled region. A steady state of neutron population will be established after a brief period, which is determined by the multiplication factor and the delayed neutron fraction. Some SCAs can operate in a pulsed mode, usually driven by pulsed neutron generators. Overall, the external source is indispensable to the SCA utilization. The following subsections discuss different types of neutron source that are usually used in SCAs. Accelerator driven systems and compact accelerator neutron sources are out of the scope of this publication and are therefore not addressed here.

2.2.1. Radionuclide neutron source

Radionuclide neutron sources are the earliest type of source utilized in nuclear activities. They typically can be divided into three main categories: radioactive (α, n), spontaneous fission and photo-neutron sources. The former two are often used in SCAs. The source material is doubly sealed in stainless steel capsules. Their size is relatively small, with height and diameter ranging from millimetres to centimetres. Their typical intensity of radionuclide neutron source is on the order of 10^6–10^7 n/s.

Radioactive (α, n) source

When an alpha emitting nuclide is mixed with a light element, usually beryllium or lithium, neutrons are produced by the following reaction:

$$ {}^{9}_{4}Be + {}^{4}_{2}He \rightarrow {}^{12}_{6}C + {}^{1}_{0}n + \gamma $$

The most commonly utilized alpha emitters are ^{239}Pu and ^{241}Am due to their long half-lives. The mean energy of the emitted neutrons is approximately 4.5 MeV. The typical neutron emission rate of a radioactive (α, n) source is about 5.4×10^{-5} $n\cdot s^{-1}\cdot Bq^{-1}$.

Spontaneous fission source

Some heavy nuclei decay by spontaneous fission, where neutrons are emitted as by-product. The most commonly utilized spontaneous fission source is ^{252}Cf. The emitted neutrons have a mean energy of 2.3 MeV. The source has a high specific activity of 0.12 $n\cdot s^{-1}\cdot Bq^{-1}$, but its relatively short half-life of 2.6 years means that sources may to be frequently replaced.

2.2.2. Neutron generator

Neutron generators are compact electronic devices containing small linear particle accelerators that produce neutrons by fusing hydrogen isotopes. Commercially available units usually use one of the following fusion reactions:

$$ {}^{2}_{1}H + {}^{2}_{1}H \rightarrow {}^{3}_{2}He + {}^{1}_{0}n $$

$$ {}^{2}_{1}H + {}^{3}_{1}H \rightarrow {}^{4}_{2}He + {}^{1}_{0}n $$

These generators consist of an ion source which produces ionized deuterium gas and a target containing either deuterium or tritium. The deuterons are accelerated to the target where fusion reactions occur and neutrons are generated. The energy of the neutrons produced by the D-D and D-T reactions are 2.5 and 14.1 MeV, respectively. Because the yield is much higher in the latter (typically by more than two orders of magnitude), D-T tubes are more widely used. For both reaction types, shielding against fast neutrons is needed. Commercial neutron generators can usually produce more than 10^9 n/s.

The advantage of neutron generators is that they are free of nuclear materials, they provide an intense quasi-monoenergetic neutron flux, and they can operate in pulsed mode allowing the SCA to operate under both pulsed and continuous modes. Neutron generators can also be switched on and off at any time.

2.2.3. Adjacent research reactor

The driven neutrons for SCA operations can come from other non-traditional sources such as an adjacent research reactor. A desired neutron energy field can be delivered from the research reactor by using a combination of supporting systems, collimators, shutters and filters, delivering the source neutrons to the SCA in either a pulsed or steady state mode depending on its utilization. The source intensity from the adjacent research reactor is design dependent and may vary from case to case. Such a non-traditional setup might introduce an additional risk in terms of a criticality accident during fuel handling at the two co-located nuclear installations.

2.3. PHYSICS OF SUBCRITICAL ASSEMBLIES

Subcritical assemblies, by definition, have an effective multiplication factor (k_{eff}) less than 1, meaning that the neutron production (from the fission source) is smaller than the sum of neutron absorption and leakage. This also means the current generation of the chain reaction contains fewer neutrons than the previous generation, while no external neutron source is in place. Subcriticality is the most important neutronics feature for SCAs.

Operating organizations may perform a comprehensive evaluation of the facility k_{eff} to ensure criticality safety. The analyses may focus on, but not be limited to, the most reactive design configurations under the most reactive operating states. Such k_{eff} is usually also referred as $k_{eff,max}$ for licensing purposes, which can be obtained by calculations. The computational results need to demonstrate a sufficient subcritical margin being maintained, after subtracting the deviations against relevant benchmark tests. Data uncertainties (materials and cross-section) and simulation biases (model simplification and numerical accuracy) may also be taken into account in the subcritical margin. In Annex III and IV, the examples contributed by the Member States have determined k_{eff} of their respective SCAs to be 0.98 (KIPT in Ukraine) and 0.95 (PRR-1 SATER in the Philippines). These serve as part of the licensing basis of the above SCAs. One may note that k_{eff} is an inherent neutronics feature. The value is only associated with the SCA design and is independent of the characteristics of any external neutron sources.

Subcritical assemblies cannot sustain a chain reaction. One or more external neutron sources are needed for either pulsed or steady mode operations. The concept of subcritical multiplication is thus introduced, for describing the phenomenon of amplifying the source neutron (S) to the total neutron counts (M) via the SCA. The magnitude of subcritical multiplication based on the given neutron source(s) can be numerically quantified by the source driven multiplication factor k_{src}, where:

$$M = \frac{S}{1 - k_{src}} \tag{1}$$

Calculation of the source driven mode is an important computational capability for organizations that operate SCAs. For instance, the radiation level in the surroundings during SCA operation can only be accurately predicted by source driven calculations.

Operating an SCA is a case dependent phenomenon. It is necessary to differentiate k_{eff} (an inherent neutronics feature) from k_{src} (source driven subcritical multiplication). The former only represents the facility design characteristics, whereas the latter, in addition to the facility design characteristics, also depends on the characteristics of the external source, such as neutron energy or spectrum, and the relative location between the source and the SCA. Take, for instance, an SCA with $k_{eff} = 0.95$. If a neutron source is placed at its core centre, where the adjoint neutron flux (or the neutron importance) is high, the case dependent k_{src} can be larger than k_{eff} due to better neutron economy. On the other hand, if the same neutron source is placed at its core periphery, where the adjoint neutron flux (or the neutron importance) is rather low, the case dependent k_{src} can be noticeably smaller than k_{eff} due to a pronounced leakage term. The difference can be quantitively described by introducing a parameter $\varphi*$ (source importance), where:

$$\varphi* = \frac{1 - k_{eff}}{k_{eff}} \cdot \frac{k_{src}}{1 - k_{src}} \tag{2}$$

In practice, k_{eff} (or $k_{eff,max}$) determines the upper neutronics bound for SCA licensing purposes; whereas k_{src} is associated with case dependent operating conditions, including the facility design configuration and the external source characteristics. The former is not a measurable parameter. All experiments, including $1/M$ (steady mode) and source jerk (pulse mode), measure k_{src}, because an external neutron source(s) is involved. When an SCA is subcritical by only a few percent, k_{eff} and k_{src} may become numerically similar, as the importance of the neutron source is less pronounced. Under such operating condition, one can assume the experimentally obtained k_{src} is a reasonable estimate of the SCA k_{eff}.

The establishment of a steady subcritical multiplication is a dynamic process. The governing parameters are the mean generation time (with delayed neutrons) and the system subcritical multiplication factor. The former is almost a constant value (~ 0.1 s) for any ^{235}U fuelled system, whereas the latter is case dependent. For a system with more intensive subcritical multiplication, it takes a longer time to reach a steady state after source injection, because more neutron generations are needed to establish an equilibrium neutron population.

3. SAFETY CONSIDERATIONS OF SUBCRITICAL ASSEMBLIES

3.1. APPLICATION OF A GRADED APPROACH TO SUBCRITICAL ASSEMBLIES

Subcritical assemblies have multiple design safety features, including subcriticality, low power and small source term (see Section 2.1). These facilities are categorized as a low potential hazard. Consequently, a graded approach is to be applied to the implementation of the safety requirements: see para. 1.3 of SSR-3 [1]. Paragraph 6.18 of SSR-3 [1] states that “The use of a graded approach in the application of the safety requirements shall not be considered as a means of waiving safety requirements and compromise safety.”

IAEA Safety Standards Series No. SSG-22, Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors [4] provides recommendations on the application of a graded approach for research reactors (including SCAs). The method to determine the graded approach may be quantitative, qualitative or a combination of both. In general, the process presented in SSG-22 [4] consists of two steps: 1) categorization of the facility in accordance with potential hazards and 2) analysis and application of a graded approach.

3.1.1. Step 1: Categorization

According to SSG-22 [4], qualitative categorization of the facility ought to be performed on the basis of the potential radiological hazard. Most SCAs fall into the lowest category, i.e. have no potential radiological hazard beyond the facility building. Additional typical characteristics to be considered in deriving the category of the facility in accordance with its potential hazard are:

(a) The power;
(b) The source term;
(c) The amount and enrichment of fissile material and fissionable material;
(d) Spent fuel elements, high pressure systems, heating systems and the storage of flammables, which might affect the safety of the facility;
(e) The type of fuel and its chemical composition;
(f) The type and mass of moderator, reflector and coolant;
(g) The amount of reactivity that can be introduced and its rate of introduction, reactivity control, and inherent and engineered safety features;
(h) The quality of the containment structure or other means of confinement;
(i) The utilization (experimental devices, tests and reactor physics experiments);
(j) The site evaluation, including external hazards associated with the site and the proximity to population groups;
(k) The ease or difficulty in changing the overall configuration;
(l) The type and intensity of neutron sources.

Table 1 provides specific considerations of the SCA design characterizations in deciding whether the application of certain requirements may be graded.

TABLE 1. SUBCRITICAL ASSEMBLY DESIGN CHARACTERIZATIONS

	Design Characterizations	Considerations for a Graded Approach
a	Power	Very low: Most SCAs have power output on the order of mW–W
b	Source term	Very low: Most SCAs use fresh fuel. Combined with their very low power level, the minimal fuel burnup results in a very small source term. Note that operating SCAs requires external neutron source(s), which needs an appropriate radiation protection programme.
c	Amount and enrichment of fissile and fissionable material	Mostly natural uranium or low enriched uranium; Some SCAs use MOX fuel.
d	Spent fuel elements	Low burnup (≈ fresh fuel)
	High pressure systems	Usually none
	Heating systems	Usually none but possible at low temperature
	Storage of flammables	Usually none
e	Type of fuel and its chemical composition	Uranium metal, U-Al_x, UO_2, UZrH, MOX
f	Moderator	Light water, graphite
	Reflector	Light water, graphite
	Coolant	Usually no forced convection
g	The amount of reactivity that can be introduced and its rate of introduction, reactivity control, and inherent and engineered safety features	Remain subcritical for normal operation and design basis accidents by definition; Ensure safe shutdown under design extension conditions
h	Quality of the containment structure or other means of confinement	Limited confinement structures
i	Utilization	Case dependent, but mainly for reactor physics experiments, training and education
j	Site evaluation	More flexible than research reactors due to low potential hazards
k	The ease or difficulty in changing the overall configuration	Usually easier than research reactors due to more specialized utilization
l	The type of neutron sources	Capsulation of radionuclide neutron source, electrical hazards of neutron generator, safety considerations of adjacent research reactor

3.1.2. Step 2: Analysis and application

Following the categorization of the facility in step 1, analysis is needed to determine the appropriate way of applying specific safety requirements using a graded approach. All the safety requirements that are established by SSR-3 [1] need to be analysed in order to determine the appropriate application to the SCA in accordance with its potential hazard. The requirements of SSR-3 [1] cover the safety assessment and preparation of the safety analysis report, regulatory supervision, management and verification of safety, site evaluation, design, operation and preparation for decommissioning. Recommendations on use of a graded approach in application of the requirements of SSR-3 [1] to SCAs are provided in SSG-22 [4]. Additional insights and practical examples of use of a graded approach in application of these requirements are provided below.

Considerations based on subcriticality

Para 6.66 of SSR-3 [1] states that "For subcritical assemblies, the likelihood of criticality shall be sufficiently remote to be considered a design extension condition." SCAs cannot sustain a chain reaction without the presence of external neutron source(s) under normal operation states and design basis accidents. When the neutron source is removed and properly secured, the rate of fission reaction and, accordingly, the thermal power decrease exponentially due to the physical characteristics of the SCA, thus ensuring effective control of reactivity and safe shutdown of the facility. The shutdown condition is monitored by the instrumentation and control system. There might be some design extension conditions (DEC) that could lead to criticality. Therefore, DEC that could lead to an inadvertent criticality have to be analysed for the purposes of identification and implementation additional design and demonstrative measures to prevent criticality accidents or mitigate their consequences if they occur. An example of these measures is an installation of a reactivity control system, which is one of the basic safety functions required for SCAs: see Requirement 7 of SSR-3 [1].

Considerations based on low power

Removal of heat from the core (and from the spent fuel storage in SCAs with high neutron density and non-negligible burnup) is one of the basic safety functions that is required for SCAs: see Requirement 7 of SSR-3 [1]. Due to the design feature of low power, most SCAs do not need forced coolant circulation in the core and in the fuel storage (if applicable). For example, the requirements for the design of the reactor coolant and emergency core cooling systems can be applied using a graded approach. Some SCAs, such as graphite piles, are air-cooled. The energy generated by fission reactions is low enough to be conducted via the graphite matrix and will not result in significant temperature elevation. Some SCA designs are based on a submerged core in a water container (to allow for more accurate control of the fuel-to-moderator ratio). Such designs allow for cooling via natural convection.

It is important to note also that loss of coolant will result in a more subcritical state of the facility, although this may cause a slightly higher radiation levels as most SCAs contain a small source term and the water coolant is not primarily needed to provide shielding. A graded approach can also be applied to the depth of analysis and the resources needed for performing analysis of loss of coolant accidents in SCAs. For example, conservative assumptions and methods may be used. An emergency core cooling system is usually not needed for SCAs.

Considerations based on a small source term

Confinement of the radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitations of accidental radioactive releases is one of the basic safety functions that required for SCAs: see Requirement 7 of SSR-3 [1]. Combined with their subcriticality and low power design features, these facilities are categorized as a low potential hazard, which is the basis for the application of a graded approach. For example, a graded approach can be applied to the requirements for defence in depth and the design for emergency preparedness and response for SCAs in which accidents need mitigation by the fourth or fifth level might not be physically possible. Another example is use of a graded approach in the application of the requirements on reactor core and fuel design. The SCA core design might not need comprehensive shielding analysis. Neutron reflector and/or gamma shield might not be equipped in some SCAs. Additionally, the radiation protection considerations for the neutron source and experimental activities may become more relevant, given the fact that some SCAs allow fuel handling during operation due to low fuel burnup and insignificant radiation levels. Moreover, the qualification process of the SCA fuel may be simplified due to less demanding operating conditions, for example a small source term, low temperature and no flow oscillation. The long service life of SCA fuel and the relatively frequent handling, however, may involve consideration for ageing management specifically with respect to corrosion or other degradation mechanisms of the fuel cladding material.

Other considerations for applying the graded approach to subcritical assemblies

Despite subcriticality, low power and small source term, the graded approach cannot be applied to some safety requirements.

For example, Requirement 67 of SSR-3 [1] states that "**The operating organization for a research reactor facility shall have the prime responsibility for the safety in the operation of the facility**." The general responsibilities and functions of the operating organization as well as responsibilities, functions, and line of communications of the key positions within the operation organization, apply equally to all SCAs regardless their potential hazards. The application of this requirement by means of staff positions that require a licence or an authorization in accordance with the legal framework of the State (see para. 7.5 of SSR-3 [1]) is not subject to the use of a graded approach. Responsibility for the safety of the SCA cannot be delegated. However, staffing arrangements, the number of operating personnel, the contents and duration of the training programme can be determined by using a graded approach.

Requirement 3 of SSR-3 [1] to establish and implement a safety policy cannot be applied using a graded approach. The safety policy is a central component of an integrated management system, to ensure that any activities across the SCA operating organization place safety as the highest priority.

Requirements 11 and 90 of SSR-3 [1], which relate to the interfaces with security and with safeguards cannot be applied using a graded approach for SCAs, because these requirements are not dependent on the low potential hazard. In fact, more rigorous considerations may be considered, if access for experimenters, trainees, students and visitors is more likely than for higher power research reactors. Additional information is provided in Section 3.7.

Requirements 16 and 29 of SSR-3 [1], which relate to safety classification of structures, systems and components, and the qualification of items important to safety, apply to SCAs regardless of the potential hazard. Thus, these two requirements cannot be applied in accordance with a graded approach.

Requirements 25 and 28 of SSR-3 [1] on single failure criterion and fail-safe design cannot be applied using a graded approach for SCAs. The groups of equipment designed to fulfil the basic safety functions are required to be designed with redundancy, independence and diversity to ensure high reliability, and fail-safe features.

Requirement 52 of SSR-3 [1] on the use of computer-based equipment in systems important to safety, including the verification and validation of computer-based equipment in systems important to safety, cannot be applied using a graded approach for SCAs.

Requirement 63 of SSR-3 [1] on lifting equipment cannot be applied using a graded approach for SCAs. The design of lifting equipment is required to prevent the lifting of excessive loads, prevent the dropping of loads with radiological consequences, permit the safe movement of lifting equipment and permit periodic inspection. Lifting equipment used in areas where equipment important to safety is located, is required to be seismically qualified. In addition, all lifting equipment in SCAs may be designed in compliance with regulatory requirements and national codes and standards.

Example of a graded approach application to an SCA in Italy is provided in Annex V.

3.2. SAFETY ASSESSMENT AND SAFETY ANALYSIS REPORT FOR SUBCRITICAL ASSEMBLIES

3.2.1. Safety assessment

IAEA Safety Standards Series No. SSG-20, Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report [3] notes that safety assessment activities during the authorization (licensing) process for a research reactor, from siting and design to construction and commissioning, can be extensive and continue throughout all stages of the research reactor's lifetime, including operation and decommissioning. SSG-20 [3] also indicates that safety assessment can be conducted in accordance with the potential magnitude and nature of the hazard associated with the particular research reactor or activity. For SCAs, which have low potential hazard in general, a graded approach can be applied to the individual stages in the licensing process. Some processes can be simplified based on the design safety features of SCAs (e.g. subcriticality, low power, and small source term). In addition, some activities in a typical research reactor lifetime (e.g. commissioning programme Stage C: power tests) might not be relevant to SCAs. These are, at least in part, the reasons that multiple Member States choose an SCA as the initial step in the development of their country's nuclear programme. Additional insights and practical examples of SCA specific licensing process are provided below.

Siting and site evaluation for subcritical assemblies

Operating organizations applying the requirements for site evaluation can use a graded approach for SCAs, provided that there is an adequate level of conservatism in the design and siting criteria, to compensate for a simplified site hazard analysis and simplified analysis methods. Most SCAs have a low potential hazard, and accordingly, they may be more suitable for locations such as a university campus or an urban area, in terms of educational and training purposes. Special attentions may be paid to the status of the radioactive inventory at the site, since many SCAs do not incorporate a concrete structure that accommodates the active zone. For example, a water tank type SCA may simply sit on a metallic structure above the ground and a graphite pile is assembled by just stacking up a large number of graphite stringers, including some movable ones (for inserting irradiation foils). The site evaluation needs to take these design features into account as a potential hazard. In addition, the distribution and location of radioactive sources on the site needs to be appropriately addressed in the SCA site evaluation.

Design and construction of facilities for subcritical assemblies

In a typical research reactor licensing process, the construction authorization depends on a design document with satisfactory demonstration of distinct safety objectives and suitable applications of a graded approach. Even though design and construction mutually interact with each other, there is generally a fixed research reactor core configuration established for routine operation. This might not be the case for SCAs. Their core configurations are usually designed for convenient modifications in terms of experimental purposes. In addition, the SCA operating conditions largely depend on the status of external neutron source (e.g. position, intensity and spectrum). In this context, the safety assessment for the SCA design can be conducted based on a bounding analysis, where the core configuration is at the optimal state for neutronics and with the strongest available external source(s) placed in a position possessing the highest neutron importance. The construction can be authorized if the final SCA configuration will not exceed the bounding safety design.

Commissioning of subcritical assemblies

Requirement 73 of SSR-3 [1] requires that commissioning tests are arranged in functional groups and in a logical sequence. They are usually divided into three sequent stages: Stage A (tests prior to fuel loading), B (fuel loading test, initial criticality tests and low power tests), and C (power ascension tests and power tests). The initial criticality tests and low power tests of Stage B and the tests of Stage C are not applicable for SCAs (see footnote 41 of SSR-3 [1]). Instead, relevant tests, such as verification that the configuration is adequately subcritical and measurements of neutron flux, may be performed. These measurement data may also be used to verify the results from computational models and tools that are used for SCA design and safety analysis. In summary, a two-stage commissioning programme (i.e. Stage A plus a modified Stage B) for SCAs may be suggested, in which the stages can be separated by the milestone of initial fuel loading.

Operation, including utilization and modification, of subcritical assemblies

Most SCAs will not be operated routinely, but rather on demand in accordance with utilization, in which training and educational activities usually dominate. It can be beneficial if the operating organization requests approval from the regulatory body for a bounding experimental setup for each type of utilization (e.g. the approach to criticality and nuclear activation), so that more flexibility can be obtained in practice. Similar considerations also apply to the modification of the SCA core. There are advantages if the regulatory approval can be obtained for a bounding modification setup, which is analogous to the conservative design configuration for construction. Once approved, the core modification can be conducted within the envelope.

Another aspect to consider is the systematic periodic safety review (PSR), especially given the fact that most SCAs are not operated routinely and some facilities may be under extended shutdown. It is important to ensure compliance with up-to-date safety standards and to address cumulative ageing effects of SCAs in a timely manner.

Decommissioning of subcritical assemblies and release from regulatory control

The scope, extent and level of detail of the safety assessment of decommissioning and the decommissioning plan may be commensurate with the potential hazard. Most SCAs have a small source term, where nuclear fuel is lightly burned and the contamination (or activation) of the surrounding SSCs is also expected to be low. Accordingly, such SCAs could be disassembled and transported to alternative locations without the comprehensive decommissioning efforts that are necessary for high-power research reactors. The implementation process will also be different. A graded approach may be used in determining the appropriate extent and type and level of details of surveillance and radiation protection measures during transition from operation to decommissioning and release from regulatory control.

3.2.2. Safety analysis report for subcritical assemblies

Requirement 1 of SSR-3 [1] states that "**A safety analysis report (SAR) shall be prepared by the operating organization for a research reactor facility.**" This requirement fully applies to SCAs as do the objectives and regulatory approval process of the SAR, which is required to be periodically updated over the operating lifetime of the facility to reflect any modifications of the physical setup and the regulatory guidelines. SSG-20 [3] provides specific recommendations on preparing the SAR for research reactors. The methodology for the SAR development as well as the scope of the technical content also fully apply to SCAs; therefore, a graded approach cannot be applied. The depth of analysis and the resources needed for performing analysis, however, could be subject to a graded approach in the sense that conservative assumptions and evaluation methods may be used. The following paragraphs aim to provide additional considerations for the SAR development by taking into account the SCA design features.

Development of the safety analysis report for a subcritical assembly

Safety objective and engineering design requirements: All nuclear installations share the fundamental safety objective of protecting people and the environment from harmful effects of ionizing radiation. There is no exception for SCAs. Nevertheless, as a facility having low potential hazard, the design requirements can be applied using a graded approach. In the discussion of general design requirements, emphasis may be placed on the SCA safety features: subcriticality, low power and small source term. Meanwhile, certain specific design requirements might not be applicable to most SCAs, including power oscillations induced by flow instability, provision for reactivity control, provisions for leaktightness of the reactor building and the ventilation system.

Engineered safety features: This SAR chapter can be simplified in terms of the features covering all safety related events, because the anticipated operational occurrences and accident conditions are rather limited for SCAs. Due to their design safety features, most SCAs do not need a reactivity protection system, an uninterruptible power supply or an emergency core cooling system. Some SCAs are equipped with a source removal system and/or draining of the vessel (if applicable). The former is the most effective protection measure for terminating the chain reactions in the SCA core. The latter can lead to a deeper subcritical state of the facility for achieving a safe shutdown condition.

Instrumentation and control systems: Considering the SCA design safety features, the instrumentation and control systems for supporting regular operations and radiation monitoring can be simplified. Neutron flux and/or radiation level may be the only parameters that need to be measured during operation and to perform protective actions during design basis accidents and, if applicable, during DEC. As stated in Section 3.1.2, the requirements for single failure criterion and fail-safe design cannot be applied using a graded approach. Nevertheless, most SCAs do not need a supplementary control room designated for off-site monitoring and control.

Management system: This SAR chapter needs to describe the structure of the operation organization and how a safety committee conducts the advisory role in terms of all relevant aspects of the safety of the SCA and the safety of its utilization, even in the case of a facility having low potential hazard.

