History, Development and Future of TRIGA Research Reactors

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HISTORY, DEVELOPMENT
AND FUTURE OF
TRIGA RESEARCH REACTORS

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HISTORY, DEVELOPMENT AND FUTURE OF TRIGA RESEARCH REACTORS

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2016

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FOREWORD

Ever since the training, research, isotopes, General Atomics (TRIGA)[1] reactor concept was developed in the second half of the 1950s, these facilities have played an important role worldwide in contributing to both basic and applied nuclear research. Within the first two decades, approximately 60 TRIGA research reactors of various designs had been constructed and licensed in four continents mainly for academic research and industrial applications. Thousands of scientific articles refer to research performed with TRIGA reactors, many Master of Science theses and doctoral dissertations have been successfully accomplished with experiments conducted at these facilities, and even specific TRIGA topical conferences have been organized, resulting in the development of a unique community, which is now nearly 60 years old. Since its inception, TRIGA technology and the staff involved have arrived at an era when knowledge retention and ageing management has become more crucial than ever if this community and technology are to continue to deliver state of the art basic and applied research. Indeed, in most cases technology experts and staff participating in the startup and operation of their TRIGA reactors will retire or have retired, many ageing reactor components have had to be replaced, development of digital instrumentation and control systems has phased out the original systems based on now obsolete vacuum tubes and obsolete solid state circuitry, and many TRIGA research reactors had even to close down owing to the lack of end users, budgetary constraints or other issues.

The basic incentive for this publication started at the IAEA Technical Meeting on Research Reactor Coalitions: Global Issues of TRIGA Research Reactors, held in Vienna, Austria, from 4 to 8 November 2013. The overall objective of the meeting was to define and establish the initial composition, organization and strategy (road map) of the Global TRIGA Research Reactor Network around goals such as: strengthened regional and global cooperation among TRIGA facilities towards the development of solutions for common issues and challenges; enhancement of TRIGA reactor utilization through common and complementary products and services as well as exchange of experiences and practices; increased viability and visibility of TRIGA reactor operation in the future through contacts and effective relationships with national and international stakeholders; and development of promotional tools advertising TRIGA capabilities and advocating continued operation.

The meeting participants also jointly agreed to initiate the preparation of this publication, which contains compact information and a summary on the founding, history and continued development of TRIGA reactors over the last

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[1] TRIGA is a registered trademark of General Atomics (USA).

60 years, on specific topics such application and utilization aspects, ageing management, core conversion and other fuel issues, and presents challenges and future perspectives of TRIGA research reactors worldwide.

This publication will be a useful source of information for a much broader stakeholder community than just representatives of TRIGA reactors themselves. In addition, a number of individual Member State contributions have been collected and included on the attached CD-ROM to illustrate the historical developments of TRIGA reactors through individual facility examples and experiences.

In the preparation of this publication, the IAEA acknowledges the valuable contributions of the individual authors and of the experts who provided input and reviewed this publication, in particular H. Boeck (Austria), J. Razvi (United States of America) and M. Villa (Austria). The IAEA officer responsible for this publication was D. Ridikas of the Division of Physical and Chemical Sciences.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Training, research, isotopes, General Atomics (TRIGA) constitute a 'class of their own' among the large variety of research reactors that have been installed since the 1950s. Developed in the late 1950s following US President Eisenhower's Atoms for Peace Initiative announcement at the United Nations, many were mainly constructed in the late 1950s through the early 1980s in 23 different countries, and many of them have continued to operate successfully for nearly 60 years.

While many of the reactors exported by the United States of America early on following the Atoms for Peace announcement were the so called materials testing reactor (MTR) technology, the rapid and widespread acceptance of TRIGA technology following its introduction also sparked interest in widely disseminating information on how these reactors were being operated, utilized and maintained. As early as 1970, therefore, the first of many TRIGA Users' Conferences was organized by the TRIGA reactor in Helsinki, Finland, followed by annual, rotating US and European TRIGA Users' Conferences until 2000. In 2002, the first combined World TRIGA Users' Conference took place in Pavia, Italy. Later on, these conferences were added as topical events to the annual International Topical Meeting on Research Reactor Fuel Management (RRFM) conferences organized by the European Nuclear Society, and to the annual conference of the National Organization of Test, Research, and Training Reactors (TRTR), in the United States of America. During the earlier years, General Atomics published proceedings of each of these conferences; the majority of these conferences are now documented in a dedicated database compiled by the Atominstitut, in Vienna, Austria, and are available from the IAEA International Nuclear Information System (INIS) [1]. Reference [1] collects an extensive volume of papers and presentations from the TRIGA conferences from 1970 to 2008, and includes more than 1000 searchable contributions from individual TRIGA facilities worldwide.

As both the front end and the back end of TRIGA fuel became increasingly important to the continued sustainability of TRIGA reactors, the IAEA took the initiative to invite all TRIGA reactor operators to a dedicated Technical Meeting in Vienna, Austria, in November 2013 to discuss important TRIGA issues and challenges, such as utilization, management, technical support, fresh fuel supply and spent fuel options. One of the outcomes of this meeting was the unanimous decision to compile this publication, describing the history, present status and future perspectives of TRIGA facilities worldwide.

1.2. OBJECTIVE AND SCOPE

This publication intends to summarize the information on the past and present of TRIGA reactors, as well as to give an outlook forward, especially in view of potential challenges that need to be addressed by TRIGA operating organizations in the near future. As the staff who participated in the construction and operated their TRIGA facilities throughout the decades have already retired or can be expected to do so in the not too distant future in many of the institutions, there was an urgent need to gather their knowledge from already published sources or available as internal reports and documents into a single publication.

1.3. STRUCTURE

This publication covers the historical development (Section 2) and basic TRIGA characteristics (Sections 2 and 3), followed by TRIGA utilization (Section 4), TRIGA fuel conversion (Section 5) and ageing management of TRIGA research reactors (Section 6). The publication continues with issues and challenges (Section 7), an introduction to the Global TRIGA Research Reactor Network (Section 8) and concludes with future perspectives (Section 9).

The CD-ROM accompanying this publication includes 22 individual papers, describing operational TRIGA facilities worldwide, reporting on their status, key research results, issues and challenges they are facing as well as future plans.

2. SHORT HISTORY

2.1. THE HISTORY

The TRIGA concept has its origins in August 1955, when a large international conference, sponsored by the United Nations on Peaceful Uses of Atomic Energy, was held in Geneva, Switzerland. Subsequent events are recollected personally in the book by F. Dyson [2]. One of the two US organizers of that meeting was F. de Hoffmann, a nuclear physicist employed by General Dynamics Corporation in San Diego, California. After the conference, de Hoffmann persuaded General Dynamics Corporation that the time was ripe for commercial development of nuclear reactors and nuclear energy, which responded by creating the General

Atomic Division in 1956 (now known as General Atomics), with de Hoffmann as its first president.

In June 1956, in the absence of any other site facilities, the General Atomic Division rented a former schoolhouse from the San Diego public schools and convened a group of scientists, mostly veterans from the US Manhattan Project, to consider the commercialization of nuclear reactors (see Figs 1 and 2).
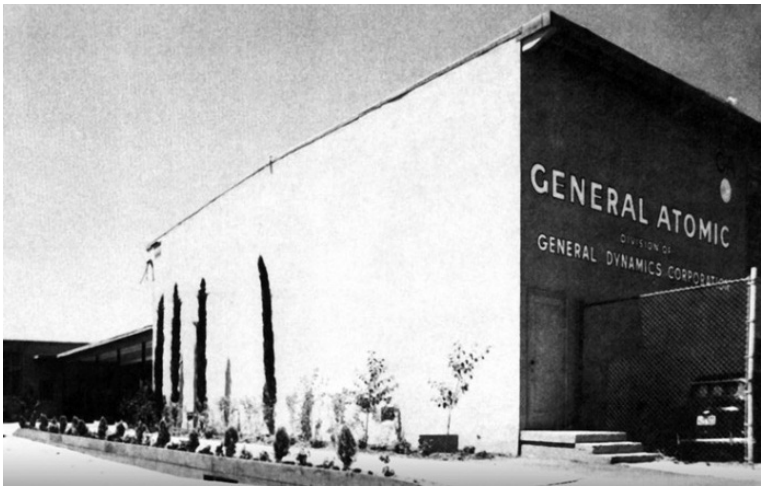
FIG. 1. Barnard School in July 1956 (courtesy of General Atomics).