Specific considerations on safety analysis

Requirement 5 of SSR-3 [1] requires that a safety analysis of the design of SCAs is conducted by comprehensive deterministic safety analysis and complementary probabilistic analysis, as appropriate. The safety analysis considers the response of the facility to a range of postulated initiating events (PIEs): a comprehensive list is provided in Appendix I of SSR-3 [1]. As discussed in Section 3.1, many PIEs are not applicable to SCAs because of their design safety features. The implementation of the safety analysis can be applied using a graded approach. Among the PIEs relevant to SCAs, criticality and the erroneous handling or failure of a neutron source are worth considering and are highlighted in the following paragraphs.

Subcritical assemblies are designed to be subcritical under normal operation and design basis accident. Paragraph 6.66 of SSR-3 [1] states that "For subcritical assemblies, the likelihood of criticality shall be sufficiently remote to be considered a design extension condition." If inadvertent criticality is considered credible, even if very unlikely, a reactivity control system becomes necessary to ensure the safe shutdown of the SCA. In any case, criticality analysis is expected to demonstrate an adequate safety margin, under the optimal neutronics configuration and under the scenario with highest level of excess reactivity inserted. Nuclear data variance, modelling uncertainty, and statistical error (if applicable) may be excluded from the safety margin. A comprehensive verification and validation study of the computational results may be necessary for the criticality analysis. Exceptions can be made if the SCA is deeply subcritical.

Subcritical assemblies can only be operated with presence of one or more external neutron sources. Most SCAs can be maintained in a safe shutdown condition when the neutron source is removed and properly secured, e.g. via pneumatical ejection, shielding coverage, or neutron generator switched off. The manipulation of neutron sources is frequent for SCA operation. Erroneous handling or failure thus becomes one of the most credible PIEs that may result in radiological consequences. A quantitative evaluation of the resulting doses may be included in the SCA safety analysis. Corresponding procedures may be developed to prevent excessive contamination.

An example of safety considerations for a TRIGA-fuelled SCA is provided in Annex III.

3.3. OPERATIONAL LIMITS AND CONDITIONS AND OPERATING PROGRAMMES FOR SUBCRITICAL ASSEMBLIES

3.3.1. Operational limits and conditions for subcritical assemblies

Requirement 71 of SSR-3 [1] requires that a set of operational limits and conditions (OLCs) important to reactor safety, including safety limits, safety system settings, limiting conditions for safe operation, requirements for inspection, periodic testing and maintenance and administrative requirements, is established and submitted to the regulatory body for review and assessment. IAEA Safety Standards Series No. NS-G-4.4, Operational Limits and Conditions and Operating Procedures for Research Reactors [12] provides recommendations on developing and documenting the set of OLCs. For SCAs, OLCs may be based on their design safety features and on the information from the SAR concerning conduct of operations. Compared to the relatively comprehensive OLCs for other types of research reactor, OLCs for SCAs may have a simplified implementation, since neutron flux and/or radiation level can be the only measurable parameters used for monitoring the SCA operation.

Safety limits for subcritical assemblies

Safety limits are used to protect the integrity of the principal physical barrier that guards against uncontrolled radioactive releases in all operational states and design basis accidents. For most SCAs, the principal barrier is the fuel cladding. The safety limits are often established based on a high temperature that may result in a fuel failure or based on a thermal-hydraulics condition that can lead to a critical heat flux. However, most SCAs have insignificant heat generation and the SCA fuel is not capable of reaching the limiting conditions. In this context, safety limits might not be applicable to the SCA operation.

Safety system settings for subcritical assemblies

The safety system settings are designed to protect safety limits being not exceeded. When safety limits become not fully relevant to SCAs, the requirement for automatic safety actuation can be applied using a graded approach.

Limiting conditions for safe operation for subcritical assemblies

Limiting conditions for safe operation (LCO) are administratively established constraints. Acceptable assurance can be provided if the operation is conducted within a pre-defined envelope. For many SCAs, these administrative constrains can be essential, because safety system settings and safety limits might not be applicable in terms of the safety assurance. NS-G-4.4 [12] gives an example of grouping of LCO topics. The generally applicable groups for SCA operation include a) fuel handling and storage, b) core configuration, c) fuel loading, start-up and operation, d) operational radiation protection, e) instrumentation and control systems and f) experimental devices.

Surveillance requirements for subcritical assemblies

Surveillance requirements can specify the frequency and scope of tests and the acceptance criteria. The objective is to demonstrate the satisfactory performance of the items subject to safety system settings and LCO, which can be applied using a graded approach for SCA applications. The frequency, scope, and depth of surveillance requirements can be reduced accordingly, considering the low potential hazard of the facilities. NS-G-4.4 [12] indicates that some special surveillance requirements may be necessary during an extended shutdown period. Such considerations may apply to those SCAs that do not have a regular operating schedule.

Administrative requirements for subcritical assemblies

Administrative requirements consist of administrative controls concerning the organizational structure and responsibilities, minimum staffing requirements, and actions following an OLC violation. A graded approach cannot be applied to these administrative requirements, regardless of the potential hazard.

3.3.2. Maintenance programme for subcritical assemblies

IAEA Safety Standards Series No. NS-G-4.2, Maintenance, Periodic Testing and Inspection of Research Reactors [7] recommends a maintenance, periodic testing and inspection programme for all SCAs regardless of their potential hazard. The scope, extent of the programme, and the resources for planning, implementation and assessing this programme, however, can be commensurate with the potential hazard and could vary considerably depending on the design, size and complexity of the nuclear facility.

According to SSG-22 [4], there are two steps in determining the provision for inspection, testing and maintenance:

1. The types and frequencies of inspections, tests and maintenance operations needs to be determined, with account taken of the importance to safety of the SSC and its required reliability, and all of the effects that may cause progressive deterioration of the SSC.
2. The provisions to be included in the design and to facilitate the performance of these inspections, tests and maintenance operations need to be specified, with account taken of the frequency, the radiation protection implications and the complexity of the inspection, test or maintenance operation. These provisions include accessibility, radiation shielding, remote handling and in-situ inspection, self-testing circuits in electrical and electronic systems, and software, and provisions for easy decontamination and for non-destructive testing.

Subcritical assemblies usually feature a simple design, where the number of SSCs important to safety is fewer than those in a typical research reactor. The necessary maintenance activities, including service, repair, replacement, testing, calibration, and inspection, are correspondingly fewer. Therefore, a reduced scope and extent of the SCA maintenance programme can be considered.

For SCAs with a low potential hazard, the procedure for a simple maintenance task on a component in a non-active system with low safety significance could be developed by an experienced member of the engineering personnel and reviewed by a maintenance supervisor. The conduct of SCA maintenance activities can be performed by the operating personnel, whereas dedicated and specially qualified maintenance personnel may be needed for high potential hazard facilities. Staffing for maintenance can also be commensurate with the low potential hazard of the SCA.

3.3.3. Ageing management for subcritical assemblies

IAEA Safety Standards Series No. SSG-10, Ageing Management for Research Reactors [6] listed ten ageing mechanisms. Considering the SCA design features, the ageing mechanisms may have different levels of relevance (from low to high) for the SCA ageing management. Additional insights are provided in Table 2.

Except for few recent developments, most existing SCAs were constructed in the early nuclear era. They usually have been in service for more than 50 years. The following paragraphs will highlight the safety considerations of ageing mechanisms with high relevance level to SCAs.

Change of technology

Subcritical assembly design options, types of neutron source, and detectors for low neutron fields might not have significantly changed in recent decades, but some instrumentation, such as the back-end electronics, undergo considerable technological advancements. Digitalization, integration, and portable features can improve the accuracy and reliability of the nuclear measurements, which in turn have a positive effect on the safe operation of SCAs.

Change of regulations

Compliance with the latest international and national safety standards is essential. For example, Requirement 7 of SSR-3 [1] requires fulfilment of the three main safety functions for all states of the facility, including DEC, which were not considered in the safety requirements that were superseded by SSR-3. Fulfilling all the safety requirements and following all the regulation changes, even after applying a graded approach, can be a significant effort for very old SCAs, especially if these facilities are not operated regularly. An effective management system can be of great importance in terms of accommodating changes in regulations.

TABLE 2. TYPICAL AGEING MECHANISMS AND THEIR RELEVANCE TO SUBCRITICAL ASSEMBLIES

	Ageing Mechanisms	Relevance to SCA	Notes
1	Changes of properties due to neutron irradiation	Low	SCAs operate at very low power, i.e., low flux level
2	Changes of properties due to temperature service conditions	Low	SCAs operate at room temperature
3	Stress or creep (due to pressure and temperature service conditions)	Low	SCAs operate at room temperature
4	Motion, fatigue or wear (resulting from cycling of temperature, flow and/or load, or flow induced vibrations)	Low	SCAs operate at room temperature and require no forced coolant convection
5	Corrosion	Medium	No particular deterioration is expected for SCAs, except for usually long serving time
6	Chemical processes	Medium	No particular deterioration is expected for SCAs
7	Erosion	Medium	No particular deterioration is expected for SCAs
8	Changes of technology	High	See above
9	Changes of regulations	High	See above
10	Obsolescence of documentation	High	See below

Obsolescence of documentation

The obsolescence of documentation can be associated with changes in both technology and regulations. An effective management system may again be the most effective solution. Ageing of human resources and loss of knowledge transfer also falls into this ageing mechanism category.

3.3.4. Radiation protection and waste management for subcritical assemblies

Radiation protection programme for subcritical assemblies

The goals of radiation protection programme are to ensure the effective control of external exposure and internal exposure of workers and the public, and of releases to the environment, to ensure conformance with regulatory requirements and to enable further optimization of operational practices [15]. Subcritical assemblies have a low potential hazard; consequently, public exposure due to radioactive discharges or releases during normal operation or design basis accidents might not be a concern. Accordingly, the programme may primarily focus on the radiation exposures within the SCA facility. In general, routine SCA operations might not generate any significant doses in addition to the background radiation, because of its low power level (normally in the range of few milli Watts, or at most, few Watts). Attention may be paid to the handling of SCA fuel and the external neutron source(s), where the risk of exposures is higher due to the frequency of such manipulations. Experimental utilizations, e.g. foil activation and neutron detector measurement, can also result in radiation exposure. The dose limits for workers are much higher than the typical doses received by personnel operating SCAs. The optimization of protection and safety is more relevant to SCA activities in this context.

Waste management programme for subcritical assemblies

Subcritical assemblies are not expected to generate any gaseous and liquid waste. In practice, air ventilation and water circulation systems are not necessary in most SCA facilities. The primary focus of the SCA waste management programme may be cleaning materials (e.g. tissue paper) and laboratory waste (e.g. gloves and glassware). Attention may be paid to the reuse of experimental components (e.g. activation foils and neutron absorbing materials), because most activation products have short half-lives (from minutes to days). An optimized utilization plan may have positive effect in terms of minimizing waste generation. No refuelling and/or fuel discharge is expected for most SCAs prior to decommissioning.

3.4. SAFETY OF EXPERIMENTAL DEVICES FOR SUBCRITICAL ASSEMBLIES

Requirement 66 of SSR-3 [1] states that "**Experimental devices for a research reactor shall be designed so that they will not adversely affect the reactor safety in any operational states or accident conditions.**" IAEA Safety Standards Series No. SSG-24, Safety in the Utilization and Modification of Research Reactors [9] lists general and specific safety considerations for the design of experiments. Most aspects can be applied using a graded approach due to the low potential hazard of SCAs; some aspects might not be relevant because of the design safety features of the SCA. The reactivity related criteria are the key safety considerations. The following aspects may be addressed at the design and operation stages of the SCA experiments:

(a) Total reactivity worth of the experimental devices;
(b) Reactivity effect of non-fixed experimental devices;
(c) Reactivity effect of fast moving experimental devices;
(d) Reactivity effect associated with voided locations (if applicable);
(e) Radiation protection aspects for using specific experimental devices, such as nuclide activation and neutron generators;
(f) Selection of material including compatibility, corrosion, and final disposal aspects.

SSG-24 [9] recommends that the radiological consequences of any experiments should be within accepted limits. Most SCAs have a low potential hazard, and the experiments might not result in higher radiological consequences than the facilities themselves. A specific set of OLCs may be needed for the total reactivity worth of the experiments, so that neither insertion nor removal of the device can result in an unacceptable change in reactivity of the SCA. Experiments might not create any risk for criticality safety. Necessary measurements may be used to verify the a priori reactivity worth calculations before conducting the experiment regularly. The same set of measurement data can also be used to verify the subcritical safety margin for the SCA operation, which cannot be compromised by the experimental devices when they are installed at the most reactive state.

3.5. EMERGENCY PREPAREDNESS FOR SUBCRITICAL ASSEMBLIES

Requirement 81 of SSR-3 [1] states that "**The operating organization for a research reactor facility shall prepare emergency arrangements for preparedness for, and response to, a nuclear or radiological emergency.**" SSG-22 [4] specifies that some aspects of this requirement cannot be applied using a graded approach, such as having sufficient number of escape routes and effective means of communication within the facility. Also, the need to meet national requirements for occupational and industrial safety cannot be subject to a graded approach. A graded approach can, however, be applied to the design of the escape routes and of the communication system to be used during an emergency, due to the low potential hazard of SCAs. For such facilities, which are typically attended by a small number of operating personnel and have all the SSCs located in one or two rooms, the emergency routes and the communication system could be designed in a simplified way.

SCAs, having no radiological consequence expected beyond the facility building, might not need an off-site emergency plan. Accordingly, these facilities might not need emergency response facilities such as an emergency control centre, a supplementary control room or complex equipment for emergency management. Furthermore, emergency response equipment and supplies for SCAs could be stored and controlled in a designated area within the facility building. External resources may be used for emergency preparedness activities, if adequate training is provided that is specific to the SCA design.

The potential radiation risk to SCA workers depends on the facility siting characteristics and experimental utilization. Some SCAs are co-located inside a building with another nuclear installation, in which case the escape routes need to be clearly identified. When an SCA is located amongst other scientific or industrial facilities in the same building, the information and training for operating personnel and users need to be adequate and comprehensive. Some experiments may involve a low level radiological hazard. Appropriate procedures need to be developed during the design stage of the experiment, with all the credible failures evaluated and emergency response prepared. Due to their design safety features, most SCAs do not need an uninterruptible power supply or an emergency core cooling system for emergency preparedness.

Although a remote possibility, for some SCAs inadvertent criticality may be considered as a DEC. The roles and responsibilities for emergency preparedness and response need to be clearly identified within the operation organization and the necessary human and financial resources need to be allocated.

3.6. PREPARATION FOR DECOMMISSIONING OF SUBCRITICAL ASSEMBLIES

Requirement 89 of SSR-3 [1] states that "**The operating organization for a research reactor facility shall prepare a decommissioning plan and shall maintain it throughout the lifetime of the research reactor**". IAEA Safety Standards Series No. SSG-47, Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities [5] provides recommendations on preparing for decommissioning during design and construction stages as well as reviewing and updating the plan periodically. In general, the preparation for decommissioning an SCA is relatively simple compared to the equivalent plan for a high-power research reactor. This is because of the less complex SSCs and the low potential hazard of the SCA. The nuclear fuel of most SCAs is lightly burned and the contamination and activation of the surrounding SSCs is also expected to be insignificant. Accordingly, most SCAs can be disassembled and transported to alternative locations without needing extensive provisions for moving support and for confinement.

Except for few recent constructions, most SCAs are more than 50 years old. It is common that decommissioning might not have been considered at their design stage or during construction and subsequent operation. For these older facilities, planning for decommissioning may start as early as possible once the deficiency has been recognized. Possible modifications to the buildings and SSCs could accordingly be conducted during the remaining operating lifetime, for better preparation for the eventual decommissioning.

Other considerations for preparing for decommissioning are the additional staffing, expertise, and resources needed in the planning and implementation of decommissioning activities, noting that the size of an SCA operating organization is usually limited. Further, it is also necessary to determine the appropriate scope, type and level of detail of surveillance and radiation protection measures during transition from SCA operation to decommissioning and release from regulatory control. Despite the light SCA burnup, the fuel and radioactive waste generated during operation will need to be addressed in accordance with the recommendations provided in IAEA Safety Standard Series Nos SSG-15 (Rev. 1), Storage of Spent Nuclear Fuel [17] and SSG-40, Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors [18].

3.7. INTERFACE BETWEEN SAFETY AND SECURITY FOR SUBCRITICAL ASSEMBLIES

Requirement 90 of SSR-3 [1] states that "**The interface between safety and security for a research reactor facility shall be addressed in an integrated manner throughout the lifetime of the reactor.**" This requirement cannot be applied using a graded approach. In fact, more rigorous interface considerations may be relevant to SCAs, due to the relative ease of physical access for experimenters, trainees, students, visitors. Procedures may be developed to address the access control and identification of such persons.

Reference [19] provides additional insights of the issues, challenges, and general considerations in the safety–security interface for research reactors (SCAs included). Reference [19] highlights the roles of the Government, the regulatory body, and the operating organization, and their responsibilities for safety and security, where the operating organization has the primary responsibility for implementing safety and security regulations and requirements at the SCA facility. An important aspect needed to achieve the highest degree of both safety and security, and to effectively manage the interface between them, is to maintain a culture within the operating organization that emphasizes awareness of both safety and security at the highest levels of personnel and management in the operating organization.

Management of the interface between safety and security is important during all phases of the SCA lifetime, from siting and design to construction and operation, and eventually decommissioning. During a period of extended shutdown (many SCAs are not operated regularly due to lack of a utilization programme and staffing), the number of personnel at the SCA may be much lower than during operation. This may lead to vulnerabilities in terms of safety and security, or their interface. Specific measures that differ from those applicable to the normal operation and decommissioning phases might be considered.

4. UTILIZATION OF SUBCRITICAL ASSEMBLIES

This section outlines the main areas of utilization of SCAs. Reference [2] presents descriptions of the typical research reactor applications, outlining criteria and minimum requirements to enable an application to be performed. A simplified research reactor capability matrix is presented in Ref. [2] to assist in the determination of the various applications that may be appropriate for a research reactor of a particular power level, as well as providing information on the time needed to develop a given application, on the associated investment costs and the level of staffing needed for the application. According to Ref. [2], research reactors in the lowest power category (i.e. <1 kW, corresponding to the IAEA Research Reactor Database (RRDB) [24] categorization of low power research reactors), are utilized primarily for education and training (E&T). They also have other capabilities, e.g. for research and development (R&D), for demonstration purposes, for neutron activation analysis (NAA) and for testing of instrumentation and control systems.

Subcritical assemblies, to some extent, can be utilized for similar purposes as low power research reactors, including experiments (e.g. neutron flux mapping, approach to criticality, reactivity measurement), instrumentation tests, computational code benchmarking and verification of subcriticality of fissile materials. Kinetics experiments can also be conducted in many SCAs utilizing a moving neutron source or a movable fuel design. Historically, a number of SCAs have been built specifically to conduct research and to provide data for supporting the development of new research reactor concepts and nuclear power programmes. Many newer designs are dedicated to E&T purposes, with other applications still playing an important role. In SCAs coupled to an intense neutron source, high neutron fluxes can be achieved, making a wider range of applications possible.

Table 3 shows the number of SCAs involved in each utilization area, according to the RRDB. Some SCAs, in particular several decommissioned ones, do not have information about their utilization programme in the RRDB. The following activities were grouped under “R&D”: Innovative nuclear research, benchmarking and modelling, detector design and testing, fuel and control rod tests and diverse physics measurements and experiments, including reactivity studies and subcritical physics experiments, research in lattice physics and other activities.

4.1. STRATEGIC PLANNING FOR SUBCRITICAL ASSEMBLIES

Research reactors need to have effective and achievable strategic plans (SP) to support their long-term sustainable utilization. In publication Nuclear Energy Series No. NG-T-3.16, Strategic planning for research reactors [22], information is provided on how to develop an SP for both existing and planned research reactors, including SCAs.

One essential point when developing an SP is the identification of existing and potential stakeholders, and assessment and prioritization of their needs, in order to adjust the capabilities of the facility to the users’ needs. Subcritical assemblies that were built to support a specific reactor concept have a clear stakeholder (the promoter of the new concept) with a clear need. When the need has been fulfilled, the SCAs transitioned to extended shutdown status and/or were decommissioned. In such cases, long-term sustainable utilization is not a consideration and developing an SP for the SCA might not be needed. For SCAs that were built to develop capacity in nuclear science and technology, sometimes as the first nuclear installation in the country, the considerations in Ref. [22] are fully applicable, and a thorough assessment of stakeholder needs leading to a justification (or otherwise) to build or continue to operate an SCA may be made.

TABLE 3. UTILIZATION OF SCAs ACCORDING TO THE RRDB

Application	Total	Under construction	Operational	Extended or permanent shutdown	Decommissioned
Education	17	2	10	4	1
Training	8	2	6		
NAA	5	1	4		
Materials/fuel irradiation	2	1	1		
Radioisotope production	1	1			
Neutron therapy	1	1			
Nuclear data measurements	6		3		3
R&D	11	2	5	3	1

4.2. UTILIZATION OF SUBCRITICAL ASSEMBLIES FOR EDUCATION AND TRAINING

The main purpose of most SCAs is to provide hands-on practical education and training to reactor operators, radiation protection personnel, industrial partners, students of different disciplines, and other persons. While education is conducive to an academic degree, training is part of professional development. In practice, the activities carried out in SCAs are similar for both cases. Public tours and visits are also considered part of education and training [2].

4.2.1. Utilization of subcritical assemblies for education

Education using SCAs is directed at nuclear science and technology students. The aim is to provide the students with an opportunity to verify the predictions of reactor theory for the behaviour of the reactor in the subcritical state by performing experimental measurements of basic nuclear reactor parameters. This includes the spatial distribution and energy spectrum of the neutron flux, approach to criticality, and measurements of reflector and moderator effects. Typical experiments that can be conducted at SCAs are shown in Table 4.

TABLE 4. TYPICAL SUBCRITICAL ASSEMBLY EXPERIMENTS

Experiment	Method
Approach to criticality	Varying number of fuel rods, Moderator levels
Reactivity and multiplication factor quantification	Rossi- & Feynman-α, Source-Jerk methods
Flux measurement & mapping	Neutron detection with foils, small size gas detectors, thermoluminescent detectors, micro-scintillators
Measurement of moderator and reflector effects	Varying amount of reflector and/or moderator; change of temperature; varying void fraction;

These experiments are selected from many examples where SCA operations can provide hands-on experience for practical education and training that involves the interaction of students' theoretical knowledge and the corresponding necessary practical competencies and skills. Details of the experimental devices, preparation and procedure vary from facility to facility.

4.2.2. Utilization of subcritical assemblies for training

The experiments performed at SCAs can also be part of a training programme for personnel such as reactor operators, radiation physicists and technicians and radiation protection personnel. This is often a more effective use of resources than to perform the same or similar experiments in a research reactor, considering cost of operation, number of staff needed, and the impact on other uses of the research reactor.

4.2.3. Typical experiments performed at subcritical assemblies

Initial Start-up and Approach to Criticality Experiment

Education and training programmes at SCAs often start with the safe and reliable start-up of the assembly, accompanied by a preliminary understanding of safety characteristics and subcritical state of the assembly. Approach to criticality can be part of this experiment by changing the amount of fuel or absorption, reflector and/or moderator material.

Measuring Reactivity Experiment: Source-Jerk, Rossi-α and Feynman-α methods

The Source-Jerk method can be one of the most accurate and convenient methods for measuring the negative reactivity of subcritical assemblies. To perform this experiment, the facility needs to be equipped with a mechanism for prompt external neutron source removal such as a fast pneumatic transport system to instantly remove the source from the reactor core after the neutron counts reach an equilibrium state. This method is based on the study of the transient response of the reactor to the rapid removal of the source to derive the subcriticality of the SCA. With this method, no extra experimental equipment is needed and no criticality safety issues are created.

Furthermore, measurement of the reactor dynamic parameter α with the Rossi-α or Feynman-α method can be directly correlated to the subcritical state of the SCA. These are advanced methods not usually employed in introductory courses. The Rossi-α method has proved to be valuable for the determination of prompt neutron lifetimes in fissile assemblies having known reproduction numbers at or near delayed critical. For facilities that are deeply subcritical, the Feynman approach is usually more precise.

Absolute flux measurement and mapping experiments

Flux measurements experiments are designed to measure absolute flux values, whereas flux mapping experiments use relative values proportional to the detector responses. The purpose of absolute flux measurements is usually to measure the absolute thermal neutron flux of the SCA. The ratio of thermal to fast neutron fluxes can also be measured using threshold detectors such as activation foils or miniature fission chambers.

The neutron measurement method has to be chosen carefully in SCAs, due to their low neutron flux levels. A proportional counter could be an appropriate choice for measurement of fluxes in positions outside the core. Boron compensated chambers and miniature fission chambers are typically used in-core.

Several methods can be used to conduct the relative neutron flux measurement on an SCA. The nature of the method is chosen to fit the neutron flux values. The foil activation technique can be utilized to measure the relative flux axial and radial distributions. It is noted that the direct measurement is associated with the neutron reaction rate rather than neutron flux. This can also be used to introduce students to different methods of neutron detection and compare their characteristics.

4.2.4. Utilization of subcritical assemblies for public communication purposes

Public communication can be considered to be part of education and training activities. It has, however, a different purpose, not being conducive to academic degrees or professional certification. The methods employed are also different.

One of the important roles of nuclear installations such as research reactors or SCAs is to promote the peaceful use of nuclear energy and ionizing radiation. Subcritical assemblies are very suitable for such purposes. Subcritical assemblies can be used in outreach actions to promote the peaceful use of nuclear energy and ionizing radiation, mainly by hosting visits to the facility from members of the public. These visits are intended to improve public relations and demonstrate the safety and reliability of the nuclear facilities and explain the significance of the peaceful use of nuclear energy.

The promotion activities can be based mainly on a field trip and facility walkthrough, during which the facility operation, experiments and other activities can be demonstrated. The fuel, experimental equipment and other component can also be shown.

During the activity, it is advisable to explain the basic aspects of the peaceful use of nuclear energy and ionizing radiation. It is important to promote not only the benefits, but also to inform about the possible risks and how to minimize them. This places significant demands on personnel of the facility who needs to have good communication skills and also to be able to explain the issue simply and clearly.

The existence of a changing room for preparing visitors before entering the facility and of sufficient space around the facility to allow safe movement of visitors is necessary. For safety and security reasons, the number of visitors at the facility may be limited.

Examples of application of SCAs for training and education are provided in Annexes VI-VIII.

4.3. UTILIZATION OF SUBCRITICAL ASSEMBLIES FOR BENCHMARK EXPERIMENTS

Reactor physics and shielding benchmarks are expected to play important roles in reactor design, safety analysis and in the validation of analytical tools used to design reactors. For existing reactor technology, benchmarks are used to validate computer codes and test nuclear data libraries as well as for evaluating nuclear data uncertainties [25–27].

The critical experiments performed at research reactors are primarily a standard source of experimental data for the verification and validation of computer codes. Establishing benchmarks in operating SCAs can still be very useful due to the large diversity and specific technological features of research reactors. In fact, many SCAs have been built specifically to study existing or planned research reactors, including establishment of benchmarks for the associated reactor calculations. Subcritical assemblies provide flexible core configurations. This means, for instance, that the fuel pitch can be variable, various materials (absorbers, moderators, reflectors) can be added to the core and the amount of moderator (e.g. water level) can be changed.

The main limitation of the usefulness of SCAs for benchmarking is the accuracy of the determination of k_{eff}, which may be poor if the system is deeply subcritical, and also due to the influence of a point neutron source that distorts the flux distribution as compared to the critical state in a “zero” power research reactors.

4.4. UTILIZATION OF SUBCRITICAL ASSEMBLIES FOR VERIFICATION OF SUBCRITICALITY

The knowledge of the degree of subcriticality is a fundamental aspect of criticality safety of fissile material storage and transport configurations. This is typically demonstrated through extensive analyses relying on calculation tools used to predict the effective multiplication factor k_{eff} of the analysed system. Increasingly sophisticated calculation tools are available to perform such analyses. As discussed in Section 4.3, SCAs can provide benchmarks for code validation and verification [28, 29].

Subcritical assemblies can also serve for direct justification of criticality safety and radiation protection aspects of fissile material configurations. The justification process can be based on determination of the degree of subcriticality of fissile material configurations.

The influence of fuel pitch or flooding of fissile material configurations with water on reactivity can be studied experimentally. In addition, the effect of specific absorbing and shielding materials such as boron-containing stainless steel, aluminium boron alloy, borated and lithium polyethylene, iron or cast iron can be explored.

4.5. UTILIZATION OF SUBCRITICAL ASSEMBLIES FOR NEUTRON ACTIVATION ANALYSIS

Neutron activation analysis is a quantitative and qualitative method for the simultaneous determination of a number of main, minor and trace elements in different types of sample [2, 30].

Common neutron sources used are research reactors with power ratings starting at 30 kW with a maximum thermal neutron flux starting around 5×10^{11} neutrons/(cm^{2}s). High sensitivity NAA induced by epithermal and thermal neutrons is reached for neutron fluxes around 10^{13} neutrons/(cm^{2}s).

The thermal neutron fluxes in subcritical assemblies are generally several orders of magnitude lower than the optimal flux range. Nevertheless, applications of NAA are possible, especially for daughter radioisotopes with relatively short half-lives (less than a few days). Sensitivity in this case is negatively affected by the low flux. Elements such as Au, In, Cu, Ag, Mn, Eu, Sm, Ba, Sr and Cs can be measured, by using long irradiation and counting times and a low background detection system.

4.6. UTILIZATION OF SUBCRITICAL ASSEMBLIES FOR RADIOISOTOPE PRODUCTION

Although SCAs typically have low flux, they can still be used to produce small amounts of short-lived radioisotopes for training and education applications. Suitable target materials, ideally high purity materials with high neutron cross-section, can be introduced in the core and irradiated for a specified amount of time. Examples of radioisotopes that can be produced are ^{116}In, ^{198}Au, ^{56}Mn, and others. Another application is the activation of carrier materials in analytical radiochemistry. Table 5 shows examples of such isotopes [31, 32].

TABLE 5. RADIOISOTOPES THAT CAN BE PRODUCED BY TYPICAL SUBCRITICAL ASSEMBLIES

Target	Radionuclide	$T_{1/2}$	Production yield
^{153}Eu	^{154m}Eu	46 m	Very good
^{154}Sm	^{155}Sm	22.1 m	Very good
^{130}Ba	^{131m}Ba	14.6 m	Good
	^{131}Ba	11.8 d	Low
^{84}Sr	^{85m}Sr	67 m	Good
^{133}Cs	^{133m}Cs	2.91 h	Good

Typical fluxes in production processes of radioisotopes for medical diagnoses and therapy are higher than 10^{13} neutrons/(cm^{2}s) which are not commonly produced by SCAs. Accelerator driven SCAs may reach fluxes high enough to be suitable for commercial production of medical radioisotopes, such as ^{99}Mo. These are not within the scope of this publication.

4.7. UTILIZATION OF SUBCRITICAL ASSEMBLIES FOR MEASUREMENT OF NEUTRON SPECTRA

Measurement of neutron energy spectra is important for several applications, such as determination of transmutation cross-sections. Such measurements can be performed in an SCA by an (n,γ) foil activation. It has advantages compared to other techniques, due to the usually high cross-section of these reactions for thermal neutrons. These advantages include high precision and low sensitivity to other kinds of radiation [33].

In the energy region $E < 30$ keV the following foils are some of those that can be used, where E_{res} is the energy of the main resonance (the given mass values are only indicative):

- ^{115}In (E_{res} = 1.457 eV, m = 0.12 g);
- ^{196}Au (E_{res} = 4.906 eV, m = 0.06 g);
- ^{183}W (E_{res} = 18.8 eV, m = 0.39 g);
- ^{56}Mn (E_{res} = 337 eV, m = 0.04 g).

Unfolding of neutron energy spectra in the energy region E > 30 keV can be performed on the basis of the method for utilizing the effective cross-sections of threshold reactions that was developed for the measurements of neutron spectra.

Although other possibilities exist, solving the Fredholm integral equations set of first kind [34, 35] has the advantage of not requiring a knowledge of the reference spectrum in cases where only the ratio of cross-sections is needed. For the measurement of the neutron spectrum in a subcritical system, some of the reactions that can be used are: ^{111}Cd (n,n′) ^{111}Cd^{m}, ^{115}In(n,n′)^{115}In^{m} , ^{55}Mn(n,α)^{52}V, ^{204}Pb(n,n′)^{204}Pb^{m}, ^{90}Zr(n,p)^{64}Cu, ^{58}Ni(n,p)^{58}Co, ^{59}Co(n,p)^{59}Fe, ^{65}Cu(n,p)^{65}Ni, ^{27}Al(n,p)^{27}Mg, ^{24}Mg (n,p)^{24}Na, ^{48}Ti (n,p)^{48}Sc, ^{56}Fe(n,p)^{56}Mn, ^{59}Co(n,α)^{56}Mn and ^{27}Al(n, α)^{24}Na.

The neutron spectrum combined with specially designed experimental techniques can provide information about transmutation rates of radioactive nuclides in neutron spectra with different hardness. For example, using cadmium containers in SCAs where the subcriticality level is close to a fraction of β_{eff}, can provide the transmutation rates of fission products and minor actinides in spectra with resonances and fast neutrons.

4.8. UTILIZATION OF SUBCRITICAL ASSEMBLIES FOR CROSS-SECTION MEASUREMENTS

Solid state track detectors are widely used for detection of heavy ions, mainly fission fragments, for the measurement of spatial distribution of fission rates of various isotopes, for cadmium ratio and spectral indices estimation. Using the experimental distributions of track densities it is possible to obtain spectral conversion factors, $\overline{\sigma}_c(^{232}Th)/\overline{\sigma}_f(^{235}U)$, $\overline{\sigma}_c(^{238}U)/\overline{\sigma}_f(^{235}U)$ and others, as well as the distribution of ^{235}U, ^{238}U, ^{233}U, ^{239}Pu fission rates in the core of SCA. Other neutron detectors used for this purpose include miniature fission chambers.

4.9. UTILIZATION OF SUBCRITICAL ASSEMBLIES FOR DEVELOPMENT OF CORE START-UP PROCEDURES

The most important stage of putting an SCA into operation is the fissile material loading pattern and its loading sequence. The subcritical margin needs to be closely monitored during this stage.

The setting of operations for the loading procedure of the SCA (corresponding to the critical loading in a critical assembly) is well studied. However, the loading procedure is specific for every facility depending on the core design. Usually the reciprocal counting method is used. The basic rule of loading of subcritical (and critical) systems is based on the safe run of the feedback count curve:

$$m = f\left(\frac{1}{N}\right) \tag{3}$$

where: m – mass of fissile material,
N – the counting rate of the monitor detector.

The loading procedure demands the solution of specific problems. It is may be technically challenging, as the reactor operator can either miss the most optimized fuel/moderator/neutron source geometry or misplace the monitor detector according to a heterogeneous flux distribution so that its efficiency is not optimal. It is sometimes necessary to return to a previous state and search for a better configuration.

The commonly used fissile materials loading methodology is presented in Annex IX.

4.10. UTILIZATION OF SUBCRITICAL ASSEMBLIES FOR RESEARCH ON THERMAL AND FAST NEUTRONICS AND KINETICS

Dedicated subcritical (as well as critical) facilities can be used to study neutronics and kinetics to support the development of innovative nuclear energy systems such as Generation IV fast reactor projects [36, 37]. It is possible to create different neutron spectra in the SCA and use different neutron sources to study neutronics of thermal and fast reactors. Subcritical assemblies may have advantages from the point of view of nuclear safety. They also allow to carry out the research on:

- Nuclear and neutron physics.
- Measurement of transmutation reaction rates for minor actinides and fission products in different neutron spectra. Specially designed SCAs (as well as fast reactors) can have appropriate σ_f /σ_c ratios for most actinides and overall neutron economy. Some transmutation reactions using thermal neutrons are:
  - $n + {}^{129}I$ (T = 1.57 × 10^7 years) → ${}^{130}I$ (T = 12. 36 hours) + X
  - $n + {}^{237}Np$ (T= 2.14 × 10^6 years) → ${}^{238}Np$ (T = 2.117 days) + X
  - $n + {}^{241}Am$ (T = 432 years) → ${}^{242}Am$ (T = 16.02 hours) + X
  - $n + {}^{243}Am$ (T = 7.4 × 10^3 years) → ${}^{244}Am$ (T = 10.1 hours) + X.
- Studying neutronics and kinetics of critical and subcritical systems.
- Developing on-line monitoring methods of criticality level.
- Studying neutronics of coupled accelerator-reactor systems.
- Evaluation of nuclear data for radioactive nuclides.

4.11. UTILIZATION OF SUBCRITICAL ASSEMBLIES FOR DEMONSTRATION OF AUTONOMOUS OPERATION

In recent years, SCAs are also utilized in demonstrating machine learning based autonomous operation for nuclear reactors, due to their subcritical design feature and the relevance of applying a graded approach in the application of the safety requirements. Examples of an ongoing research project are given in Refs [38, 39].

5. SUMMARY

Subcritical assemblies have a long history, starting in the first decade of the nuclear era, in supporting the development of early nuclear programmes. More than half of all existing SCAs were commissioned between the 1950s and the mid 1970s, often as test facilities to be used in R&D related to the development of new reactor concepts, performing pioneering experiments in support of their designs, to determine nuclear data and for validation of various reactor physics codes and associated modelling tools. As the goals of many of those early SCAs were reached and their purpose was fulfilled, they were shut down and, in many cases, decommissioned.

A number of these early SCAs continue to operate and new SCAs continued to be commissioned, at a rate of two to four per decade since the 1980s. They purpose has been diverse, with applications in education and training and in nuclear research, including computational code benchmarking, cross-section measurements, detector calibration and development of innovative measurement techniques. These applications take advantage of the design safety features of SCAs, namely subcriticality, low power, and small source term, that categorize these facilities as a low potential hazard.

In recent years, there has been growing interest from Member States in SCAs, with a higher number of new projects being under development than was the case in the past decades. In several cases, the new SCA is planned as the first nuclear installation in the country and is seen as an important step towards developing national capacity in nuclear science and technology, sometimes integrated in a roadmap for establishment of a nuclear power programme. In this respect, the justification for building many of the SCAs recently commissioned or planned is centred on education and training and on development of the national nuclear infrastructure, sometimes followed by building the first research reactor in the country.

This publication was developed in view of these recent developments of the scientific programmes in multiple Members States. It supplements the IAEA Safety Standards by providing guidance and practical information for safety in design and operation of SCAs. It also provides information and examples of utilizing SCAs for various types of research and training experiments. By doing so, it is expected that this publication will contribute to the safe operation and utilization of SCAs.

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ANNEX I. STATISTICS ON SUBCRITICAL ASSEMBLIES IN THE MEMBER STATES

The statistical data compiled in this Annex are a product of the IAEA's Research Reactor Database (RRDB). RRDB is an authoritative database containing technical information on over 800 research reactors, including critical and subcritical assemblies (SCAs) in 67 States. The information in the database, provided by facility focal points nominated through official channels, also includes utilization and administrative information. The data reflect the status of SCAs in Member States as of April 2021. The updates of the national programmes reported in Ref. [I–1] and other relevant sources [I–2] have been also considered in the statistics.

Information on 39 SCAs has been reported by Member States, as shown in Table I–1, Fig. I–1 (by stage of the lifecycle) and Fig. I–2 (by State). Fig. I–3 shows SCAs that are in operation in the Member States. Fig. I–4 shows information on starting and ceasing operation by decades.

TABLE I–1. SCA STATUS

SCAs status	Reported numbers	Country
Planned	3	Indonesia, Mongolia, Tunisia
Under construction	3	Czech Republic, Ukraine, Philippines
Operational	15	Belarus, Belgium, Canada, China, Greece, Iran, Italy, Jordan, Mexico, Netherlands, Russian Federation, United States of America
Temporary shutdown	0	-
Extended shutdown	4	Mexico, Russian Federation
Permanent shutdown	6	Greece, Russian Federation, United States of America
Under decommissioning	0	
Decommissioned	8	Finland, Germany, Iran, Italy, Poland, Romania, United Kingdom

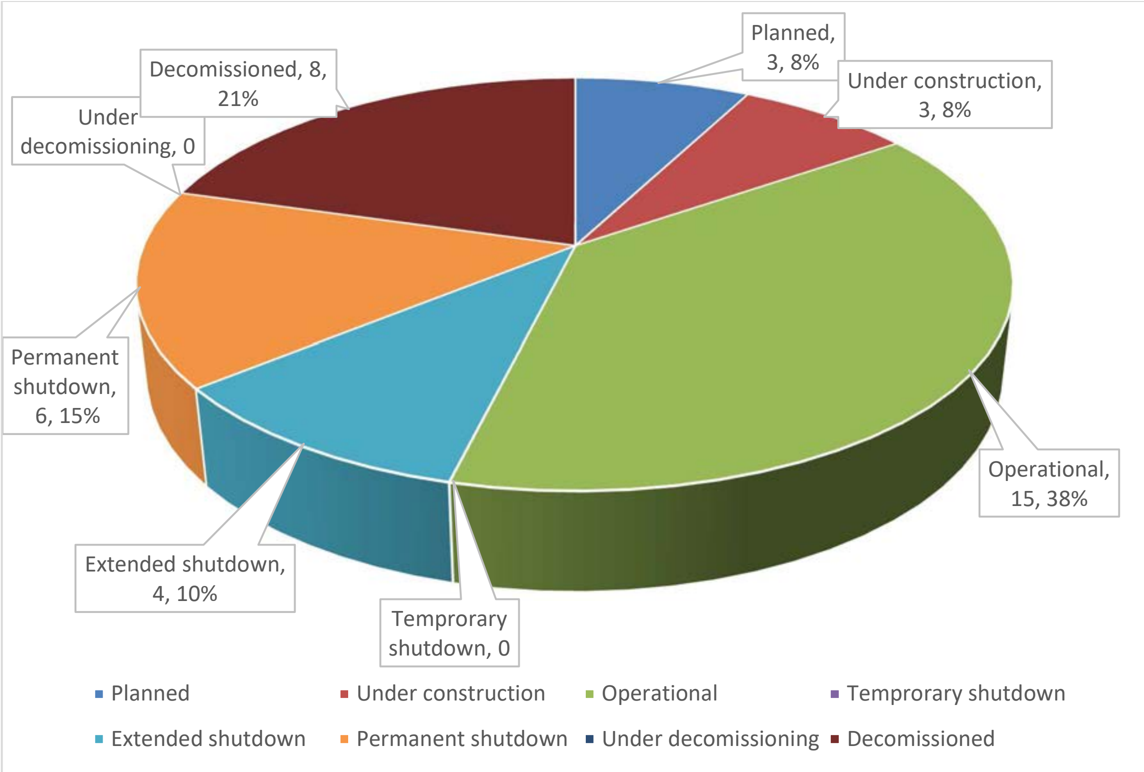

FIG. I–1. SCAs status (by the stage of the lifecycle).

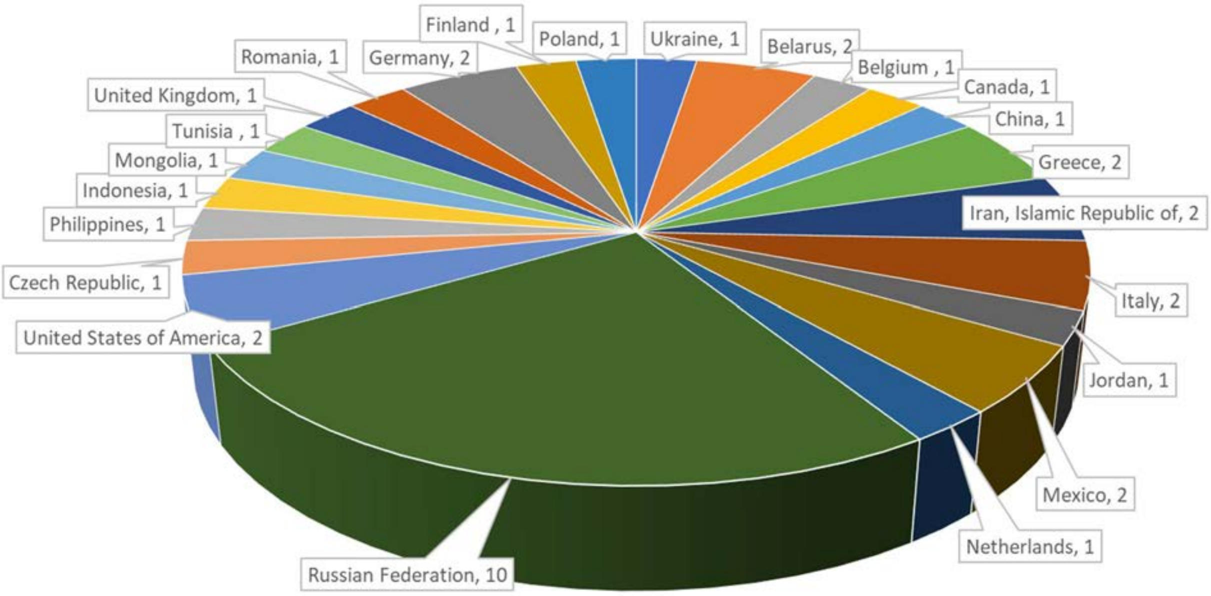

FIG. I–2. SCAs status (by State).

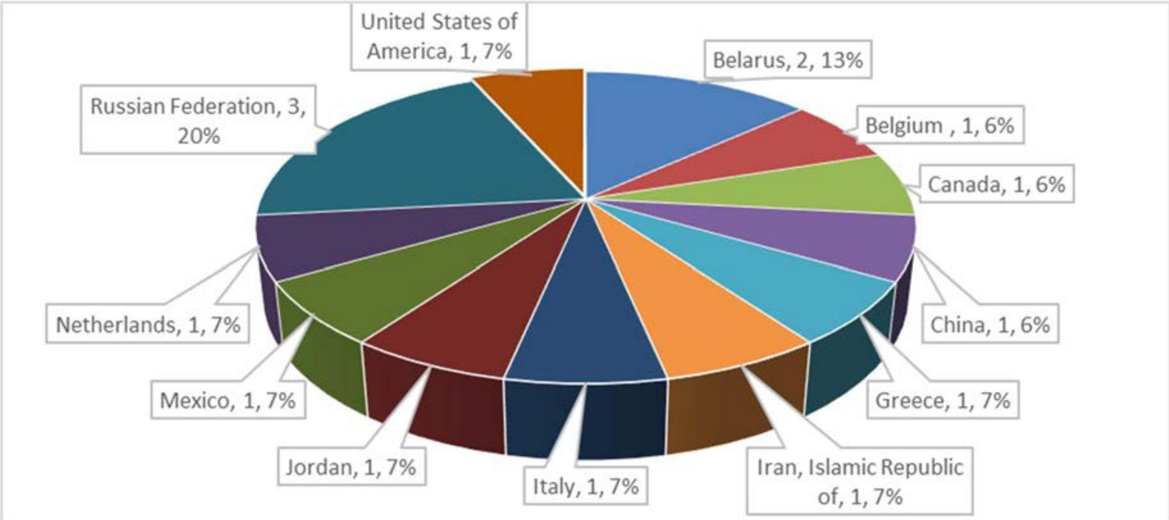

FIG. I–3. SCAs in operation.

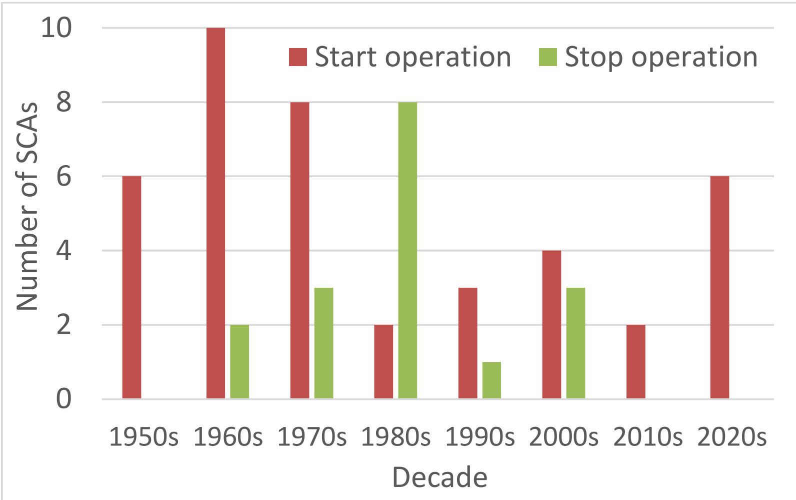

FIG. I–4. SCAs starting and ceasing operation by decades. 2020s includes planned and under construction SCAs.

REFERENCES TO ANNEX I

[I–1] INTERNATIONAL ATOMIC ENERGY AGENCY, Research Reactors: Addressing Challenges and Opportunities to Ensure Effectiveness and Sustainability. Summary of an International Conference (25-29 November 2019, Buenos Aires, Argentina), IAEA Proceedings Series, STI/PUB/1927, IAEA, Vienna (2020).

[I–2] ANGGRAINI, N. D. A., RIYATUN, S., KHAKIM, A., Criticality analysis of Kartini Reactor connected to test facility of subcritical assembly Molybdenum-99 production (SAMOP), Journal of Physics: Conf. Series 1153 (2019) 012105.