FIG. 2. A group photo of General Atomics' founders (courtesy of General Atomics).

Three groups were formed, one of which was assigned the challenge of designing a 'safe reactor'. Notably, British mathematician F. Dyson, Iranian metallurgist M. Simnad and US weapons specialist E. Teller joined this group. Dyson recollects in his book that Teller was in charge of this group and insistent that a safe reactor must be one that "could be given to a bunch of high school children to play with, without any fear that they would get hurt" [2]. Furthermore, that "engineered safety was not good enough" and the reactor fuel itself should have inherent safety characteristics for reactivity insertion events [2]. In other words, with the possibility that its control rods could be totally and rapidly removed, the guiding principle adopted was that the reactor would end in a stable condition with no fuel damage or melting.

In his book, Dyson discusses the evolution of the "warm neutron principle", which is described in detail elsewhere in this publication, and which forms the basis of the inherent safety characteristic for reactivity insertion events of TRIGA reactors [2]. (The Dyson Effect, which describes a key component of the kinetic behaviour of TRIGA reactors, was first identified by him.) It took only two years for some experiments with neutron moderation [3], and intense metallurgical work by Simnad [4] to complete a reactor design, manufacture fuel [5], carry out subcritical measurements and create a working reactor. The prototype, named TRIGA Mark I, was first taken critical on 3 May 1958 at the General Atomic Division's new facilities in La Jolla, near San Diego, California (see Figs 3 and 4) and following successful operation, was announced via a press release on 2 June 1958 (see Fig. 5).

FIG. 3. Original 10 kW TRIGA Mark I core (courtesy of General Atomics).

FIG. 4. TRIGA Mark I reactor tank being installed (courtesy of General Atomics).

The initial power level was only 10 kW and was cooled by freon cooling coils, but this was soon raised to operate regularly at 250 kW. It routinely operated at this power level, and still used some fuel from the original 1958 core until it was permanently shut down in 1995. Several years later, full patents on the TRIGA reactor were granted to its early designers (see Figs 6 and 7) [5, 6]. In 1985, this reactor was designated as a Nuclear Historic Landmark by the American Nuclear Society (see Fig. 8).

The idea that control rods could be rapidly removed safely was tested early on. Dyson describes how, in June 1959, as part of formal dedication of the new General Atomic Laboratories, the distinguished invited speaker, N. Bohr, pulled a switch that 'pulsed' the TRIGA reactor to about 1500 MW, returning safely to a modest 500 kW power level immediately thereafter without any human intervention. Interestingly, the highly esteemed group apparently only went to view the reactor after the pulse and did not observe the blue flash of Cerenkov radiation that is now so familiar to all pulsing TRIGA reactor operators.