ANNEX II. RUSSIAN FEDERATION REGULATIONS FOR SUBCRITICAL BENCHES AND EXPERIENCE FROM THEIR APPLICATION

The Russian Federation has various nuclear research installations (NRIs), including subcritical nuclear benches (SCBs)[1], for which specific nuclear safety rules have been developed. The national case below includes the background to the development of the current Rules, an overview of the main provisions, safety criteria, and principles, as well as experience of their application in the Russian Federation. The materials do not apply to subcritical accelerator-driven system (ADS), which is out of the scope of this publication.

II–1. INTRODUCTION

From the 1950s to the 1970s, during the development of nuclear power in the USSR, the safety of SCBs was provided for by industrial regulations and special safety rules. In 1975, the State Committee for the use of atomic energy in the USSR put in force Rules PBJa-01-75. After the collapse of the USSR in 1991, 22 SCBs were assigned to the supervision of the newly-created regulatory body of the Russian Federation. The development of the regulatory framework was based on the Federal law "On the use of atomic energy" 170-FZ (1995), which defines the facilities covered by this law, including SCBs. This law defines Federal Rules and Regulations (hereinafter - FRR) in the field of nuclear energy use in the Russian Federation that ought to be developed and approved in compliance with the procedure laid down by the Government of the Russian Federation. The developed FRR “General Safety Provisions for NRIs”, NP-033-11 [II–1] establish the purpose and basic principles of safety assurance for nuclear research installations, general requirements for safety assurance at nuclear research installations of various types, as well as specific requirements for research reactors, critical and subcritical benches as potential sources of radiation exposure for the workers (personnel), the public and the environment. The Rules NP-059-2005 [II–2] (hereinafter referred to as the Rules with an indication of specific paras of the regulations) were developed to replace the PBJa-01-75 and specify the requirements of NP-033-11 in terms of the safety of SCBs.

II–2. OVERVIEW OF MAIN PROVISIONS AND SPECIFIC SAFETY CRITERIA AND PRINCIPLES FOR SUBCRITICAL BENCHES IN THE RUSSIAN FEDERATION

Chapter 1 of Rules NP-059-2005 [II–2] defines the general provisions for ensuring nuclear safety for SCBs, the scope of the Rules, and the requirements for nuclear safety in design, construction and operation of SCBs.

Chapter 2 includes requirements for the design of SCBs for nuclear safety. The safety parameters of subcritical assemblies of the SCBs are established, the most significant of which is the effective neutron multiplication factor of subcritical assemblies k_{eff} and its maximal achievable value $k_{eff,max}$ for licensing purposes. In compliance with safety requirements (Ref. [II–2] paras 2.1.2. and 2.1.4.)[2] the value of k_{eff} need to be determined and controlled carefully during SCB operation with high accuracy, and any possibility of an unauthorized increase ought to be excluded. The certified calculation codes ought to be used for safety substantiation

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[1] A subcritical bench (SCB, sometimes referred to as a subcritical stand) is a nuclear installation, which comprises a subcritical assembly and premises, systems and experimental devices located within a specifically designed site> A subcritical assembly is for experimental study of the medium multiplying neutrons, the composition and geometry of which provide under normal operation effective neutron multiplication factor $k_{eff} < 1$

[2] The items in brackets refer to the numbering of the Rules.

of the subcritical assembly. The condition $k_{eff} < 1$ may be realized by choosing the fuel composition and moderator, by provisions for the configuration of the core, and by excluding any possibility to increase the loading of additional fuel or using of a more efficient moderator. Moreover, the value of k_{eff} may be experimentally verified during commissioning of the subcritical assembly. The possibility of unauthorized placement of nuclear materials or of a more effective moderators in the core of a subcritical assembly ought to be excluded, as well as a possibility to increase the concentration of fissile materials and effectiveness of the moderator in the case of using a fuel in solution form (Ref. [II–2] para 2.2.1).

The processes in the core can be controlled by moving of an external neutron source or changing its intensity. The necessity to use reactivity control devices (rods) in the subcritical assembly may be substantiated in the SCB safety analysis report (SAR) (Ref. [II–2] para 2.2.2.).

The Rules determine the main numerical criterion $k_{eff} = 0.98$ in ensuring the nuclear safety of the subcritical assembly of a SCB. The core with $k_{eff} > 0.98$ is treated as a critical assembly. A protection system has to be provided for such SCBs to prevent the occurrence of a nuclear accident[3]. In this case the availability and functions of the emergency protection system and the control safety system and their compliance with the requirements of the FRR have to be provided (Ref. [II–2] para 2.3.).

Chapter 3 provides requirements for nuclear safety in commissioning and operation of SCBs and nuclear materials management in SCB facilities. The nuclear safety requirements cover the physical start-up[4] of SCB, the SCB start-up mode[5], the SCB temporary shutdown mode[6], the SCB long-term shutdown mode[7], and the SCB final shutdown mode[8].

The Rules include requirements on the scope of the main SCB documentation related to nuclear safety, as well as requirements on the form of the SCB technical certificate (Ref. [II–2] paras 3.1.4.; 3.2.11; annexes 1, 2).

Chapter 4 contains requirements to control compliance with the Rules. The operating organization ought to conduct an annual inspection of nuclear safety at SCBs. The results of the inspection ought to be reflected in the annual report of operating organizations on SCB nuclear safety.

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[3] A nuclear accident at an SCB is an accident caused by a disruption of monitoring and control of the intensity of the chain nuclear fission reaction, the formation of a critical mass in the core of the subcritical assembly, or in handling nuclear materials outside of the subcritical assembly.

[4] The physical start-up of an SCB is the commissioning phase, including the loading of nuclear fuel into the core, the achievement of the value of k_{eff} of a subcritical assembly provided in the design of the SCB, and examination of neutron-physical characteristics of the subcritical assembly for experimental confirmation of the safety of the SCB.

[5] The SCB start-up mode is the SCB operation mode, which provides the intensity of the nuclear fission chain reaction necessary for experimental studies by increasing the k_{eff} of the subcritical assembly and (or) using an external neutron source.

[6] The SCB temporary shutdown mode is the SCB operation mode, which consists of carrying out maintenance work on the SCB and the preparation of experimental studies.

[7] The SCB long-term shutdown mode is the SCB operation mode, which consists of carrying out work on the conservation of individual SCB systems and maintaining them in working condition during the time when no experimental studies are planned on the SCB.

[8] The SCB final shutdown mode is the SCB operation mode, consisting of carrying out work on the preparation of the SCB for decommissioning, including unloading nuclear fuel from the subcritical assembly core and removal of nuclear fuel and other nuclear materials from the SCB.

II–3. EXPERIENCE ON THE APPLICATION OF THE RULES ON NUCLEAR SAFETY FOR SUBCRITICAL BENCHES IN THE RUSSIAN FEDERATION

The description and technical characteristics of the SCBs are given in the survey [II–3]. In the Russian Federation, there were no violations of the operational limits and conditions during the SCBs operation.

Currently, under the state supervision there are seven SCBs with subcritical assemblies having $k_{eff,max}$< 0.98, and one SCB with a subcritical assembly having $k_{eff,max}$ > 0.98, for which the experience of applying the Rules is shown below.

The following "University" uranium-water subcritical assemblies are classified as subcritical assemblies with $k_{eff,max}$< 0.98: UV-1 ($k_{eff,max}$ = 0.80), MEPhI; UV-2 ($k_{eff,max}$ = 0.81), MEPhI; uranium-water assembly with the variable lattice spacing UVPSh ($k_{eff,max}$ = 0.83), MEPhI; UV ($k_{eff,max}$ = 0.83), MEI. These subcritical assemblies use fuel elements incorporating uranium metal fuel with a natural concentration of isotopes (0.7 % ^{235}U).

In the SAR of the SCBs, it has to be demonstrated that SCAs composed from uranium metal of natural enrichment and light water have a multiplication factor less than 1 in an infinite lattice and cannot become critical in principle. In the uranium-water subcritical assemblies UV-1, UV-2, UVPSh, UV the necessary value of k_{eff} is provided by the core composition and configuration (Ref. [II–3], Ref. [II–2] para 2.2.1.1.).

Uranium-water subcritical assemblies with a $k_{eff,max}$< 0.98 also include the test assembly of the SCB VVER, MEPhI, that is a replica of the VVER fuel assembly with the specified spacing of fuel elements, at different water-to-fuel ratios. The fuel of experimental fuel rods is a sintered uranium dioxide with 6.5 % enriched ^{235}U. Two types of absorber rod in the assembly are used: non-dismountable and dismountable. Depending on the lattice spacing, the multiplication factor k_{eff} value may vary from 0.66 to 0.88. Reaching the critical state of the assembly is not possible due to a lack of channels for introducing fuel rods more than substantiated by the design and approved in the principal programme of experiments (Ref. [II–2] para 3.3.1.1.-3.3.1.3.). The volume of the assembly core does not exceed the portion of the critical volume (Ref. [II–2] para 2.2.1.1.). An external neutron source for the assembly is inserted to the horizontal experimental channel of the research reactor IRT MEPhI. There is a control panel for remote control of the gate position to regulate the neutron flux of the beam (Ref. [II–2] paras 2.2.2.1(4); 2.2.2.7.). After the irradiation of the assembly, the induced activity of fuel rods and facility components decreases to a negligible value after closing the gate of the beam channel in the time justified in the SAR of the SCB. Different fuels, which can be used at the SCB VVER, have appropriate marking (Ref. [II–2] para. 2.2.1.4.). Fuel handling is carried out in accordance with the "Instructions for ensuring nuclear safety during storage, transportation and reloading of nuclear fuel at the SCB VVER" and meets the requirements of the Rules (Ref. [II–2] para 3.4.) and other FRR for handling nuclear materials.

The uranium-water subcritical assemblies with a $k_{eff,max}$< 0.98 include the high pressure SCB 7VD that is located in the experimental workshop of the Joint Stock Company “Gidropress” within the State Atomic Energy Corporation “Rosatom”. The SCB 7VD is intended for operational lifetime testing of the VVER-1000 dummy fuel assemblies (FAs) and drive mechanisms of the control rods and protection system (CPS). Only fresh (not irradiated) FAs with a fuel enrichment of no more than 2% ^{235}U are used for tests at the SCB 7VD. The design of the SCB 7VD limits the number of FAs that are tested and allows testing of only 7 FAs in the bundle with a $k_{eff} \leq 0.90$ that is substantiated in the SAR of the SCB (Ref. [II–2] paras 2.2.1.1.; 3.1.6.). The SCB 7VD is not intended to research neutron breeding and an external neutron source is not used. There are panel board to control technological parameters and fittings at the subcritical assembly (Ref. [II–2] paras 2.2.2.1(4); 2.2.2.16.).

Subcritical assemblies with a $k_{eff,max} < 0.98$ include the uranium-graphite assembly UG ($k_{eff}^{max} = 0.81$), MEPhI. This subcritical assembly uses fuel elements that incorporate uranium metal fuel with a natural concentration of isotopes (0.7 % ^{235}U). The volume of the core composed of natural uranium metal and graphite does not exceed the limits that are justified in the SAR of the SCB (Ref. [II–2] paras. 2.2.1.1.; 3.1.6). It is impossible to achieve criticality in the subcritical assembly UG due to the lack of the necessary amount of graphite and fuel (Ref. [II–2] para 3.2.8.).

In compliance with established requirements (Ref. [II–2] paras 3.1.4; 3.1.6), all SCBs have an SAR [II–4], a quality assurance programme [II–5], an emergency plan [II–6], and a set of operational documents. The design documentation and SAR of the SCB include the following (Ref. [II–2] paras 2.1.2. (1), (2), (4); 2.1.3. (2), (3) 3.1.6.; 3.2.11.; 3.3.1.3.(1),(4)):

- The fuel burnup is negligible for the entire operating time ;
- The temperature of components in the core of SCAs (excluding SCA 7BD) is determined by the ambient temperature and there is no need to use a circuit with a coolant for cooling the subcritical core;
- The consequences of any abnormal situation do not extend beyond the SCB premises.

In the design of SCBs UV-1, UV-2, VVER, UVPSh, VG, UV and 7BD, it is technically proved in the SAR that $k_{eff,max} < 0.9$ for any violations of normal operation. If this condition is met, the Rules allow for the absence of neutron flux control channels, a protection system and CPS control devices (Ref. [II–2] paras 2.2.2.15, 2.3.1.2.)).

All SCBs belong to the IV category of potential radiation hazard, for which the establishment of a sanitary protection zone is not required in compliance with the requirements of the main sanitary norms OSPORB-99/2010 [II–7].

All SCBs fully meet the requirements of the Rules. Some requirements of the Rules might not be applied to the specific SCB with appropriate justification in the SAR. For example, there is no need for a control panel for subcritical assemblies UV-1; UV-2; VVER, UVPSh, UG (para 2.2.2.16) and there is no need to attain the physical start-up for SCB 7VD (Ref. [II–2] paras 3.2.4-3.2.9).

SCBs, which include subcritical assembly with $k_{eff,max} > 0.98$, include the SCB FS-2 that operates in the mode of a subcritical neutron breeder and has a dual functionality — a test facility and a training facility for qualified personnel for the nuclear industry — that is reflected in the operational procedures for person involved in the work (Ref. [II–2] para 3.1.3). The construction of the subcritical assembly ensures inaccessibility of the working elements of the core and prevents changes to its configuration during experimental work (Ref. [II–2] paras 2.2.1.1. (1), (2)). The SCB FS-2 is equipped with a CPS that provides control of the technological parameters and the intensity of the chain nuclear fission reaction in the core, including the fast suppression (reduction of its intensity) of the reactivity, if needed (Ref. [II–2] paras 2.3.1.1.; 2.3.1.3; 2.3.2.1; 2.3.2.3; 2.3.2.8.). The emergency protection system includes two safety rods. Power is controlled by the moving of the assembly with an external neutron source (Ref. [II–2] para 2.2.2.1(1)). The process is remotely controlled from the operator board

(Ref. [II–2] paras 2.2.2.1(4); 2.2.2.16.). The control panel receives signals about the state of the subcritical assembly and elements of the control system (Ref. [II–2] paras 2.3.1.5; 2.3.2.8.; 2.3.2.10-2.3.2.12.).

The full set of the safety documents of each SCB has been reviewed in the licensing process in compliance with established procedures [II–8].

In the Russian Federation, there are no plans to build a new SCB. Three SCBs, namely UV-1, UV-2, 7VD, are in the final shutdown mode. The SCB UV is in a long-term shutdown mode and procedures to extend its service life are carried out in accordance with the requirements of FRR NP-024-2000 [II–9]. Further operation is planned for five SCBs with subcritical assemblies VVER, UVPSh, UG, UV, FS-2.

The Rules [II–2] do not cover safety in SCB decommissioning, which are included in the FRR NP-028-16 [II–10].

II–4. CONCLUSIONS

Compliance with the requirements of the Rules NP-059-05 ensures nuclear safety in the design, construction and operation of SCBs for various purposes. The Rules apply a graded approach, depending on the design features substantiated in the SCB SAR. Currently, there are no plans to revise the Rules in the Russian Federation.

REFERENCES TO ANNEX II

[II–1] FEDERAL ENVIRONMENTAL, INDUSTRIAL AND NUCLEAR SUPERVISION SERVICE (ROSTECHNADZOR), The Order of Rostechnadzor from 30.06.2011 **348** «On approval and put in force the FRR «General Safety Provisions for NRIs», NP-033-11.

[II–2] FEDERAL ENVIRONMENTAL, INDUSTRIAL AND NUCLEAR SUPERVISION SERVICE (ROSTECHNADZOR), The Resolution of Rostechnadzor from 05.05.2005 **2** «On approval and put in force the FRR «Nuclear Safety Rules for Subcritical Benchs», NP-059-05.

[II–3] GELIOS A.R.B., Research Reactors of the Commission of the Commonwealth of Independent States (CIS), (2016) 470.

[II–4] FEDERAL ENVIRONMENTAL, INDUSTRIAL AND NUCLEAR SUPERVISION SERVICE (ROSTECHNADZOR), The Order of Rostechnadzor from 5.12.2017 **528** «On approval of the FRR «Requirements to the Contents of NRI Safety Analysis Report», NP-049-17.

[II–5] FEDERAL ENVIRONMENTAL, INDUSTRIAL AND NUCLEAR SUPERVISION SERVICE (ROSTECHNADZOR), The Order of Rostechnadzor from 7.02.2012 **85** «On approval of the FRR «The Requirements to Quality Entrance Programme of Facilities», НП-090-11.

[II–6] FEDERAL ENVIRONMENTAL, INDUSTRIAL AND NUCLEAR SUPERVISION SERVICE (ROSTECHNADZOR), The Order of Rostechnadzor from 14.05.2019 **181** «On approval of the FRR «Contents of Action Plan to Protect Personnel in Case of an Accident at NRI», NP-075-19.

[II–7] CHIEF STATE SANITARY OFFICER OF THE RUSSIAN FEDERATION, Main Sanitary Regulations on the Radiation Safety (OSPORB-99/2010), the Decree of the Chief State Sanitary Officer of the Russian Federation from April 26, 2010, **40**.

[II–8] FEDERAL ENVIRONMENTAL, INDUSTRIAL AND NUCLEAR SUPERVISION SERVICE (ROSTECHNADZOR), The Order of Rostechnadzor from 8.10.2014 **453** «Administrative Procedures for the Public Service of Licensing Activities in the Field of Atomic Energy Use to be Provided by the Federal Environmental, Industrial and Nuclear Supervision Service».

[II–9] FEDERAL NUCLEAR AND RADIATION SAFETY AUTHORITY OF RUSSIA (GOSATOMNADZOR), The Resolution of Gosatomnadzor of Russia from 28.12.2000 **16** «On approval and put in force the FRR NP-024-2000 Requirements to Extend Service Life of Facilities.

[II–10] FEDERAL ENVIRONMENTAL, INDUSTRIAL AND NUCLEAR SUPERVISION SERVICE (ROSTECHNADZOR), The Order of Rostechnadzor from 4.04.2017 **108** «On approval and put in force the FRR «Safety Rules for Decommissioning of NRIs», NP-028-16.

ANNEX III. PASSIVE SAFETY FEATURES OF KIPT NEUTRON SOURCE FACILITY

III–1. INTRODUCTION

The neutron source facility of the Kharkov Institute of Physics and Technology (KIPT) of Ukraine has 100 kW electron beam delivered with 100 MeV electrons [III–1]. Tungsten or natural uranium is used as the target material. The electron interactions with the target material produce high-energy photons, which generate neutrons through photonuclear reactions with the target material for driving the subcritical system. The WWR-M2 fuel design [III–2] with ^{235}U enrichment of 19.7 wt% is used in the subcritical assembly (SCA). Beryllium assemblies and graphite blocks are used as a reflector to improve the neutron economy. The SCA is installed in a water tank, which is a part of the primary cooling system. The radial configuration of the SCA, the reflector and the water tank are shown in Fig. III–1. It is noted that, even though accelerator driven systems (ADS) are beyond the scope of the current publication, the focus of this annex is on the passive safety features of neutron source driven SCAs in general.

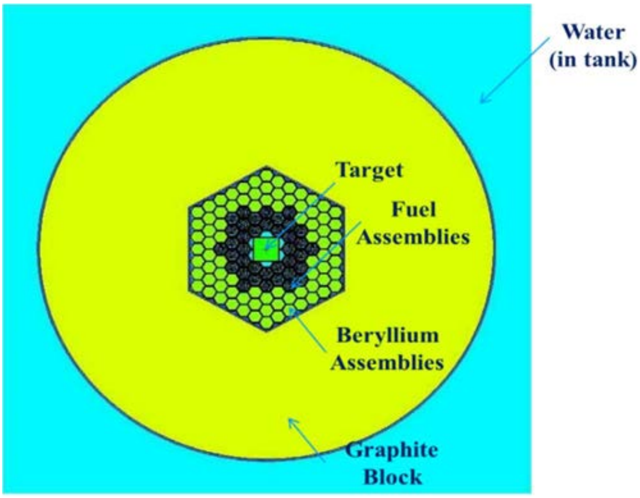


FIG. III–1. Radial configuration of the subcritical assembly, with reflector and water tank.

The SCA is designed to operate with an effective neutron multiplication factor (k_{eff}) value of less than 0.98. To keep this k_{eff} value for a fresh loaded assembly at the beginning of the operation, even when the target is removed, the tungsten target configuration uses 38 fuel assemblies and the uranium target configuration uses 37 fresh fuel assemblies [III–3]. The maximum fission power is ~200 kW with the uranium target and 100 kW electron beam power. The water coolant temperature is ~25 °C during normal operation. The SCA does not use control rods since the design has an adequate subcritical margin.

The temperature reactivity feedback provides a prompt negative reactivity to ensure subcritical operation at all times. The Monte Carlo computer program MCNPX [III–4] was utilized in the analyses with the ENDF/B-VII.0 nuclear data libraries [III–5]. The water coolant temperature changes as well as the corresponding water density changes were used to determine the temperature reactivity feedback. The water coolant temperature varies from ~20 °C to 90 °C. The fuel temperature varies from ~20 °C to ~300 °C.

III–2. REACTIVITY WORTH OF THE FUEL ASSEMBLIES

During normal operation, the k_{eff} value of the SCA is less than 0.98, which provides more than a 2000 pcm reactivity margin below the critical condition. The k_{eff} values of the SCA loaded with different number of fuel assemblies with the tungsten and the uranium targets are shown in Figs III–2 and III–3, respectively.

For the tungsten target case, the fresh loaded assembly at the beginning of operation has 38 fuel assemblies, and the k_{eff} value is only ~0.957. The tungsten material has a large absorption cross-section for thermal neutrons, and the removal of the tungsten target from the SCA results in ~2000 pcm positive reactivity feedback. In contrast, the removal of the uranium target results in a small negative reactivity feedback. To keep the SCA k_{eff} value below 0.98 in all the circumstances, even during the tungsten target removal, the assembly k_{eff} ought to be kept below 0.96 for the tungsten target configuration. The results in Fig. III–3 show that the reactivity worth of a single fuel assembly is also ~500 pcm. To achieve a critical condition, 47 fuel assemblies are needed to utilize the tungsten target.

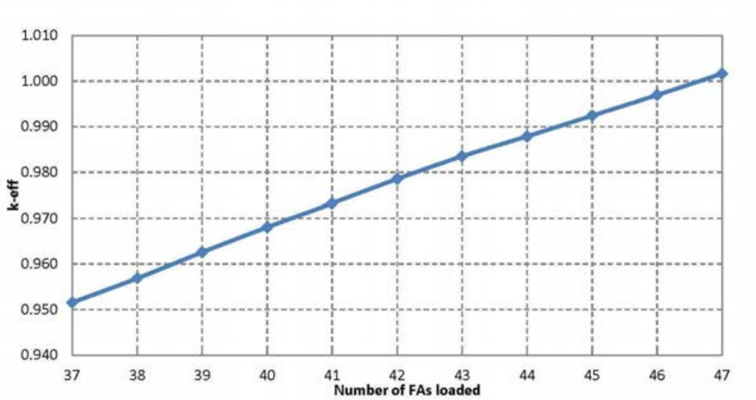


FIG. III–2. k_{eff} value as a function of the number of the fuel assemblies with the tungsten target.

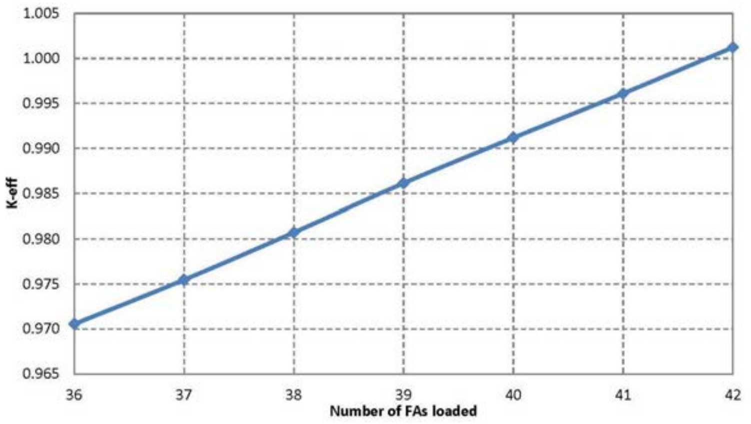


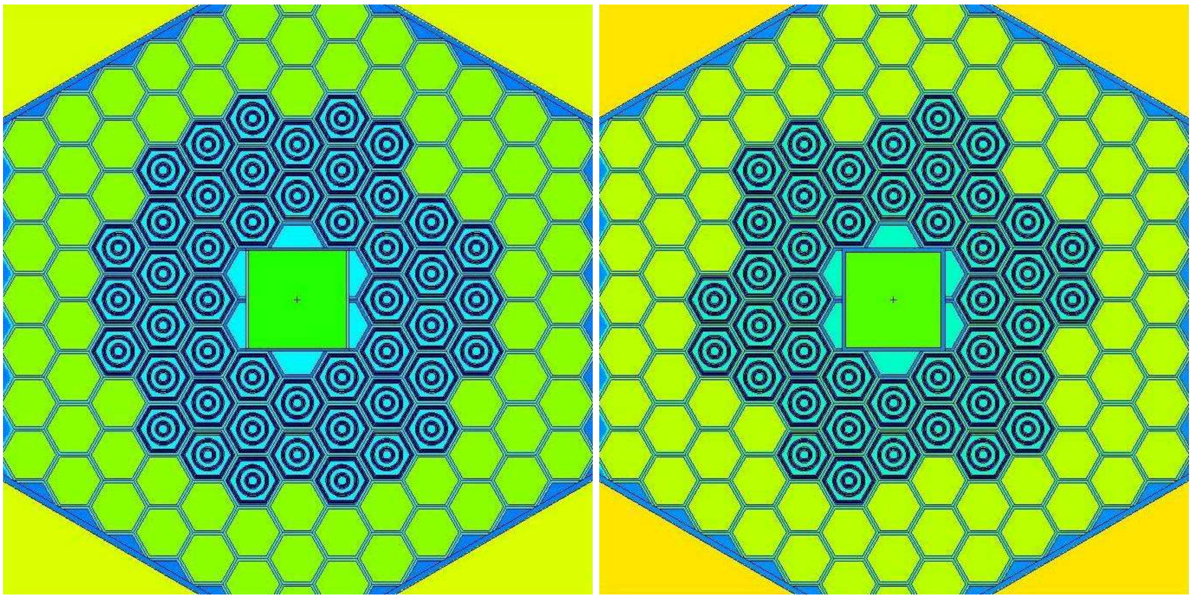
FIG. III–3. k_{eff} value as a function of the number of the fuel assemblies with the uranium target.

For the uranium target case, the fresh loaded SCA can have 37 fuel assemblies to keep the k_{eff} value below 0.98. The number of fuel assemblies loaded is varied to study the reactivity change. The results in Fig. III–2 show that the reactivity worth of a single fuel assembly is ~500 pcm. To achieve a critical condition, the SCA ought to be loaded with 42 fuel assemblies, which is 5 fuel assembles more than the normal loading condition.

III–3. SUBCRITICAL ASSEMBLY REACTIVITY FEEDBACK DUE TO THE CHANGE IN THE TEMPERATURE AND THE DENSITY OF THE WATER MATERIAL

The reference SCA configurations consist of 38 fuel assemblies with the tungsten target and 37 fuel assemblies with the uranium target to keep the k_{eff} value < 0.98 at all times. However, under hypothetical conditions, for example if additional fuel assemblies are loaded, the SCA power will increase and it will be detected. The measured neutron flux values will increase and the measured outlet coolant temperature will increase. This will provide a warning to stop the operation and check the number of loaded fuel assemblies. As long as the k_{eff} value is < 1.0, the power level will reach certain fixed value depending on the accelerator beam power and the k_{eff} value. However, analyses were also performed for unrealistic conditions, which assume that a large number of fuel assemblies were loaded and the k_{eff} value is greater than 1.0. The goal is to assess the system feedback and the consequences of such situation.

Based on the results from the previous section, 47 fuel assemblies are loaded (9 fuel assemblies more than the reference case) with the tungsten target and 42 fuel assemblies are loaded (5 fuel assemblies more than the reference case) with the uranium target to produce a k_{eff} value > 1.0. The two configurations used in this analysis are shown in Fig. III–4.



The tungsten target with 47 fuel assemblies laded at room temperature, k_{eff} = 1.00165 (± 0.00012)

The uranium target with 42 fuel assemblies loaded at room temperature, k_{eff} = 1.00122 (± 0.00012)

FIG. III–4. SCA configurations with k_{eff} value greater than 1.0.

The k_{eff} values of these two configurations were calculated with different water temperature values and the water density was adjusted corresponding to the temperature value. The water density as a function of the temperature is shown in Table III–1. The water density decreases slightly with temperature, before reaching the boiling point. The ENDF/B-VII.0 nuclear data library (in MCNPX code package) has only cross-sections for some isotopes at certain temperature values. Therefore, the cross-section at room temperature (20 °C) is selected and the temperature tmp card is used to adjust the cross-sections.

TABLE III–1. WATER DENSITY AT DIFFERENT TEMPERATURE

Temperature	Density (g/cm^3)
0 - ~20 °C	~1.0
40 °C	0.9922
50 °C	0.9881
60 °C	0.9832
70 °C	0.9778
80 °C	0.9718
90 °C	0.9653

In the KIPT neutron source facility, the fuel region and target assembly have separate coolant loops. Therefore, the temperature of coolant in fuel region and target assembly can be treated separately. Tables III–2 and III–3 give the calculated k_{eff} values as a function of the water temperature. In Table III–2, the water temperatures in the target assembly and the fuel region were simultaneously changed, while Table III–3 gives the results from changing only the water temperature in the fuel region. The results in Tables III–2 and III–3 show that the target water temperature changes have an insignificant impact on the calculated k_{eff} values.

TABLE III–2. KEFF VALUES AS A FUNCTION OF THE WATER TEMPERATURE OF THE TARGET ASSEMBLY AND THE FUEL REGION FOR THE REFERENCE SUBCRITICAL ASSEMBLIES WITH EXTRA FUEL ASSEMBLIES

Water temperature °C	Assembly with the tungsten target and 47 fuel assemblies		Assembly with the uranium target and 42 fuel assemblies	
	k_{eff}	Δk_{eff}	k_{eff}	Δk_{eff}
20.46	1.00165 (± 0.00012)	-	1.00122 (± 0.00012)	-
40	0.99928 (± 0.00012)	-0.00237	0.99874 (± 0.00011)	-0.00248
50	0.99853 (± 0.00012)	-0.00308	0.99771 (± 0.00012)	-0.00351
60	0.99745 (± 0.00011)	-0.00420	0.99668 (± 0.00012)	-0.00454
70	0.99596 (± 0.00012)	-0.00569	0.99549 (± 0.00013)	-0.00573
80	0.99465 (± 0.00012)	-0.00700	0.99398 (± 0.00012)	-0.00724
90	0.99323 (± 0.00012)	-0.00842	0.99260 (± 0.00012)	-0.00862

TABLE III–3. KEFF VALUES AS A FUNCTION OF THE WATER TEMPERATURE OF THE FUEL REGION FOR THE REFERENCE SUBCRITICAL ASSEMBLIES WITH EXTRA FUEL ASSEMBLIES

Water temperature, °C	Assembly with the tungsten target and 47 fuel assemblies		Assembly with the uranium target and 42 fuel assemblies	
	k_{eff}	Δk_{eff}	k_{eff}	Δk_{eff}
20.46	1.00165 (± 0.00012)	-	1.00122 (± 0.00012)	-
40	0.99949 (± 0.00011)	-0.00216	0.99885 (± 0.00011)	-0.00237
50	0.99866 (± 0.00012)	-0.00299	0.99816 (± 0.00012)	-0.00306
60	0.99757 (± 0.00011)	-0.00408	0.99669 (± 0.00012)	-0.00453
70	0.99632 (± 0.00012)	-0.00533	0.99553 (± 0.00012)	-0.00569
80	0.99489 (± 0.00012)	-0.00676	0.99446 (± 0.00012)	-0.00676
90	0.99352 (± 0.00012)	-0.00813	0.99299 (± 0.00011)	-0.00823
90*	0.99349 (± 0.00012)	-0.00816	0.99330 (± 0.00011)	-0.00792

*Only the density of water coolant is changed, no tmp cards used to adjust the cross-section data

The last row of Table III-3 has the k_{eff} results for changing the water density without the use of the tmp card to adjust the temperature of the cross-section data. Comparing the results of the last two rows shows that the change in the k_{eff} is mainly due to the change of the water density.

These results show that increasing the water temperature from room temperature to 40 °C results in a negative reactivity feedback, which changes the k_{eff} value from greater than 1.0 to less than 1.0. Such feedback provides a significant safety feature. In addition, if the water temperature reaches 80 °C, well below the water boiling point, the negative reactivity feedback is ~676 pcm, which is more than the reactivity worth of a single fuel assembly.

The previous analyses were also performed for the reference configurations where the water temperature change and the corresponding density change were considered. The results are given in Tables III–4 and III–5. In the reference configurations, the increase of the water temperature results also in a negative reactivity feedback. As expected, the reactivity change values are smaller than the previous configurations with additional fuel assemblies.

TABLE III–4. K_{EFF} VALUES AS A FUNCTION OF THE WATER TEMPERATURE OF THE TARGET ASSEMBLY AND THE FUEL REGION FOR THE REFERENCE SUBCRITICAL ASSEMBLIES

Water temperature, °C	Reference configuration with the tungsten target		Reference configuration with the uranium target	
	k_{eff}	Δk_{eff}	k_{eff}	Δk_{eff}
20.46	0.95686 (± 0.00013)	-	0.97547 (± 0.00011)	-
40	0.95507 (± 0.00011)	-0.00181	0.97366 (± 0.00011)	-0.00181
50	0.95416 (± 0.00012)	-0.00270	0.97269 (± 0.00012)	-0.00278
60	0.95272 (± 0.00012)	-0.00414	0.97156 (± 0.00011)	-0.00391
70	0.95180 (± 0.00012)	-0.00506	0.97047 (± 0.00012)	-0.00500
80	0.95038 (± 0.00012)	-0.00648	0.96902 (± 0.00013)	-0.00645
90	0.94908 (± 0.00012)	-0.00778	0.96760 (± 0.00012)	-0.00787

TABLE III–5. K_{EFF} VALUES AS A FUNCTION OF THE WATER TEMPERATURE OF THE FUEL REGION FOR THE REFERENCE SUBCRITICAL ASSEMBLIES

Water temperature, °C	Reference configuration with the tungsten target		Reference configuration with the uranium target	
	k_{eff}	Δk_{eff}	k_{eff}	Δk_{eff}
20.46	0.95686 (± 0.00013)	-	0.97547 (± 0.00011)	-
40	0.95512 (± 0.00012)	-0.00174	0.97371 (± 0.00012)	-0.00176
50	0.95422 (± 0.00012)	-0.00264	0.97259 (± 0.00012)	-0.00288
60	0.95321 (± 0.00012)	-0.00365	0.97199 (± 0.00012)	-0.00348
70	0.95196 (± 0.00012)	-0.00490	0.97078 (± 0.00011)	-0.00469
80	0.95076 (± 0.00012)	-0.00610	0.96929 (± 0.00012)	-0.00618
90	0.94928 (± 0.00012)	-0.00758	0.96812 (± 0.00013)	-0.00735
90*	0.94958 (± 0.00012)	-0.00728	0.96802 (± 0.00012)	-0.00745

*Only the density of water coolant is changed, no tmp cards used to adjust the cross-section data

III–4. SUBCRITICAL ASSEMBLY REACTIVITY FEEDBACK DUE TO THE CHANGE IN THE FUEL MATERIAL TEMPERATURE

The results and analyses of the previous section show that the decrease of water density due to the temperature increase is sufficient to keep the system in a subcritical state even if additional fuel assembles are loaded. However, the fuel material temperature will increase first and it will reach higher values. To examine the reactivity impact of this temperature increase, MCNPX analyses were performed with different fuel temperatures without changing the temperature of the water and clad materials. In fact, the fuel temperature increases instantaneously with the power increase before the temperature increase of the cladding and the water. The tungsten target assembly with 47 fuel assemblies and the uranium target assembly with 42 fuel assemblies, which have the k_{eff} above 1.0 at room temperature, were selected for the analyses. The calculated k_{eff} values are given in Table III–6.

TABLE III–6. KEFF VALUES AS A FUNCTION OF THE FUEL MATERIAL TEMPERATURE FOR THE REFERENCE SUBCRITICAL ASSEMBLIES WITH EXTRA FUEL ASSEMBLIES

Fuel Temperature, K	Assembly with the tungsten target and 47 fuel assemblies		Assembly with the uranium target and 42 fuel assemblies	
	k_{eff}	Δk_{eff}	k_{eff}	Δk_{eff}
293.6	1.00165 (± 0.00012)	-	1.00122 (± 0.00012)	-
600	0.99592 (± 0.00011)	-0.00573	0.99546 (± 0.00012)	-0.00576
900	0.99159 (± 0.00011)	-0.01006	0.99156 (± 0.00012)	-0.00966
1200	0.98858 (± 0.00012)	-0.01307	0.98814 (± 0.00011)	-0.01308
2500	0.97815 (± 0.00012)	-0.02350	0.97844 (± 0.00012)	-0.02278

The WWR-M2 fuel design has UO_2 smeared in an aluminium matrix, and its melting temperature is ~933K. The k_{eff} result shows that increasing the fuel temperature to 600 K, well below the aluminium melting point, results in a negative reactivity feedback of ~570 pcm. This negative reactivity decreases k_{eff} to less than 1.0 to achieve a subcritical status.

The results of this section and the previous one show that a temperature increase of the SCA materials produces negative reactivity feedback. This negative reactivity feedback is large enough to keep a subcritical state even if additional fuel assemblies are loaded by mistake resulting in a value of k_{eff} greater than 1.0.

Similar analyses were performed for the reference configurations, the tungsten target with 38 fuel assemblies and the uranium target with 37 fuel assemblies. Again, the increase of the fuel material temperature results in a significant negative reactivity feedback. Other analyses were performed to evaluate the impact of cladding temperature and the results show the cladding temperature change has an insignificant negative effect on the k_{eff} value of the SCA.

III–5. REACTIVITY CHANGE DURING FUEL LOADING PROCESS

The SCA of the KIPT neutron source facility has a loading capacity of 120 hexagonal assemblies surrounding the target assembly. The fully loaded fresh subcritical assembly with the tungsten target assembly has 38 fuel assemblies and 82 beryllium assemblies and with the uranium target assembly it has 37 fuel assemblies and 83 beryllium assemblies. At the beginning of the loading process, the beryllium reflector assemblies will be loaded first at the pre-determined locations, while the fuel locations are loaded with dummy assemblies. In the fuel loading process, fuel assemblies replace the dummy assemblies one by one, until all the fuel assemblies are loaded.

Criticality analyses examined possible loading mistakes of the beryllium assemblies during the loading process. The results showed that replacing beryllium reflector assemblies with dummy assemblies or a corresponding water volume decreases the value of k_{eff}. As expected, the reactivity worth of a beryllium assembly is larger than that of a dummy assembly or the corresponding water volume. Therefore, missing beryllium reflector assemblies will not cause criticality concern, although it will decrease the neutron flux because of lower neutron multiplications. More details for the different mistaking scenarios are given in Ref. [III–6].

III–6. CONCLUSIONS

Argonne National Laboratory (ANL) and the National Science Center-Kharkov Institute of Physics and Technology (NSC-KIPT) have been collaborating on developing a neutron source facility based on the use of an electron accelerator driven subcritical system. The construction of the neutron source facility is finished and the start-up process is currently underway. The safety performance of the KIPT neutrons source facility was analysed for abnormal operating conditions.

First, the effective neutron multiplication was analysed as a function of the number of loaded fuel assemblies. The reference configurations have 38 fuel assemblies with the tungsten target and 37 fuel assemblies with the uranium target. The analyses show that for both the tungsten target and the uranium target configurations, the reactivity of a single fuel assembly is ~500 pcm. It is also shown that 47 and 42 fuel assemblies are necessary for the tungsten target and uranium target configurations to reach criticality, respectively. Therefore 9 and 5 extra fuel assemblies need to be loaded in the tungsten and uranium reference configurations, respectively to reach a critical condition.

The temperature reactivity feedback was analysed as function of the temperature of the different materials for different configurations. The reactivity feedback from the temperature change of each material was analysed separately. A minor decrease of water coolant density, due to small temperature increase results in a significant negative reactivity feedback. If the water temperature increases from room temperature to 90 °C, before reaching the boiling point, the reactivity drop would be ~800 pcm, which is more than enough to offset the reactivity gain by loading an extra fuel assembly. The temperature of the fuel material increase also results in a significant negative reactivity feedback. When the fuel material temperature increases from room temperature to 600K, which is below the melting point of the aluminium alloy, the reactivity drop is ~500 pcm. Such negative feedback enhances the subcritical margin.

The analyses of the transient conditions during the fuel and beryllium loading steps do not present any criticality concern because of negative reactivity feedback. Also, incorrectly loading dummy assemblies instead of beryllium assemblies results in a negative reactivity feedback. The analyses of this study evaluated the passive safety features of the neutron source facility during the fuel loading process of the SCA.

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ANNEX IV. SAFETY CONSIDERATIONS FOR THE PRR-1 TRIGA-FUELED SUBCRITICAL ASSEMBLY

The PRR-1 subcritical assembly for training, education, and research (SATER) is being established by the Philippine Nuclear Research Institute (PNRI) to support capacity building in nuclear science and technology. Slightly irradiated TRIGA fuel will be deployed in PRR-1 SATER, which warrants distinct safety considerations on the fuel configuration and radiological hazard relative to typical SCAs. This is due to the relatively high reactivity worth of each TRIGA fuel and the radiation emitted by fission products. The safety considerations for SATER that are associated with the TRIGA fuel configuration, radiation shielding, and the external neutron source are presented in this annex.

IV–1. INTRODUCTION

PNRI is currently constructing the PRR-1 SATER. It will use slightly irradiated TRIGA fuel rods, which have been stored and maintained in the PRR-1 facility for 32 years. The establishment of SATER is an alternative to the straight-to-decommissioning option that was implemented from 2005 to 2014. While the latter option will render PNRI with no useable fuel for a nuclear facility, SATER will use the existing PRR-1 resources while providing local access to an operating nuclear facility with minimal waste generation [IV–1]. The inherent safety and flexibility of a subcritical assembly (SCA) makes it a suitable nuclear facility as the Philippines restarts its nuclear science and technology programme. PRR-1 SATER will serve as a training facility for research reactor operators, staff from the regulatory body and users, and is projected to engage and increase the nuclear stakeholder base in the country.

PRR-1 SATER is a tank-type subcritical assembly that will operate at “zero” power. The reactor core will consist of 44 TRIGA fuel rods in a square lattice at 4 cm pitch. This configuration results in a maximal k_{eff} of 0.95001 ± 0.00009. Although there are 115 slightly irradiated fuel rods that are available in the PRR-1 facility, the number of fuel rods for SATER was limited to 44 to ensure a large subcritical margin. This is to limit the amount of fuel in the core considering the relatively high reactivity worth of each TRIGA fuel rod. Likewise, since the TRIGA fuel rods that will be deployed in SATER are slightly irradiated, the radiation emitted by fission products needs to be considered in assessing the radiological hazard in the facility in addition to the hazard from the external neutron source.

IV–2. INITIAL CORE CONFIGURATION

The PRR-1 TRIGA fuel rod consists of UZrH fuel elements with ~20 wt % uranium, 19.7% ^{235}U-enriched. Due to its higher fuel enrichment compared to most fuel used in SCAs, the TRIGA fuel is also expected to have higher reactivity worth per fuel. To identify the optimum number of fuel rods that can be loaded in the SATER core while maintaining a large subcritical margin, different configurations of the PRR-1 fuel rods in square lattices were simulated using MCNP5v.1.6 [IV–2] and ENDF/B-VII.0 nuclear data library [IV–3]. A square lattice was preferred because of its simple geometry and to facilitate easier fuel handling. Details of the simulation are presented in Ref. [IV–4].

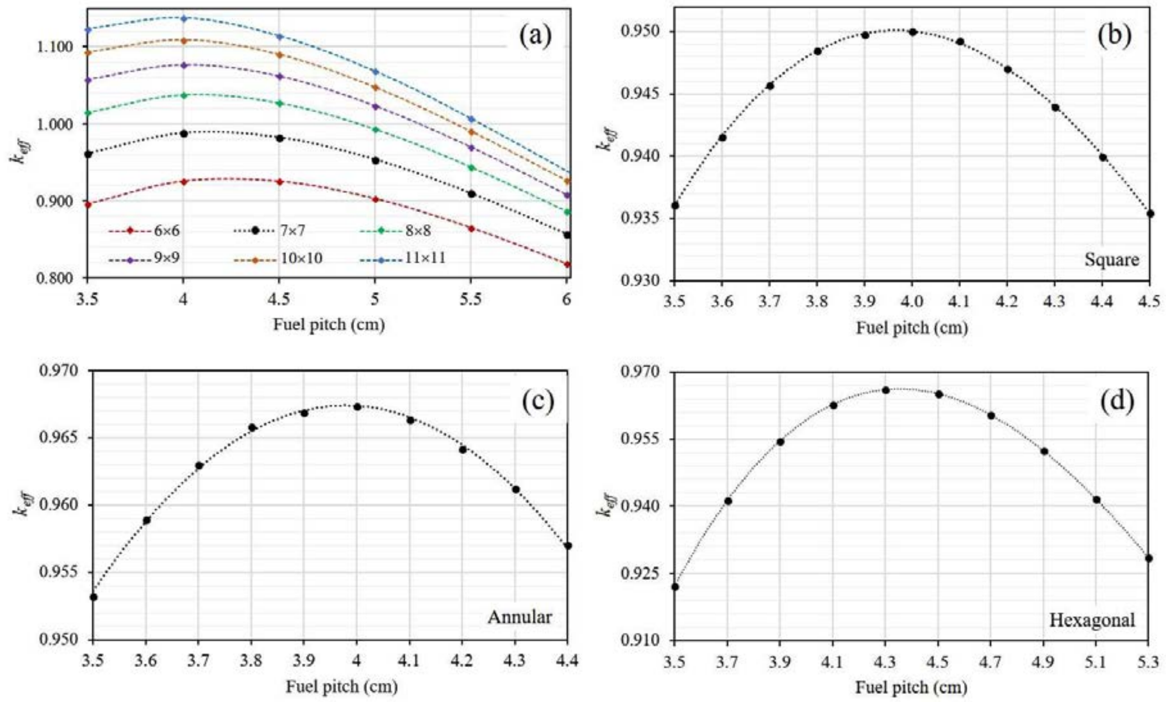

FIG. IV–1. Plots of k_{eff} vs fuel pitch for: (a) six different fuel loadings in square configuration; (b) the selected core configuration for SATER with 44 fuel rods arranged in a 7×7 square lattice with 5 empty slots; (c) alternative annular configuration for 44 fuel rods; and (d) alternative hexagonal configuration for 44 fuel rods. Uncertainties are below 10 pcm.

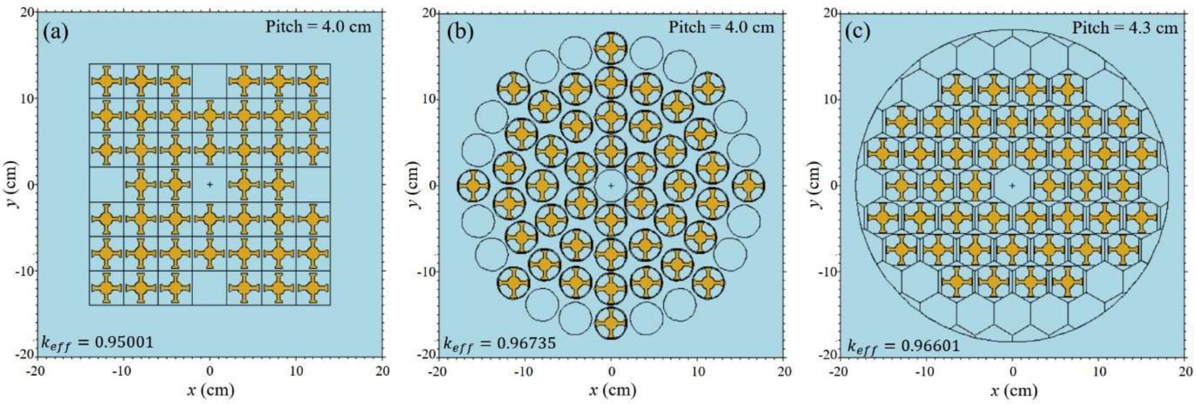

FIG. IV–2. Configurations resulting in maximal k_{eff} for 44 fuel rods in (a) the selected SATER core configuration at 4.0 cm pitch; (b) annular configuration at 4.0 cm pitch; and (c) hexagonal configuration at 4.3 cm pitch.