A remarkable demonstration of the safety and simplicity of the TRIGA reactor was the creation of an exhibition model displayed in the US exhibit at
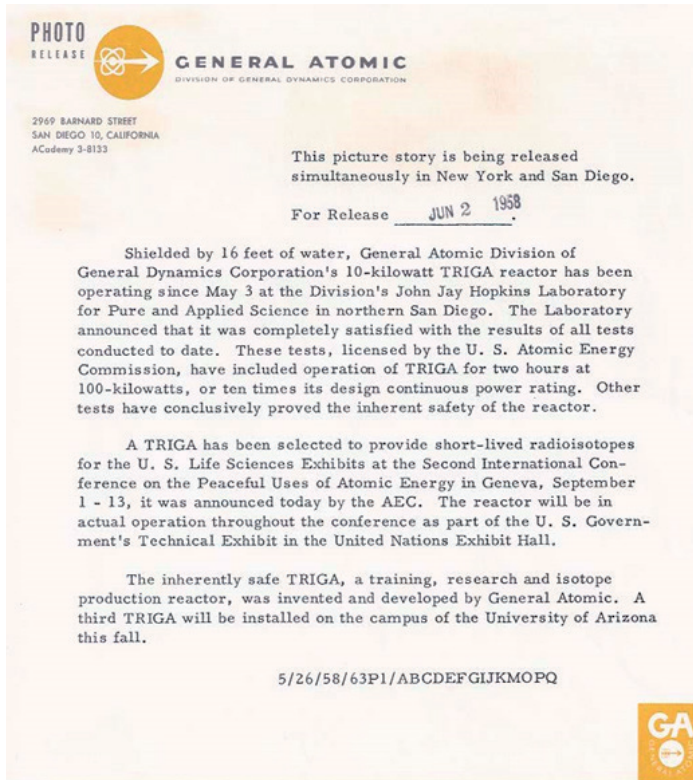
FIG. 5. TRIGA Mark I criticality press release (courtesy of General Atomics).

the Second United Nations Conference on the Peaceful Uses of Atomic Energy, in Geneva, in 1958 [7], and the World Agricultural Conference in New Delhi, in 1959–1960 [8]. In the second of these conferences, the reactor was first pulsed by President D. Eisenhower as he formally opened the US Pavilion (see Fig. 9). Well over 2 million people are thought to have visited that reactor [9].

Immediately thereafter, TRIGA reactors sold briskly. At the same time, the design was rapidly evolving to allow these reactors to operate at higher power levels. In addition to the construction of new reactors, General Atomics also found several reactors that had been operating with different types of fuel were interested in obtaining TRIGA replacement fuel. By the end of 1960, a second, 1500 kW (or 1.5 MW) reactor was in operation at General Atomics' own facilities in La Jolla, California, along with six others. By June 1967, 32 total installations had been accomplished in 13 different countries. In that year, barely 10 years since the first germ of the idea, a symposium held at General Atomics' facilities
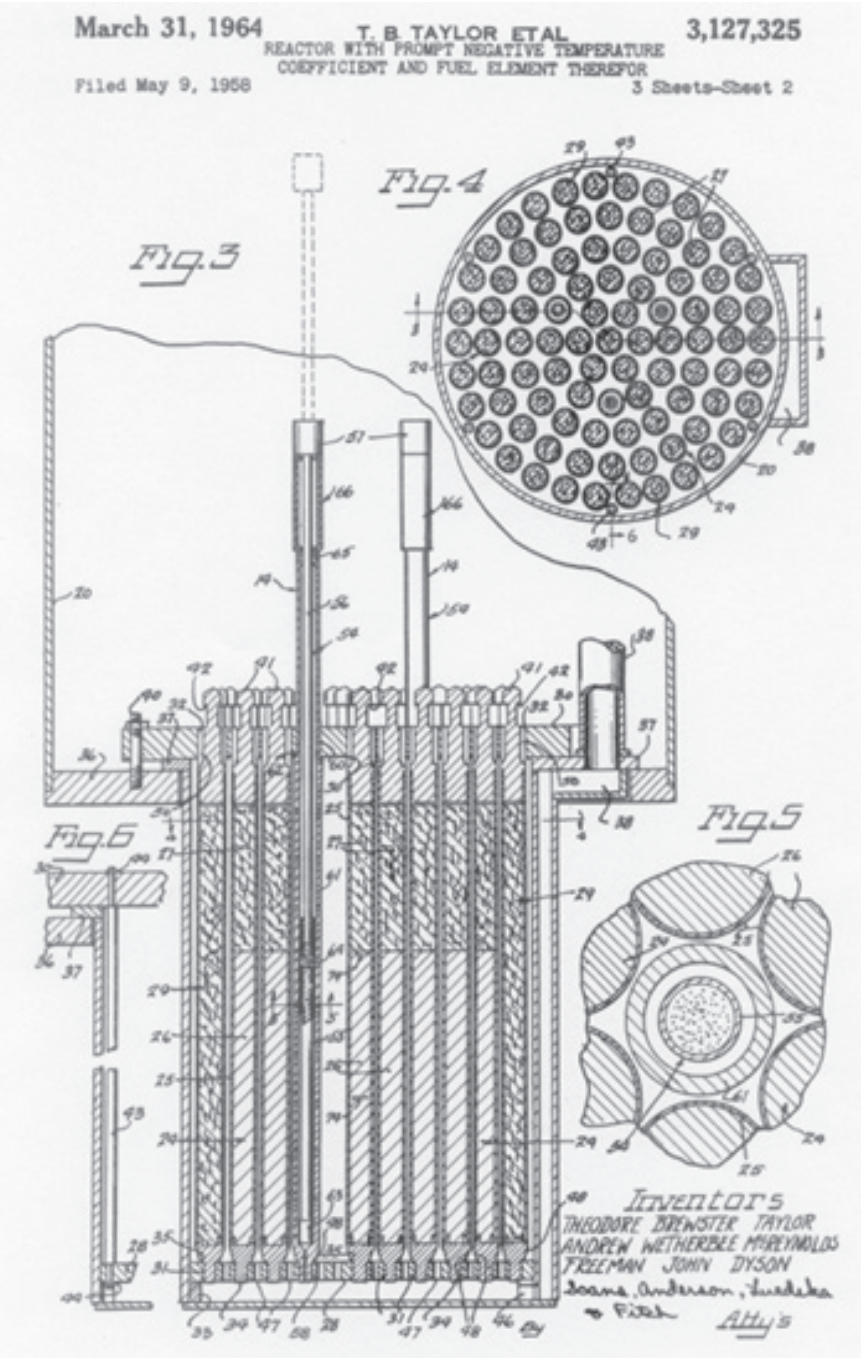
FIG. 6. *Engineering drawings of the reactor core (courtesy of General Atomics).*

Fig 1

SPACER ELEMENT 17

TOP END FIXTURE 11

.125" GAP

MODERATOR 27

BURNABLE POISON 25

LUBRICANT 33

31

FUEL BODY 23

.025" TAPER

BURNABLE POISON 25

MODERATOR 27

BOTTOM END FIXTURE 13

19

21

Fig.2

LUBRICANT 33

29

31

FUEL BODY 23

LUBRICANT 33

BURNABLE POISON 25

MODERATOR 27

Inventor
WALTER P. WALLACE
MASSOUD T. SIMNAD

By Saens, Anderson, Luedeka & Neely
Attys

FIG. 7. *Engineering drawings of the geometry of the fuel element (courtesy of General Atomics).*