The plots for the simulated k_{eff} vs pitch for different fuel loadings ranging from 6×6 to 11×11 in square configurations are presented in Fig. IV–1 (a). It is evident in the figure that the best candidate for an SCA is the 7×7 fuel configuration because its plot indicates that k_{eff} will remain lower than 1 for any fuel pitch while using the highest number of fuel rods among the configurations investigated. Empty slots in various locations were investigated until the configuration presented in Fig. IV–2 (a) was obtained, which results in the k_{eff} vs pitch plot in Fig. IV–1 (b). This configuration was selected for PRR-1 SATER due to its large subcritical margin, symmetry, availability of empty slots that can be used for irradiation, and a simple fuel arrangement, that will allow a simple core support structure. Nevertheless, alternative configurations of the 44 fuel rods were investigated to ensure that any rearrangement of the fuel will result in a subcritical condition. The annular and hexagonal configurations presented in Figs IV–2 (b) and (c) were also simulated with different fuel distances. The corresponding k_{eff} vs pitch plots are shown in Figs IV–1 (b) and (c). Similar to the square configuration, the annular configuration attains its maximal k_{eff} at 4.0 cm, whereas for the hexagonal configuration k_{eff} is maximum at 4.3 cm. The calculations performed for alternative configurations of the 44 PRR-1 TRIGA fuel rods are detailed in Ref. [IV–5]. The results summarized in Table IV-1 demonstrate that 44 TRIGA fuel rods arranged in different configurations will remain subcritical with a margin that is greater than 3% $\Delta k/k$.

TABLE IV–1. MAXIMAL k_{eff} FOR DIFFERENT CONFIGURATIONS OF 44 PRR-1 TRIGA FUEL

Fuel configuration	Maximal k_{eff}	Fuel pitch (cm)	Subcritical margin
Square	0.95001 ± 0.00009	4.0	5.26 %$\Delta k/k$
Annular	0.96735 ± 0.00003	4.0	3.38 %$\Delta k/k$
Hexagonal	0.96601 ± 0.00003	4.3	3.52 %$\Delta k/k$

The possibility of loading more than 44 fuel rods was considered as a postulated initiating event in the PRR-1 SATER safety analysis, i.e. inadvertent addition of fuel rods in the SATER core. Calculations indicate that from 44 fuel rods, at least 12 additional fuel rods have to be loaded in the core for the system to be critical. The inadvertent addition of fuel rods in the core is prevented by administrative and physical barriers. Slightly irradiated fuel rods are either secured in the storage tank, which needs special fuel handling tools and trained fuel handlers to move. While fresh fuel rods are locked in a concrete vault. Moreover, empty slots in the SATER core will either be used to hold other core elements (neutron source, irradiation guides) or fitted with a mechanical plug when not in use.

IV–3. RADIATION SHIELDING

The SATER core will be submerged in a tank filled with 11 m^3 of light water, which will serve as a shield against radiation from the slightly irradiated fuel rods and the neutron source. To determine if the shielding of SATER is sufficient, an MCNP model which incorporates the SATER tank has been prepared to calculate the gamma and neutron dose rate distribution from the SATER core. An F4 mesh tally was utilized to map the neutron and gamma doses. The flux tally was modified to dose by means of the flux-to-dose conversion factors (DE and DF card) in MCNP. The ICRP-74 [IV–6] conversion factors were used for neutrons, while for gamma radiation, the ANSI/ANS-6.1.1 [IV–7] factors were used.

The neutron dose and gamma radiation dose from secondary photon interactions in the SATER core were determined by performing criticality (kcode) calculations with MCNP. Results obtained from the calculations were scaled based on the neutron source strength $Q = 1.2 \times 10^7\ \mathrm{s}^{-1}$. The gamma radiation dose from fission products in the irradiated TRIGA fuel rods was taken into account by performing fixed source calculations where multiple volumetric photon sources coinciding with the TRIGA fuel geometry were declared. The photons transported in these calculations have an energy of 661.657 keV corresponding to ^{137}Cs. This is the main fission product expected to contribute to the dose from the fuel rods after 32 years in storage. Results obtained from the fixed source calculations were scaled by a factor that was derived from dose rate measurements obtained in contact with the irradiated fuel rods. In addition to the dose distribution under normal conditions, the potential exposure arising from a credible accident was also assessed by determining the dose distribution for a scenario when there is complete water loss in the SATER tank. The horizontal (x-y plane) distribution of dose from photons emitted by fission products for normal operation and for the water loss accident are presented in Fig. IV–3 in which the symmetrical distribution of the dose is evident.

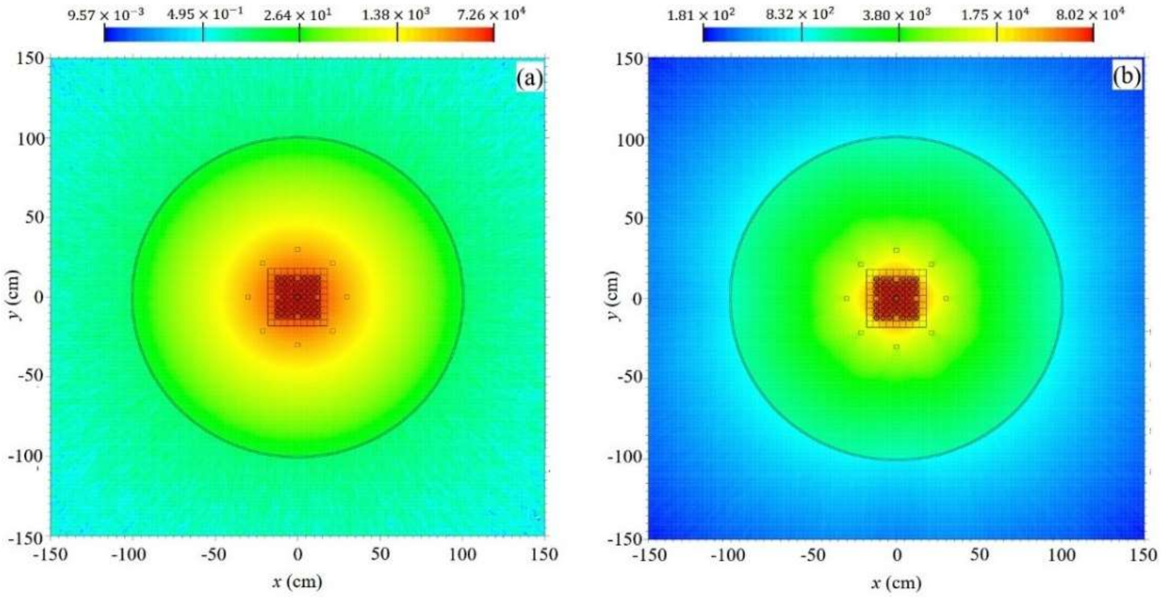

FIG. IV–3. Photon dose distribution in μSv/h from fission products in the x-y (horizontal plane) at $z = 34\ cm$ *under (a) normal conditions and (b) water loss accident.*

The relative distribution of doses from neutrons, secondary photons from nuclear reactions in the core, and photons from fission products are presented in Fig. IV–4 for the normal conditions and for the water loss accident. The dominant dose contributor for both cases analysed, are the photons from the fission products. Nonetheless, results show that the estimated radiation levels at the surface and the top of the SATER tank are about 3 μSv/h and ~0.3 μSv/h, respectively, under normal conditions, which demonstrates that a water filled SATER tank provides sufficient radiation shielding. For the accident scenario, the estimated dose rates at the surface and the top of the tank are 400 μSv/h and 250 μSv/h, respectively. These results indicate that the radiation doses will only be significant within the facility boundary and is unlikely to result in inadvertent radiation exposure of the public and the environment even in the case of complete water loss in the tank.

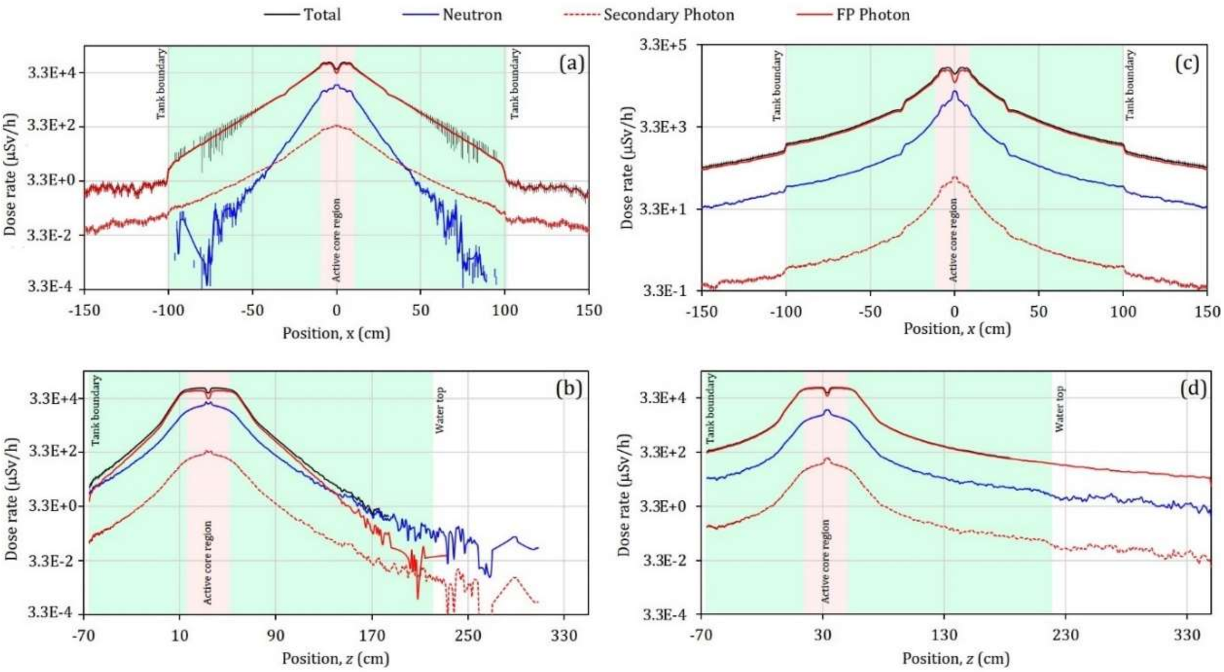

FIG. IV–4. Dose distribution for normal condition along the (a) x-axis and (b) z-axis. Dose distribution for accident condition along the (c) x-axis and (d) z-axis. Dose distributions along the x-axis are taken at $y = 0$ cm and $z = 34$ cm while the z axis distributions are taken at $x = 0$ cm and $y = 34$.

IV–4. EXTERNAL NEUTRON SOURCE

A legacy Pu-Be neutron source (NS) was initially considered to be used as the external NS for SATER because it is available in the facility and it has a very long half-life. But because of the limited documentation for the source and the difficulty of confirming the integrity of its encapsulation, the operators opted to reuse the legacy source for another application and procure a new Am-Be NS with a stainless-steel double encapsulation for SATER. An 80 cm high cylindrical container (∅ 70 cm) made of high density polyethylene (HDPE) was designed, which will also serve as radiation shielding when handling the NS. HDPE was selected as the shielding material because it has good moderating properties, it is lightweight, and it can be used in rugged conditions. The radiological hazard associated with the source was assessed by calculating the dose distribution for a 185 GBq (5 Ci) Am-Be source, which is expected to have an emission rate of $Q = 1.2 \times 10^7\ \mathrm{s}^{-1}$. The method described in the previous section was also used to calculate the dose distribution from the NS.

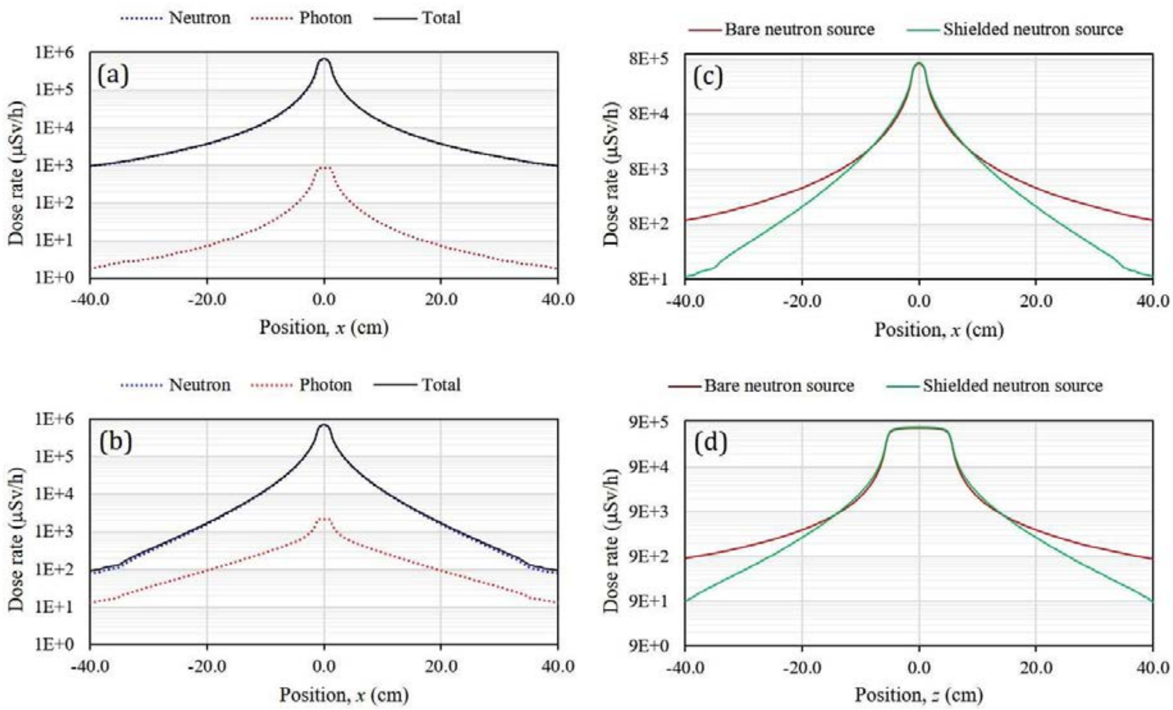

FIG. IV–5. Dose rate distribution for (a) unshielded and (b) shielded 185 GBq (5 Ci) Am-Be source. Comparison of the dose rate distribution along the (c) x-axis and (d) z-axis from the unshielded and shielded source.

Figs IV–5 (a) and (b) show the dose rate contributions from neutrons and photons that are emitted by a 185 GBq (5 Ci) Am-Be source in unshielded and shielded conditions. The neutron dose rate is 3 orders of magnitude higher than the photon dose for the unshielded NS. A comparison of the total dose rate shows that with the shielding, the dose rate at the surface of the container was reduced by one order of magnitude as shown in Figs IV-5 (c) and (d). The estimated dose rates at a distance of 40 cm from the NS are 800-900 μSv/h for the unshielded condition and 80-90 μSv/h for the shielded condition. These indicate that the radiological hazard from the NS is significantly decreased with a shielding container with an appropriate choice of materials and dimensions.

The effect of using different neutron sources (Pu-Be, ^{252}Cf, and Am-Be) on the SATER k_{eff} was also investigated. This was done by changing the neutron spectra of the declared neutron source in the SATER MCNP model. Simulations with 10^8 particle histories resulted in a value of k_{eff} that varies within one standard deviation. However, by including a volumetric source in the calculations with the corresponding material composition for the neutron sources, the Pu-Be source increased the k_{eff} of the system by 0.3% $\Delta k/k$, because ^{239}Pu is also a fissile material. This increase in k_{eff} was not observed for calculations with Am-Be and ^{252}Cf.

IV–5. CONCLUSION

Safety considerations related to the fuel configuration, radiation shielding, and external neutron source for PRR-1 SATER are presented and assessed. Results indicate that the chosen SATER fuel loading will remain subcritical in different configurations. It has also been demonstrated that the SATER has sufficient shielding against radiation under normal conditions, and that the potential exposures in the event of a loss of water accident are limited to within the facility boundary. A neutron source container was also designed, which significantly reduced the radiological hazard from the source. The results presented will be subject to independent regulatory review and will be verified during the commissioning tests.

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ANNEX V. EXAMPLE OF THE APPLICATION OF GRADED APPROACH TO THE SM1 SUBCRITICAL ASSEMBLY

V–1. INTRODUCTION AND SM1 DESCRIPTION

SM1 is a thermal subcritical assembly (SCA) located at the University of Pavia in Pavia, Italy with 206 natural uranium fuel elements (~2000 kg), moderated with light water, and a Pu-Be neutron source (Source Intensity = 7×10^6 neutrons/s) inserted in the lattice. The k_{eff}, value is 0.86 (1 mW Power). SM1 is a type one SCA and the application of a graded approach begins with categorization of the facility in accordance with its potential hazard. A first qualitative categorization identifies SM1 as a facility with a low radiological hazard potential and no radiological hazard potential beyond the research reactor hall in which the SCA is located. The characteristics considered for this evaluation are listed in para. 2.7 of IAEA Safety Standards Series No. SSG-22, Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors [V–1].

The SCA is surrounded by a radiochemistry unit that has its own licence and the only access is through this unit. It is placed in a room with natural ventilation and no direct access from outside. The fuel is at room temperature and not sealed. Air is checked by passing through a filter which is analysed following procedures approved by the regulatory body. In case of an emergency (e.g. fire) the external emergency plan of the adjacent TRIGA research reactor is activated.

System changes are managed in accordance with current legislation with reference to the method for managing the facility issued by the regulatory body. For special maintenance activities and specific maintenance contracts, external suppliers are selected, evaluated and qualified according to documented procedures. Regarding the maintenance of systems not related to safety issues, the service is provided by the University of Pavia.

All workers involved with the external plant systems are informed of radiation hazards and equipped with all the necessary personal equipment for radiation protection (e.g. gloves, overshoes).

Each person who accesses the controlled zone receives a dosimeter for individual monitoring. Specific training is organized for workers depending on the complexity and duration of the activity.

All activities are recorded and stored at the facility in accordance with document management procedures in order to ensure traceability, assessment of the status of the system, and long-term planning related to the plant ageing.

Many of the safety requirements established in IAEA Safety Standards Series No. SSR-3, Safety of Research Reactors [V–2] can be applied in accordance with a graded approach. In particular, SM1 has low power, a small source term, no enrichment, no spent fuel elements, no high pressure systems and no heating systems. Additionally, it is difficult to change the overall configuration to affect the SCA safety.

The main activities of SM1 are:

– Preparation of short-lived and medium-lived radioisotopes for experiments in the field of radiochemistry;

– Measurement of the neutron flux through neutron capture reactions;
– Irradiation of small electronic components with low neutron flux;
– Radiation hardness tests on different silicon photomultiplier devices.

The factors to be considered in the application of a graded approach for SM1 are listed in Table V–1. The main safety features for SM1 are described in Table V–2. The application of the safety requirements established in SSR-3 [V–2] to SM1 is presented in Table V–3.

TABLE V–1. FACTORS TO BE CONSIDERED IN THE USE OF A GRADED APPROACH FOR SM1

Factors	Description
Reactor power	0
Source term	Total activity of fission products in core: 5.75×10^7 Bq
Amount and enrichment of fissile and fissionable material at the facility	Amount: 2 t of U_{nat} Enrichment = 0
Amount of spent fuel	0
Existence of high pressure systems	None
Type and mass of moderator, reflector, coolant	Light water Mass: 1.5 t of water and paraffin 300 kg
Amount and rate of reactivity that can be introduced	Remains subcritical in normal operation and in the design basis accident
Quality of containment /confinement	The building provides adequate confinement related to the potential off-site hazard
Ageing of the facility	No irradiation damage is detectable, SSCs are few and simple Chemical effects, mechanical damage and electrochemical corrosion are monitored
Utilization (experimental devices, tests, experiments)	Experimental activities involve low radiological hazard. Dose rates at the edge of the tank (lid closed) are: 1.5-2.0 μSv/h (neutron), 3.0-4.0 μSv/h (gamma) and 4.5-6.0 μSv/h in total. Outside the containment (chest level): 0.4 μSv/h (neutron), 0.4 μSv/h (gamma) and 0.8 μSv/h in total. Dose rates in the hall are at background levels.
Siting – site evaluation; external hazards, and proximity to population	The assembly is located within a public university department, nevertheless the radiological consequences outside the facility are really low in consideration of the effect that the contributions of external hazards can have.

TABLE V–2. MAIN SAFETY FEATURES FOR SM1

Safety Features	Description
Reactivity control	None
Radiation protection - shielding	Shielding for the neutron source when not inserted in the SCA. Subcritical assembly has polyethylene shielding.
Radiation protection - confinement (e.g. ventilation system)	No ventilation system present. The hall is closed without windows and no direct access by off-site.
Radiation protection - monitoring	Very simple standard portable units for dose rate and contamination checks. Alarms on water level, fire and security system.
Cooling system	Not for safety. Alarm on water level for radiation protection.
Neutron source	Manual handling procedure for sealed neutron source (with simple and safe mean)
Storage room (fuel and neutron source)	Same hall with restricted access
Protection system	Very simple with just a few SSCs (digital for economical reason)
Emergency monitoring system	Very simple: portable units for dose rate and contamination checks in the SCA room, air sampling and gamma spectrometer
Architecture of systems and layout	Very simple layout (one room) Just a few systems and not intertwined
External emergency support	Mainly fire station, security and medical assistance

TABLE V–3. APPLICATION OF SSR-3 SAFETY REQUIREMENTS TO SM1

Requirement No.	Application
1: Safety analysis report	A safety analysis report has to be prepared by the operating organization, providing a justification of the site and the design and the basis for the safe operation of the SCA. The safety analysis report has been reviewed and assessed by the regulatory body.
2: Responsibility for management of safety	The operating organization has the prime responsibility for the safety of the SCA over its lifetime, including utilization, modification, decommissioning and its final release from regulatory control.
3: Safety policy	The operating organization has established and implement safety policies that give safety the highest priority.
4: Integrated management system	The operating organization has not established a proper integrated management system but procedures are present that cover all the operational aspects.
5: Safety assessment	The adequacy of the design of the SCA has been verified by means of comprehensive deterministic safety analysis and has be validated by independent verification by individuals or groups independent from those who originally performed the design work. The safety assessment has been continued with periodic safety reviews.

TABLE V–3. APPLICATION OF SSR-3 SAFETY REQUIREMENTS TO SM1 (cont.)

6: Safety committee	A safety committee (or an advisory group) that is independent from the reactor manager has been established to advise the operating organization on all the safety aspects of the SCA.
7: Main safety functions	The design of the SCA ensures for all states of the facility the removal of heat from the SCA and confinement of the radioactive material as well as shielding against radiation. Control of reactivity and accidental radioactive releases are not applicable by design.
8: Radiation protection	The design of the SCA ensures that radiation doses to workers and other personnel at the SCA facility and to members of the public do not exceed the established dose limits, and that they are kept optimized for operational states. They remain below acceptable limits and as low as reasonably achievable in, and following, accident conditions.
9: Design	The design of the SCA ensures that the facility and items important to safety have the appropriate characteristics to ensure that the safety functions can be performed with the necessary reliability, that the SCA can be operated safely within the operational limits and conditions for its entire lifetime and can be safely decommissioned, and that impacts on the environment are minimized.
10: Defence in depth	The SM1 design prevents deviations in normal operation, and prevents accidents and mitigate their radiological consequences if they occur. The SM1 has inherent safety features. Radioactive releases are kept as low as practicable by hall confinement.
11 Interfaces with nuclear security and Safeguards	The State system of accounting for nuclear material is designed and implemented in an integrated manner so that it does not compromise safety.
12: Use of the graded approach	The use of the graded approach of the safety requirements is commensurate with the potential hazard of the SM1 facility.
13: Proven engineering practices	Items important to safety are designed in accordance with the relevant national and international codes and standards. An example is the implementation of the code of conduct request by the regulatory body.
14: Provisions for construction	Items important to safety have been designed so that they can be manufactured, constructed, assembled, installed and erected in accordance with established processes that ensure the achievement of the design specifications and the necessary level of safety.
15: Features for radioactive waste management and decommissioning	The SM1 design is simple and facilitates decommissioning, for which plans are developed throughout the lifetime of the SCA after its installation and design. Any modification as well as present and future activities take into account radioactive waste management and decommissioning.
16: Safety classification of structures, systems and components	All items important to safety are identified and classified on the basis of their safety function and their safety significance.
17: Design basis for items important to safety	Items important to safety have specific capability, reliability and functionality for the relevant operational states, for accident conditions and for conditions arising from internal and external hazards. Acceptance criteria are defined.
18: Postulated initiating events	A systematic approach to identifying a comprehensive set of postulated initiating events such that all foreseeable events with the potential for serious consequences and all foreseeable events with a significant frequency of occurrence are considered has been implemented.
19. Internal and external hazards	All foreseeable internal hazards and external hazards for the SCA, including the potential for human induced events directly or indirectly to affect the safety are identified and evaluated.

TABLE V–3. APPLICATION OF SSR-3 SAFETY REQUIREMENTS TO SM1 (cont.)

20: Design basis accidents	A set of incident conditions have been considered and postulated by external initiating events.
21: Design limits	A set of design limits have been specified for all operational states and for accident conditions.
22: Design extension conditions	A set of design extension conditions have been derived for the purpose of enhancing safety. A graded approach has been applied to set design extension conditions derived on the basis of engineering judgement and using deterministic assessments. The design extension conditions have been used to identify additional accident scenarios.
23. Engineered safety features	Engineered safety features are provided to prevent anticipated operational occurrences and accident conditions and mitigate their consequences, if these occur.
24. Reliability of items important to safety	The reliability of items important to safety is commensurate with their safety significance. (e.g. redundancy of radioprotection instrumentation).
25. Single failure criterion	Safety systems are simple, but the single failure criterion is applied where possible.
26. Common cause failures	Safety systems are simple, but common cause failures are considered where possible.
27. Physical separation and independence	Physical separation and independence of safety systems has achieved where possible.
28. Fail-safe design	The concept of fail-safe design is considered for systems and components important to safety where possible.
29. Qualification of items important to safety	Items important to safety match regulatory body requirements.
30. Design for commissioning	The SM1 design is fixed and the requirement is not properly applicable but the design may include provisions to operate with transition cores of different characteristics.
31. Calibration, testing, maintenance…	Items important to safety are calibrated, tested, maintained, repaired or replaced, inspected and monitored as required to ensure their capability of performing their functions and to maintain their integrity in all conditions. In general, calibration and testing are very simple as are the safety systems.
32. Design for emergency preparedness and response	The emergency preparedness and response plan and systems are very simple and commensurate with the potential hazards. There are not active safety systems but simple radiation monitoring and a remote water level and fire alarms.
33. Design for decommissioning	It is applicable if extended to modification and experimental devices.
34. Design for radiation protection	Provisions are in place for ensuring that the exposure of operating personnel, reactor users and the public will be below dose limits and will be kept as low as reasonably achievable, and that the relevant dose constraints will be taken into consideration.
35. Design for optimal operator performance	Systematic consideration of human factors is applied at its experimental facilities in accordance with the simplicity of the design.
36. Provision for safe utilization and modification	Provisions for safe utilization and modifications are in place. Few cases are allowed in order to ensure safety.

TABLE V–3. APPLICATION OF SSR-3 SAFETY REQUIREMENTS TO SM1 (cont.)

37. Ageing management	A proportionate and simple ageing management programme is considered. Relevant mechanisms of ageing are identified and potential mechanisms for ageing related degradation are monitored. The life cycles of the technology utilized and the possible obsolescence of the technology is simple and considered.
38. Provision for long shutdown periods	There are no true differences between operation and long shutdown in term of human resource and system for safety and security.
39. Prevention of unauthorized access	Prevention of unauthorized access are implemented and are commensurate with the type and amount of nuclear material.
40. Prevention of disruptive or adverse interactions	There are a small number of simple components and systems. Where applicable, prevention of disruptive or adverse interactions between systems important to safety are evaluated and adverse interactions prevented.
41. Safety analysis of the design	In accordance with a graded approach in consideration of the potential hazards, a safety analysis of the design for the SCA is conducted with deterministic analysis states to be evaluated and assessed.
42. Buildings and structures	The buildings and structures important to safety are designed to keep radiation levels and radioactive releases on and off the site as low as reasonably achievable and below authorized limits for all operational states, for design basis accidents and, as far as practicable, for design extension conditions.
43. Means of confinement	Commensurate with the low level of radiological consequences, a means of confinement is provided to ensure the confinement of radioactive substances in operational states and in accident conditions, and to protect the reactor against natural external events and human induced events.
44. Reactor core and fuel design	Due to very low energy deposition (Power 1mW) in the SM1 fuel elements, the structural integrity is in good condition.
45. Provision of reactivity control	SM1 cannot be critical by design and no reactivity control system is installed.
46. Reactor shutdown systems	SM1 cannot be critical and no reactivity control system is installed. The shutdown is achieved by source removal.
47. Design of reactor coolant systems	No cooling systems and related systems are present. (“zero” power).
48. Emergency cooling of the reactor core	No Emergency cooling and systems are present. (“zero” power).
49. Provision of instrumentation and control systems	Instrumentation is provided for monitoring the facility and determining the status of the SCA in all states.
50. Reactor protection system	The SCA simplicity and the low potential hazard maintain, by design, the reactor in the safe state condition (no criticality).
51. Reliability and testability of instrumentation and control systems	Instrumentation for items important to safety is designed for high functional reliability and periodic testability commensurate with the safety function(s) performed. (radiation protection monitoring).
52. Use of computer based equipment in systems important to safety	There are no systems important to safety dependent upon computer based equipment.
53. Control room	No control room present. All systems are limited to the reactor hall that is safely accessible in any facility state.

TABLE V–3. APPLICATION OF SSR-3 SAFETY REQUIREMENTS TO SM1 (cont.)

54. Supplementary control room	No supplementary control room is present.
55. Emergency response facilities	The facility includes the necessary emergency response facilities on the site. Their design allows that personnel to be able to perform expected tasks for managing an emergency under conditions generated by accidents as well as initiating events.
56. Electrical power supply systems	The reactor remains subcritical, by design, in any facility state. Systems are simple and these could be powered by batteries.
57. Radiation protection systems	Equipment is provided to ensure that there is adequate radiation monitoring in operational states and accident conditions.
58. Handling and storage systems for fuel and core components	The design includes provisions for the safe handling and storage of fresh and irradiated fuel and core components.
59. Radioactive waste systems	The design of the SCA and its associated experimental facilities includes provisions to enhance safety in waste management and to minimize the generation of radioactive waste. Systems are provided for treating solid and liquid radioactive waste to keep the amounts and concentrations of radioactive releases as low as reasonably achievable and below authorized limits on discharges.
60. Performance of supporting systems and auxiliary systems	The SCA remains subcritical, by design, in any facility state. Equipment is provided to ensure that safety systems remain active in any facility state.
61. Fire protection systems	Fire protection systems, including fire detection systems and fire extinguishing systems, fire containment barriers and smoke control systems, are provided throughout the hall, with due account taken of the results of the fire hazard analysis.
62. Lighting systems	Adequate lighting is provided in the hall for operational states and in accident conditions.
63. Lifting equipment	Equipment is provided for lifting and lowering items important to safety, and for lifting and lowering other items in the proximity of items important to safety.
64. Air conditioning systems and ventilation systems	No Air conditioning systems and ventilation systems are present.
65. Compressed air systems	No Compressed air systems are present.
66. Experimental devices	Experimental devices are designed so that they will not adversely affect the safety of the SCA in any facility states. In particular, experimental devices are designed so that neither the operation nor the failure of an experimental device will result in an unacceptable change in reactivity for the SCA, affect operation of the protection system, compromise confinement or lead to unacceptable radiological consequences.
67. Responsibilities of the operating organization	The operating organization has the overall responsibility for safety in the operation of the facility.
68. Structure and functions	The structure of the operating organization and the functions, roles and responsibilities of its personnel are established and documented.

TABLE V–3. APPLICATION OF SSR-3 SAFETY REQUIREMENTS TO SM1 (cont.)

69. Operating personnel	The operating organization assigns direct responsibility and authority for the safe operation of the SCA to the reactor manager. The reactor manager has overall responsibility for all aspects of operation, training, maintenance, periodic testing, inspection, utilization and modification of the SCA.
70. Training, retraining and qualification	The operating organization ensures that safety related functions are performed by suitably qualified, competent and fit-for-duty personnel.
71. Operational limits and conditions	The operating organization ensures that SM1 is operated in accordance with the operational limits and conditions.
72. Performance of safety related activities	The operating organization ensures that safety related activities are adequately analysed and controlled to ensure that the risks associated with harmful effects of ionizing radiation are kept as low as reasonably achievable.
73. Commissioning programme	The operating organization ensures that a commissioning programme for the SCA is established and implemented.
74. Operating procedures	Operating procedures for SM1are developed that apply for normal operation, anticipated operational occurrences and accident conditions, in accordance with the policy of the operating organization and the requirements of the regulatory body.
75. Main control rooms, supplementary control room and control equipment	The operating organization ensures that equipment is maintained in a suitable condition.
76. Material conditions and housekeeping	The operating organization developed and implemented programmes to maintain a high standard of material conditions, housekeeping and cleanliness in all working areas.
77. Maintenance, periodic testing and inspection	The operating organization ensures that effective programmes for maintenance, periodic testing and inspection are established and implemented.
78. Core management and fuel handling	Core management and fuel handling procedures for the facility are established to ensure compliance with operational limits and conditions and consistency with the utilization programme.
79. Fire safety	The operating organization makes arrangements for ensuring fire safety.
80. Non-radiation-related safety	The operating organization established and implemented a programme to ensure that safety related risks associated with non-radiation-related hazards to personnel involved in activities at the reactor facility are kept as low as reasonably achievable.
81. Emergency preparedness	The operating organization developed emergency arrangements for preparedness for, and response to, a nuclear or radiological emergency.
82. Records and reports	The operating organization established and maintained a system for the control of records and reports.
83. Utilization and modification	The operating organization established and implemented a programme to manage utilization and modifications of the reactor.
84. Radiation protection	The operating organization established and implemented a radiation protection programme.
85. Management of radioactive waste	The operating organization established and implemented a programme for the management of radioactive waste.

TABLE V–3. APPLICATION OF SSR-3 SAFETY REQUIREMENTS TO SM1 (cont.)

86. Ageing management	The operating organization ensures that an effective ageing programme is present to manage the ageing of items important to safety so that the required safety functions of structures, systems and components are fulfilled over the entire operating lifetime of the facility.
87. Extended shutdown	If extended shutdown in applied, the operating organization maintains the same requirements.
88. Feedback of operating experience	The operating organization established a programme to learn from events at the reactor facility and events in other SCAs.
89. Decommissioning plan	The operating organization needs to prepare a decommissioning plan and will maintain it throughout the lifetime of the facility.
90. Interfaces between nuclear safety and nuclear security	The interfaces between safety and security for SM1 have been addressed in an integrated manner throughout the lifetime of the SCA. Safety measures and security measures are established and implemented in such a manner that they do not compromise each other.

V–2. EVENTS ANALYSIS FOR SM1

A relevant list of postulated initiating events is as follows (and this was used for dose assessments):

1. Internal events:
   - Loss of sources;
   - Theft of sources;
   - Accidental and/or inadvertent exposures;
   - Explosion due to uncontrolled chemical or physical reaction;
   - Maintenance of environments or systems;
   - Fires (malicious or involuntary);
   - Accidental spillage of contaminated wastewater.

2. External events:
   - Seismic events;
   - Collapse of buildings (structural collapse or aircraft impact);
   - Flooding/flood;
   - Fire and lightning.

From the estimate of the probability of occurrence, the assessments of potential exposures from the design basis accident and from the only other hypothesis of a collapse of the building are considered.

Design basis accident

Cracking of a single fuel ingot of the 1040 present in the lattice and the release of $4x10^4$ Bq of ^{137}Cs and $6.51x10^4$ Bq of ^{131}I is postulated which would produce an iodine concentration in air of 600 Bq/m^3. This concentration would lead to an effective committed dose of 0.013 mSv, using a dose coefficient for ^{131}I of $1.1x10^{-8}$ Sv/Bq and considering an exposure time of 1 hour. The irradiation dose from submersion from an iodine cloud is negligible, as are the effective doses and the dose equivalent to the organs (skin and extremities) due to the cloud radiation and to the subsequent treatment of the reactor water contaminated by ^{137}Cs.

Collapse of the entire building

Unlike the other areas of the department, the most important radioactive sources kept in the SCA hall would be found with relative ease with adequate research and appropriate instrumentation (e.g. neutron monitors for the localization of the reactor injecting source), among the rubble of the collapsed building.

The recovery operations would have the highest priority and would be conducted by the staff of the facility under the direct supervision of the Technical Director and the Radiation Protection Officer.

Even assuming minor difficulties and inconveniences during the recovery phase, it is estimated that the effective doses to personnel would not exceed a few tens of μSv. There is no public exposure.

Similar doses are conceivable if emergency responders (who are considered part of the general population) performing excavations to save persons trapped under the rubble unknowingly encounter radiation sources kept in the SM1 room, which are partially unshielded. To address this possibility, recovery operations are coordinated and a radioactivity expert familiar with the radiological situation at the site (e.g. the Radiation Protection Officer of the Department) is included in the emergency committee.

V–3. OPERATIONAL LIMITS AND CONDITIONS FOR SM1

The following OLCs are based on a safety analysis of SM1 and its design:

- Core discharge has to be notified in advance to the regulatory body by the reactor responsible.
- Fuel grids configuration is fixed.
- The water level needs to be checked and maintained over a pre-established specific level. An alarm starts if water decreases below this level. The level has to be restored, and the cause investigated.
- Radiation protection and monitor equipment need to be operable and alarms are defined and set. The instrumentation is periodically checked and calibrated every three years by authorized institutes. In case of non-operability the activity is interrupted until the instrumentation functionality is restored and tested.
- With the reactor in operation, the responsible operator/supervisor have to be present or available on call.
- Materials to be irradiated are approved in advance.
- In- and out-core neutron source positions are predefined in the reactor room and no other positions are authorized.
- Personnel of the radiation protection service check for contamination and dose rates in defined positions following the scheduled scheme.
- If any deviations occur, these have to be recorded, and the neutron source placed out of the core until actions to remediate are taken successfully.
- Staffing has to be consistent with the organization chart approved by the regulatory body in normal operation and in accident conditions.
- Limited use and quantity of flammable materials.

REFERENCES TO ANNEX V

[V–1] INTERNATIONAL ATOMIC ENERGY AGENCY, Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors, IAEA Safety Standards Series No. SSG-22, IAEA, Vienna (2012). (a revision of this publication is in preparation.)

[V–2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Research Reactors, IAEA Safety Standards Series No. SSR-3, IAEA, Vienna (2016).

ANNEX VI. EDUCATIONAL UTILIZATION OF THE PENNSYLVANIA STATE UNIVERSITY SUBCRITICAL GRAPHITE REACTOR FACILITY

The graphite subcritical reactor facility (GSR) was constructed at Penn State in 1958 as part of a graduate student project. The pile was intended to expand upon the research reactor facility's capabilities to educate students in the burgeoning field of nuclear engineering. Since then, it has been used continuously for 55 years as part of the reactor physics curriculum. Currently the GSR is used as the basis for teaching subcritical physics to 100 undergraduate students each year. Additionally, the facility is used by researchers who need a well-thermalized neutron field for their experiments. Recently, the facility has been used to develop sensitive neutron detectors for nuclear safeguards purposes. The inherent simplicity and flexibility of the GSR ensures that it will be useful for many years to come.

VI–1. FACILITY DESCRIPTION

The GSR is constructed of several hundred blocks of reactor-grade graphite to form an array that measures 266 cm x 161.5 cm x 178 cm. The entire facility is clad in cadmium sheeting covered in aluminium as shown in Figs VI–1 and VI–2. The cadmium cover shields the users from the neutron flux from the pile. The facility can be used as a sigma pile by inserting from one to five 37 GBq (1 Ci) Pu-Be neutron sources at various points. It can also become a subcritical reactor by replacing some of the graphite with natural uranium rods. The fuel was donated by the Atomic Energy Commission in 1958. The fuel can be configured in four analysed loadings for different experiments, see Fig. VI–3. The maximum flux in the pile is approximately 10^4 n/cm^{2}s. The maximum k_{eff} is approximately 0.7. By removing graphite, various measurements can be performed with neutron detectors and activation foils.

VI–2. EDUCATIONAL UTILIZATION

The students perform three experiments on the GSR facility: neutron moderation in graphite, neutron diffusion in graphite and criticality estimation. These experiments are enhanced using modern computer simulation tools. For each experiment, the students measure the neutron flux at various positions using BF_3 detectors or indium foils. They compare these measurements to simulations using MCNP and a simple Monte Carlo simulation written in the programming language of their choice. Typically, the students use MATLAB or C++. These results are compared to literature values corrected for differences in material characteristics. Because of the relative homogeneity of the GSR, the measurements and simulations agree well with published data.

The pile is initially configured as shown in Fig. VI–4, with no fuel and a single Pu-Be neutron source in the centre of the pile. The students measure neutron flux at various distances from the source using either cadmium-covered indium foils or a cadmium-covered BF_3 proportional-mode detector. From the data, the students can calculate the slowing down length of the Pu-Be neutrons in graphite. Following this experiment, the source is removed from the centre of the pile and replaced with four sources at the far end, configured as shown in Fig. VI–5. The students can now measure thermal neutron flux (bare indium foils or BF_3 detectors along the major axis of the pile to determine the extent of neutron diffusion in the pile. The GSR is then loaded with fuel (loading 1.3, Fig. VI–3) and k_{eff} is measured using critical buckling calculations. Each exercise is also accompanied by simulations in MCNP in order to train the students on combining simulation and measurement techniques. This prepares the students for the final laboratories at the TRIGA research reactor.

FIG. VI–1. Penn State Graphite Subcritical Reactor (cadmium cover partially removed).

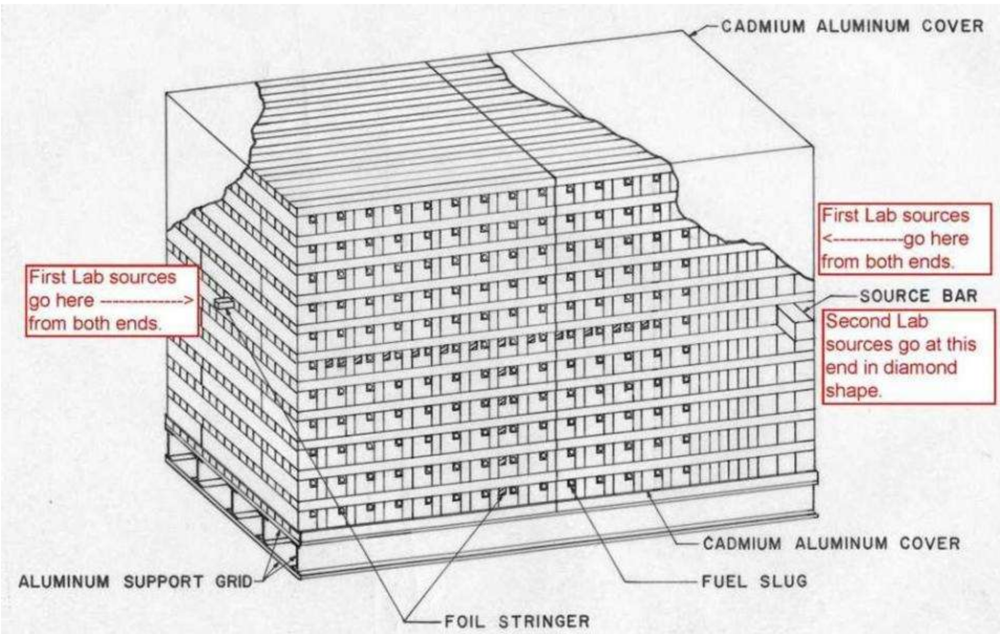

FIG. VI–2. Diagram of Graphite Subcritical Reactor showing structural components and source locations for laboratory experiments.

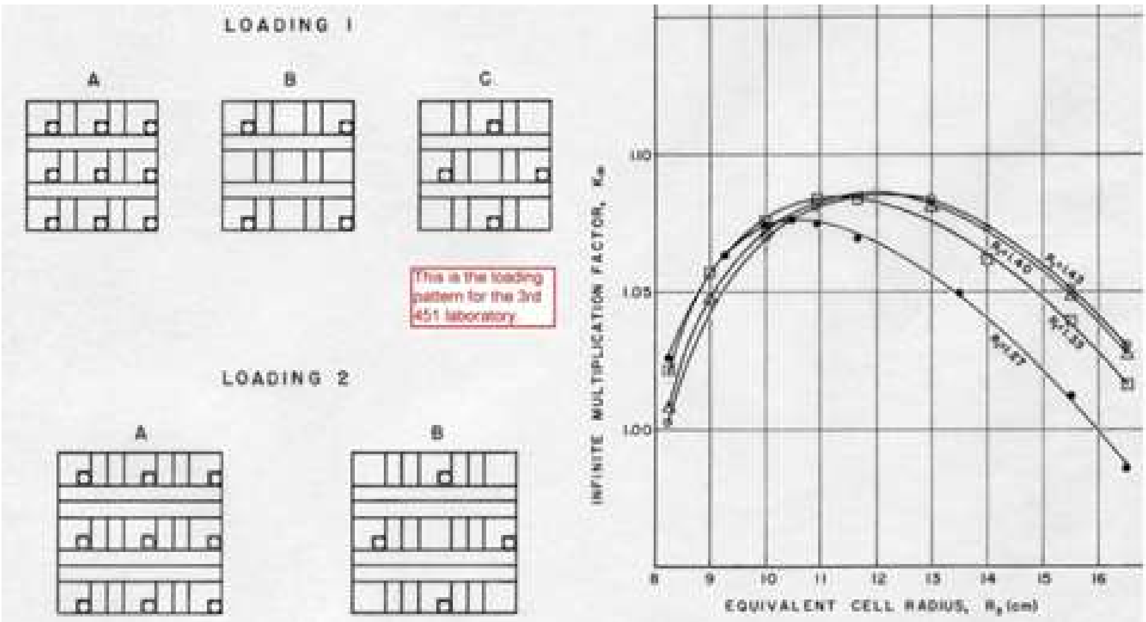

FIG. VI–3. Various possible fuel loading configurations.

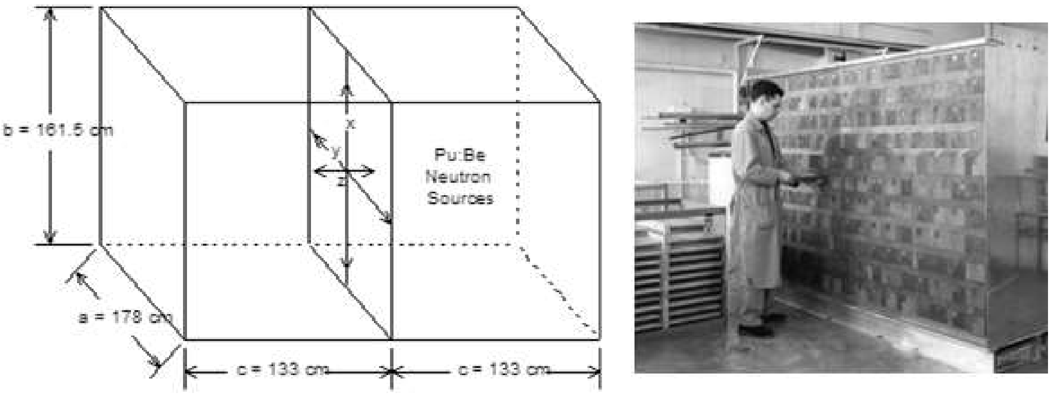

FIG. VI–4. Graphite pile configured for neutron moderation measurements.

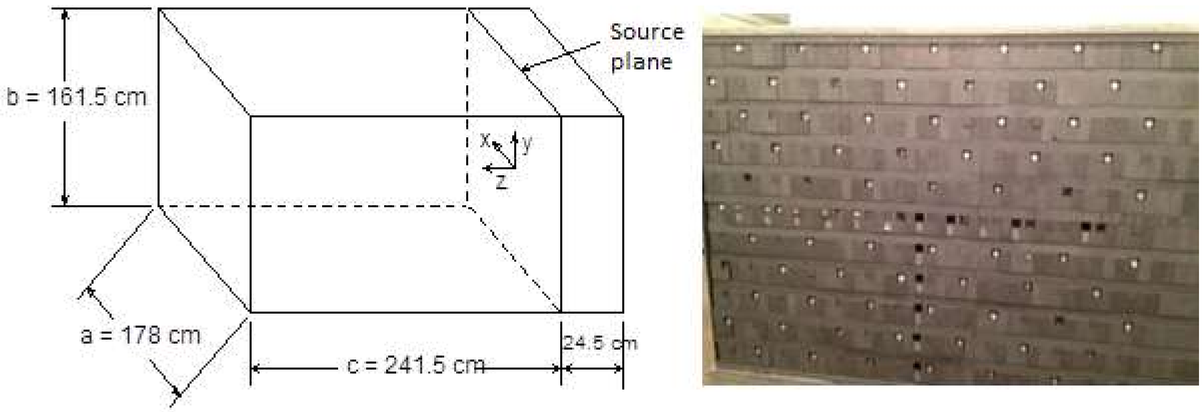

FIG. VI–5. Graphite pile configured for neutron diffusion and critical buckling (fuel loaded) measurements.

VI–3. RESEARCH UTILIZATION

The graphite assembly is used as a neutron source for a diverse field of researchers. The pile can be configured to provide a volumetric source of thermal neutrons. The spectrum can also be tuned by placing varying layers of graphite between the sources and the detectors. Using this technique, the spectrum can change from pure Pu-Be or ^{252}Cf to a fully thermalized spectrum.

Four Pu-Be sources can be loaded into one end of the pile for diffusion measurements over the length of the pile. See Fig. VI–5 for one configuration. The shorter end of the pile (towards the right) presents a thermal neutron field for researchers if the cadmium cover is removed. Measurements taken close to the pile will show variations in the field caused by the physical location of the sources. Since the flux is low at this point, the fixture can be used in this configuration for detector development and testing. Additionally, a diverse array of fields can be created using a combination of neutron sources, natural uranium fuel and neutron and gamma shielding and moderating material. The laboratory is equipped with activation foils and gamma spectroscopy equipment as well as BF_3, boron-lined and ^{3}He neutron detectors.

VI–4. CONCLUSION

The Penn State GSR is a valuable tool for education and research in the 21st century. The simplicity of the system is an advantage, not to be underestimated. Today the majority of work involves computer simulation of complex systems; from reactor design to nuclear safeguards to space research. The GSR represents a simple and easily simulated physical system with which to train students and benchmark simulations.

REFERENCES TO ANNEX VI

[VI–1] REMICK, F. J., A Graphite Moderated Subcritical Reactor – Sigma Pile, Master's Thesis, The Pennsylvania State Univ., PA (1958).

[VI–2] HEIDRICH, B.J., Experiments in reactor physics – Laboratory manual, Course: Nuclear Engineering 451, The Pennsylvania State Univ., PA (2013)

ANNEX VII. THE SUBCRITICAL ASSEMBLY OF POLYTECHNIQUE MONTRÉAL

VII–1. INTRODUCTION

At the beginning of the 1960s, the Government of Quebec initiated a nuclear power programme. In this programme, two nuclear power plants were built by Atomic Energy of Canada Ltd (AECL), Gentilly-1, operational from 1971 to 1979, and Gentilly-2, operational from 1983 to 2012. As part of the Quebecois governmental policy, the Nuclear Engineering Institute (IGN) at Polytechnique Montréal was founded in the summer of 1970. The IGN development followed two main axes: conducting research related to the operation and safety of nuclear installations; and implementing an education programme in nuclear engineering. This programme supported the design, building, operation, and the development of the regulation of Quebec's nuclear power plants. Three nuclear training and research facilities were acquired, two subcritical assemblies (SCAs) and a SLOWPOKE-2 research reactor of AECL design. In January 1974, AECL (the branch located in Chalk River Ontario, and presently consolidated as Canadian Nuclear Laboratories) provided the natural uranium for the two SCAs. The first experimental facility dedicated to training was a graphite SCA of 38 aluminium clad, natural uranium metal rods (1017 kg natural uranium) inserted in a parallelepiped graphite structure. The second training and research facility was a light water SCA containing 1473 aluminium clad, natural uranium dioxide rods (2681 kg natural uranium) in a cylindrical aluminium tank resting on a graphite pedestal shielded with cadmium sheets, and with a 19 GBq (0.5 Ci) Ra-Be source of neutrons in its base. Furthermore, in May 1976, the SLOWPOKE-2 nuclear research reactor started with 0.9 kg of 93% HEU U/Al alloy, replaced in 1997 with 5.6 kg of 20% LEU UO_2 fuel. In 1982, the in-tank SCA was dismantled, and its 2681 kg natural uranium were returned to AECL. Presently, Polytechnique Montréal owns only one SCA, the graphite exponential pile of 38 aluminium clad, natural uranium metal rods. After the shutdown of the Gentilly-2 nuclear power plant in December of 2012, the nuclear engineering programme at Polytechnique Montréal was gradually reduced and will resume in 2020-2021 [VII–1].

VII–2. POLYTECHNIQUE MONTRÉAL SUBCRITICAL ASSEMBLY – DESCRIPTION

The moderator of the SCA consists of 340 graphite blocks, Nuclear Grade II, as defined by the vendor Airco Speer Graphite, and shown in Fig. VII–1. Each block has a squared cross-section of 10.2x10.2 cm and a length of 1.5 m. The blocks are placed horizontally in 20 layers with 17 blocks to the layer. The structure forms a parallelepiped of approximately 1.5 m in length, 1.7 m in width and 2.0 m in height. All blocks in this array have round trough channels of 4 cm in diameter to accommodate fuel rods or graphite plugs. Therefore, the lattice step is relatively large, and the identification of the cells is easy. The structure of graphite is divided in two regions, the sides and the central multiplying lattice. Thus, 340 holes are provided, giving versatility for teaching purposes since they allow the selection of various fuel-to-moderator ratios. The channels not used for fuel rods can be converted to a moderator by the insertion of graphite plugs, resulting in 99% graphite blocks. The graphite lattice together with the graphite plugs inserted is nine tons of graphite. The top and the sides of the graphite pile are covered with cadmium sheets of 0.1 cm thickness. The base is secured in a steel frame in concrete.

FIG. VII–1. Polytechnique Montréal Subcritical Graphite Pile and Central Source Holder.

The SCA fuel consists in 38 natural-uranium rods, 1017 kg of U, intended for square or hexagonal lattices. Each fuel rod is a cylinder of 3.7 cm external diameter, including the 0.2 cm aluminium cladding. These rods are 1.5 m long, originating from the 2.2 m length fuel rods made initially for National Research Experimental Reactor (NRX operated by AECL between 1947 and 1993 at Chalk River, Ontario).

This assembly represents 5–10% of a bare critical reactor. Therefore, it cannot sustain a chain reaction without an external source of neutrons. A set of two Am-Be sources of neutrons, each of 11 GBq (0.3 Ci) and emitting 7.5×10^5 neutrons/sec, enable the researchers to achieve a steady distribution inside the graphite pile. The neutron source holder consists of two holes in a stainless steel cylinder, machined in the middle of the central graphite plug of the ninth row from the base, as shown in Fig. VII-1.

Along with the fuel channels, several slots are provided for neutron detector insertion, as passive stringer bearing foil, or BF_3 detectors. The passive neutron detectors are indium, gold or manganese foils. For example, in indium metal foils of 2-3 cm in diameter and having 0.5 mm in thickness. ^{115}In will activate to ^{116m}In and will emit gamma rays of 1.2 and 2.4 MeV to become ^{116}In. Consequently, standard gamma-ray counting is sufficient for a thermal neutron flux distribution and buckling characterization, assuming that there are no epithermal neutrons and the large indium resonance capture at 1.44 eV can be neglected.

The active neutron detectors are BF_3 of around 2.5 cm diameter and 20 cm active length. BF_3 are proportional counters that almost exclusively respond to thermal neutrons while the probability is very low for a higher energy neutron to be absorbed by ^{10}B. The detectors can be inserted in the regular graphite channels or in the slots provided for vertical or lateral neutron detection.

Considering that the SCA generates around 3×10^{-5} W heat (i.e. negligible) and considering that the eventual increase in temperature is minimal, the installation has no cooling system. If desired, the air channels between the rods and the graphite will reproduce the coolant channels of the reactor. Moreover, the safety measures are commensurate with the deep subcriticality and the low neutron intensity. When the installation is not used, the uranium rods are stored in a shielded and locked metal box, and the neutron sources are enclosed in a concrete store built in the floor and locked. Because of its simplicity, the SCA does not require special safety measures, and it was installed in a regular area of Polytechnique Montréal, being protected by the structure of the building and by additional security features specific to nuclear installations.

VII–3. POLYTECHNIQUE MONTRÉAL SUBCRITICAL ASSEMBLY - EXPERIMENTAL USE

The assembly serves to measure the axial and the radial flux distribution, its buckling, and for optimizing the arrangement, the spacing and the proportions of the fuel and of the moderator. In addition, it helps to demonstrate fundamental concepts in nuclear engineering such as Fermi age, diffusion length, migration area and thermal utilization factors by inter-lattice cell studies. The following paragraphs give few examples of these experiments that are carried out simply by manually inserting the fuel rods, the neutron source and the neutron detectors.

One experiment with this SCA is the measurement of the neutron flux distribution in various directions through the assembly for two particular fuel rod lattices: hexagonal and square. Further, the buckling of the flux is determined, resulting in the computation of the mass of the fuel and moderator, and the volume of the full-sized reactor that could sustain a chain reaction itself, without the neutron source.

The delayed neutrons generated from the thermal fission of ^{235}U can be detected from the activation of the SCA for around ten minutes, followed by a rapid retrieval of the sources, while monitoring the evolution in time of the neutron flux. Their half-lives are obtained by incremental subtraction of the counting rate corresponding to the longer half-life from the counting rate of BF_3 measured at specific time intervals during the experiment.

Another interesting experiment is the reflector study. The flux is measured throughout the assembly in a horizontal line, and the measurement is repeated at various levels in the assembly. Observing, in a comprehensive manner, the flattening of the flux due to the reflector thickness leads to a better understanding of the reactor design optimization for reflector savings.

The reactivity of the system is evaluated with multiplication measurements by monitoring the neutron density with BF_3 proportional counters. For the SCA of Polytechnique Montréal the classical four factors method of Lamarsh [VII–2] anticipated an infinite multiplication factor of approximately 1.06. Experimentally, with the primary Am-Be sources and the BF_3 detectors counting almost 10^3 neutrons/sec, the observed multiplication factor for the fuel available was below 0.45.

VII–4. SUMMARY

The SCA was used by Nuclear Engineering Institute of Polytechnique Montréal for almost 40 years, between 1974 and 2012. It served for the training of more than 200 graduate nuclear engineers who contributed successfully to the global nuclear industry.

REFERENCES TO ANNEX VII

[VII–1] KOCLAS, J. Réacteur Sous-Critique de l'École Polytechnique de Montréal, Rapport de Sûreté, IGE-188-Rev 2.2, (2006).

[VII–2] LAMARSH, J.R., Introduction to Nuclear Reactor Theory, Addison Wesley, (1966).

ANNEX VIII. TRAINING AND OPERATING STAFF OF SUBCRITICAL ASSEMBLY AURES01

VIII–1.INTRODUCTION

The subcritical assembly (SCA) " AURES01 " is operated by the Reactor Physics Department of the Birine Nuclear Research Centre, Algeria. One person, an engineer, is assigned permanently to ensure the routine tasks in this installation. This facility is intended primarily for education and training in nuclear engineering. This is done in two aspects: theoretical and experimental. The theoretical aspect is the validation of computer codes primarily for neutronics such as: MCNP, SCALE / KENO, CITATION and others. The experimental aspect is therefore to achieve the maximum possible experiences by exploiting all the opportunities offered by this facility. The operation of this facility is provided by a multidisciplinary team of five people.

VIII–2.DESCRIPTION OF THE SUBCRITICAL ASSEMBLY AURES01

This assembly uses natural uranium as fuel and light water as moderator and neutron reflector (see Fig. VIII–1). The description of each element is given in Table VIII–1.

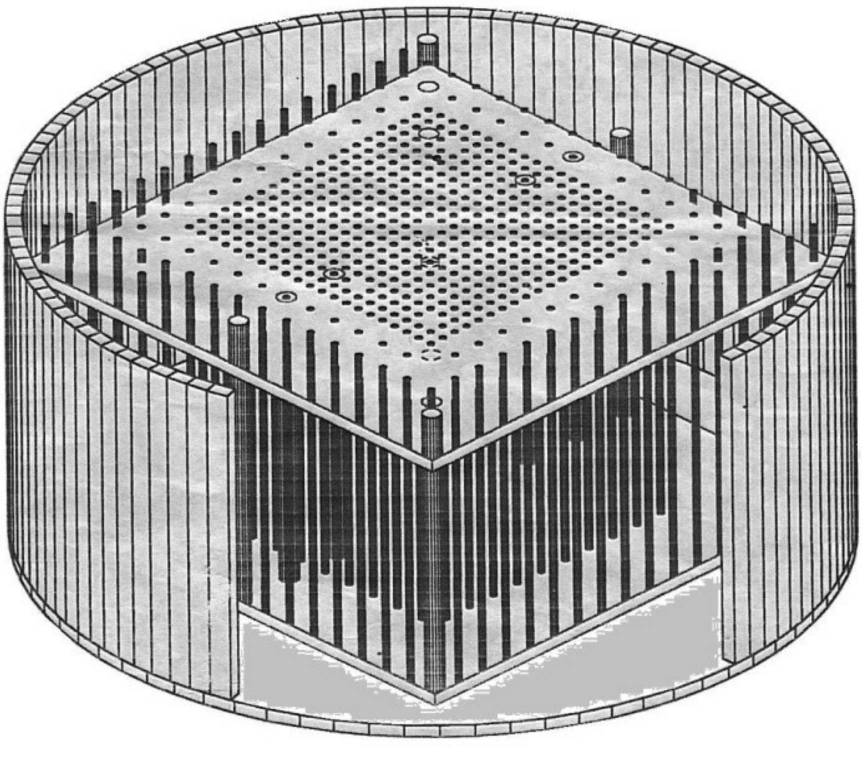

FIG. VIII–1. Subcritical assembly configuration.

TABLE VIII–1. DESCRIPTION OF AURES01 MAIN ELEMENTS

Element	Description	Element	Description
Vessel	Cylindrical shape, Diameter: 2000 mm Height: 1500 mm, Material: aluminium A8.	Spacer grids	Square shapes, Number: 2, Square sides: 1100 mm, Thickness: 20 mm, Located horizontally, spaced 900 mm, Material: Plexiglas
Fuel rod	Form of bars, Length: 1380 mm, Outer diameter: 22 mm Inner diameter: 20 mm, Average weight: 3.1172 kg, Active length: 800 mm with 56 pellets Inactive length: Al, upper and lower with Identical lengths of 310 mm.	irradiation channels:	Cylindrical shape, Material: Al, Inner diameter: 25 mm Length: 1000 mm, Height: 1380 mm, Number: 13
Pellet	Cylindrical shape, Material: natural uranium dioxide UO_2, Height: 14.44 mm, Diameter: 19.48 mm, Clad: Al, thickness of 1 mm, Gap: Helium, thickness of 0.26 mm.	Neutron sources:	Type: Pu-Be, Activity: 74 GBq (2 Ci) Neutron fluence: 3.4×10^6 n / s, Half-life: 24360 y
Control rods	Cylindrical shape, Diameter: 22 mm, Height: 1380 mm, Central region length: 800 mm, filled with B_4C boron carbide powder, Upper and lower region length: Al, with identical lengths of 310 mm.	Array	Square shape, Number: 2, inner and outer, Internal array: 58.8 cm, 420 fuel rods as maximum load, 27 mm of pith, 5 irradiation channels, External array: 96.4 cm side, 160 fuel rods as maximum load, 54 mm of pith, 8 irradiation channels

The calculated and experimental effective multiplication factor is given in Table VIII–2, confirming that the assembly is deeply subcritical.

TABLE VIII–2. CALCULATED AND EXPERIMENTAL EFFECTIVE MULTIPLICATION FACTOR

Case No.	Experiment results	Simulation results	
1	0.78	0.76	CITVAP (2D)
2	0.78	0.79	CITVAP (3D)

VIII–3. INSTRUMENTATION AND ASSOCIATED FACILITIES

The instrumentation and associated facilities of the AURES01 include neutron detectors for the measurement of neutron flux, a ^{10}B proportional counter, a ^{3}He neutron proportional counter, and a storage room, as described below.

Neutron Flux Measurement Instruments

- Neutron detector;
- High voltage (for powering the detector);
- Preamplifier (to eliminate background noise and to the current-voltage conversion);
- Amplifier (for increasing electric level of the pulses delivered by preamplifier);
- Single-channel selector (for eliminating all pulses which are not significant from the radiation to be measured);
- A digital counter (for measuring the number of pulses for some time);
- An analogue meter.

^{10}B proportional counter

- Type NLD INC 232;
- Working point, 650 V;
- Sensitivity 0.5;
- filling gas: Ar + CO_2;
- Boron is introduced into the counter as a solid crystal covering the inner face of the tube wall;
- The proportion of enrichment in the crystal is on the order of 94% ^{10}B;
- Thermal neutrons lead it to its excited state with a stable ^{7}Li. So the α particle and the 0.48 MeV γ-ray photon are emitted in coincidence.

^{3}He neutron proportional counter:

- Shape of a cylindrical tube;
- Length of 6.35 cm, 1.27 cm outside diameter, inside diameter 1.11 cm;
- Gas Volume 6.1165 cm^3;
- Effective microscopic absorption cross-section: $\sigma_a(^{3}He) = 5316$ b;
- Operation of high voltage;
- Gas pressure in the tube of the detector, the 26/06/2005 calculated (at a temperature of 20 °C, the volume being 6.1165×10^{-3} litre): P = 304 cm Hg = 4.052×10^{5} Pa;
- Efficiency is 0.55;
- Sensitivity, s = 1.8.

Storage room

Has two storage cages (left and right), to store the nuclear fuel of the AURES01 facility. The fuel rods, door-detectors and absorbing elements are stored on shelves in this room:

- The number of fuel rods (natural UO_2) is 598.
- They are identified by a stamping by number engraved on the upper cap (upper inactive area).
- The number of control rods is 20.

VIII–4. OPERATING PERSONNEL

One staff member is permanently assigned to the operation of the SCA. Four additional staff members work part-time in operation and utilization, as described below:

One engineer physicist: permanent assignment, working full time with the following tasks:

- Control and monitoring of nuclear materials (IAEA safeguards);
- Ensuring the proper functioning of equipment installation;
- Ensuring the safety and security of the facility;
- Leading the team in executing the operation programme;
- Contributing to the development of practical work procedures;
- Performing calibration of measurement channels.

One engineer physicist: temporary assignment works on the operating programme with the following tasks:

- Participating in the execution of the operating programme;
- Contributing to the development of practical work procedures;

One physicist researcher: temporary assignment works on the operating programme with the following tasks:

- Participating in the execution of the operating programme;
- Contributing partly to the development of practical work procedures;
- Performing physical calculations: e.g. neutron sensing.

One instrumentation researcher: temporary assignment works on the operating programme with the following tasks:

- Checking the instrumentation used in the operation of the programme;
- Managing the maintenance of measurement equipment;
- Contributing significantly to the calibration of measurement channels.

One technician in radiation protection: temporary assignment works on the operating programme with the following tasks:

- Periodically performing the verification of the dose in the different zones of the installation;
- Managing the maintenance of radiation protection equipment;
- Participating in the execution of the operating programme.

VIII–5. EXPERIMENTS AVAILABLE

The SCA allows experimenters to conduct several experiments relevant to education and training in neutron engineering. Some of the experiments available are listed below. Other basic experiments include neutron measurement and neutron flux mapping.

- Evaluation of source emission rate;
- Study of light water shielding behaviour and paraffin;
- Diffusion length measurement;
- Evaluation of the extrapolated distance;
- Measurement of neutron migration length;
- Measurement of neutron age;
- Experimental evaluation of the multiplication factor;
- Measurement of the buckling.

VIII–6. TRAINING PROGRAMME

The 2016-2018 three-year training programme that took place at AURES01 is described below. The training started with 19 trainees in 2016 and increased to 78 in 2018. In these training activities, the following practical works were conducted:

- TP1: Determining the operating parameters of the measurement chain of the neutron flux of the AURES01;
- TP2: Measuring the distribution of thermal neutron flux around the Pu-Be source surrounded by light water; comparison with the theoretical values;
- TP3: Measuring the effective multiplication factor of the multiplier AURES01 array;
- TP4: Measurement of the Laplacian vertical material AURES01 multiplier array;
- TP5: Measuring the effectiveness of AURES01 multiplier array control rods.

VIII–7. CONCLUSION

The SCA AURES01, operating in the Reactor Physics Department of the Birine Nuclear Research Centre, has proven to be a valuable tool for the education of training in the area of nuclear engineering, contributing to capacity building in Algeria.

ANNEX IX. FISSILE MATERIALS LOADING METHODOLOGY

IX–1. INTRODUCTION

The loading methodology for fissile materials, is developed to reduce the loading time, to eliminate the possible of discharging and reloading of fissile material, and provide maximal safety, is described here. The algorithm is based on the calculations performed by modern code for nuclear reactor calculations, for example MCNP. The code allows for the calculation of sensor response to the loading of every portion of fissile material with sufficient accuracy.

IX–2. METHOD OF LOADING OF FISSILE MATERIALS IN A SUBCRITICAL ASSEMBLY

The main rule for loading of a subcritical assembly (SCA) core driven by an external neutron source is a safety run of the curve from at least two measurement channels. It necessitates a uniform displacement of the fuel pins (slug) between the detector and the neutron source usually placed in the centre of the core.

The first and second portions of fissile material has to be less than 10% of the calculated critical loading. If the calculations show that critical loading equals K fuel pins (slugs), the first portion needs to be less than 0.1 K pins (slug) and the second portion needs also to be less than 0.1 K. The pins (slug) will be loaded in the cells.

The third and proceeding portions of loaded fuel each have to be less than 25% of the value of the rest of the extrapolated critical loading, which is calculated from the following equation:

$$m_{extr} = \frac{\left(N_2/N_1\right) m_2 - m_1}{\left(N_2/N_1\right) - 1} \qquad \text{(IX–1)}$$

Then third and following loading will be calculated as:

$$\Delta = 1/4\,(m_{extr} - \mathrm{m}_2) \qquad \text{(IX–2)}$$

or

$$\Delta = \left(1/4\right) \cdot \left(\frac{m_2 - m_1}{\frac{N_2}{N_1} - 1} \right) \qquad \text{(IX–3)}$$

where: m_{extr} is the extrapolation critical number of pins (slugs);
m_1 is the number of pins (slugs) at following to last portion;
m_2 is the number of pins (slugs) in last portion;
N_1 is the register of the counter at the number of pins (slugs) at following to last portion;
N_2 is the register of the counter at the number of pins (slugs) at last portion.

Reactivity of the assembly is defined after every loading of following portion of the pins (slugs):

$$\rho = \frac{(keff - 1)}{keff} \qquad \text{(IX–4)}$$

If, at the loading of any portion of the pins (slugs) (beginning from the third one), the safety run of the curve was not received in two channels, at a minimum the last portion needs to be unloaded, and these pins (slugs) need to be loaded in other cells. Such a procedure needs to be performed to get a safety run of the curve 1/M in two channels at minimum.

After loading of the last portion of the pins (slugs), the effective multiplication factor of this assembly k_{eff} is expected to have a value that cannot exceed maximum possible multiplication factor $k_{eff,max} = k_{eff} + k$ (e.g. 0.98).

IX–3. DETERMINATION OF THE MULTIPLICATION FACTOR BY A RECIPROCAL COUNTING METHOD

The value of the multiplication factor following loading can be calculated by a reciprocal counting method from the readings of the detectors and the intensity of neutron source:

$$N_i = \frac{C_i \cdot S}{1 - k_i} \qquad \text{(IX–5)}$$

The equation without fissile materials in the system can be written:

$$N_0 = C_0 \cdot S \qquad \text{(IX–6)}$$

where: C_i is the part of the readings of the counter from one neutron at the i^th step of loading;

- C_0 is the part of the readings of the counter without fissile materials;
- S is the intensity of neutron source;
- k_i is the multiplication factor at the i^th step of loading;
- N_i is the readings of the counter at the i^th step of loading;
- N_0 is the readings of the counter in the assembly without fissile materials;

It is possible to input the coefficients C_i for different steps of loading and C_0 for the initial state with the external source and without fissile materials, instead of one coefficient C in the standard method. It allows to take into account distinctions in spatial distributions of neutron flux and input of one neutron (averaged) in the readings of the counter at different steps of loading. These coefficients C_0 and C_i have been calculated together with readings of the counters N_0 and N_i according to the scheme of loading. The reactions (n,p) and (n,α) are used in the above mentioned counters of slow neutrons. The cross-sections of these reactions in the thermal region follow 1/v dependence. Thus, the average (in counter volumes) flux densities from one neutron of a source in thermal region C_0 and C_i are used. (It is possible to take into account all neutron spectra. But in this case, k_i needs to be generated by the MCNP code).

Dividing Eqs (IX–5) to (IX–6) produces:

$$k_i = 1 - {}^{N_0}/_{N_i} \cdot {}^{C_i}/_{C_0} \qquad \text{(IX–7)}$$

The level of subcriticality of the system is defined as:

$$\frac{\rho_i}{\beta_{eff}} = \frac{k_i - 1}{k_i - \beta_{eff}} \tag{IX– 8}$$

$$\frac{\rho_i}{\beta_{eff}} = \frac{{}^{1}/_{\beta_{eff}}}{1 - \left({}^{C_0}/_{C_i}\right) \cdot \left({}^{N_i}/_{N_0}\right)} \tag{IX– 9}$$

The value may also be calculated at every step of loading using the MCNP code.

LIST OF ABBREVIATIONS

ADS	accelerator driven systems
DEC	design extension conditions
E&T	education and training
NAA	neutron activation analysis
OLCs	operational limits and conditions
PIEs	postulated initiating events
SAR	safety analysis report
SCA	subcritical assembly
SP	strategic plans
SSCs	systems, structures, and components
RRDB	research reactor database

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Meetings

Technical Meeting, Vienna, Austria 8-12 October 2018
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