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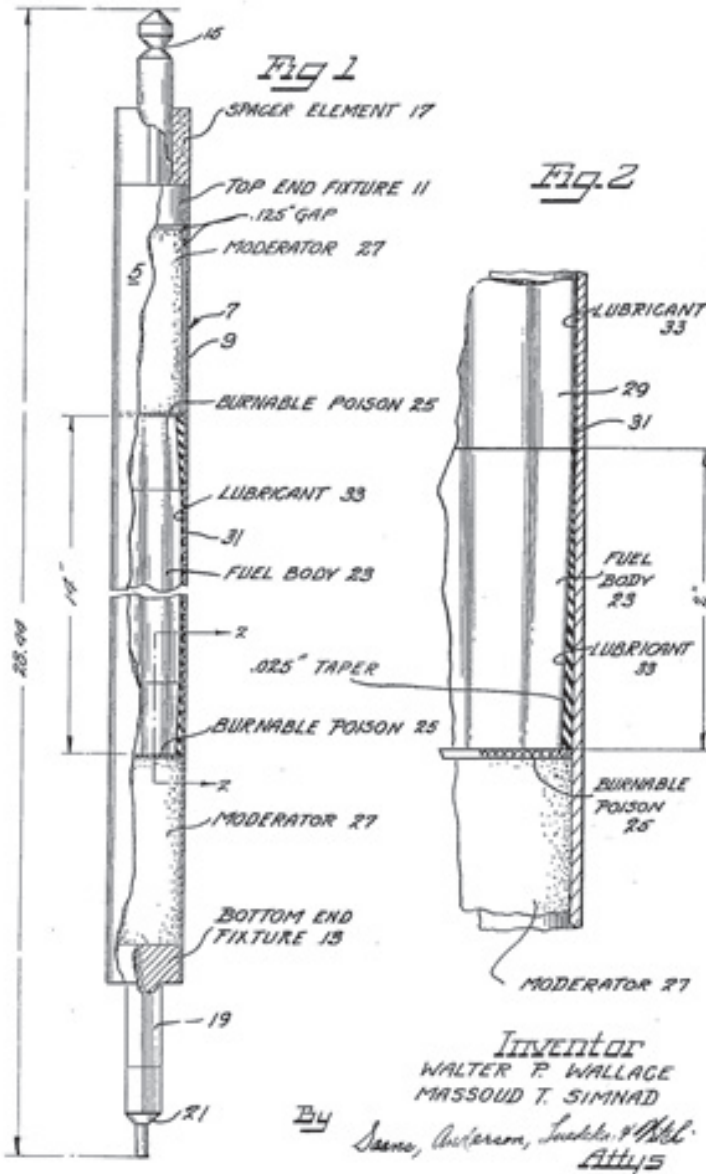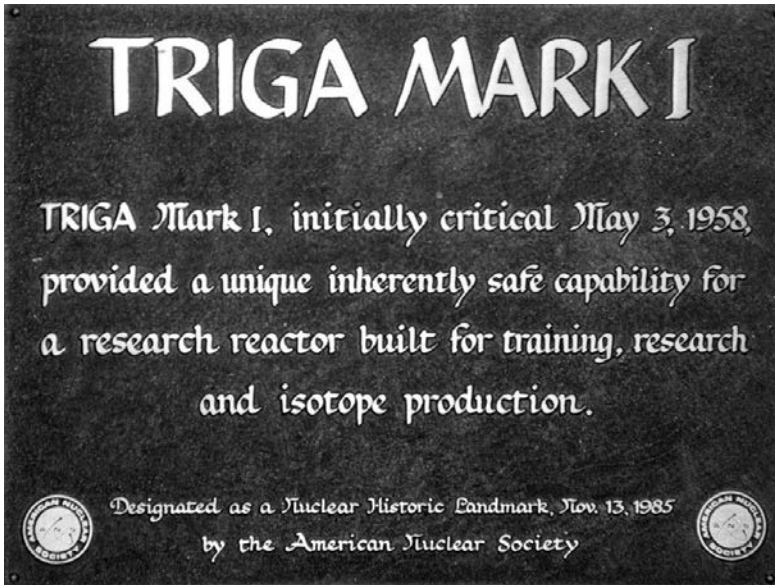
FIG. 8. Mark I historical landmark (courtesy of General Atomics).

FIG. 9. Eisenhower in 1959 at New Delhi Agricultural Fair (courtesy of General Atomics).

in San Diego on research using reactors attracted 18 presentations and described research from 7 TRIGA facilities.

2.2.   TECHNICAL STUDIES FOR TRIGA DEVELOPMENT

A 1958 landmark publication, Technical Foundations of TRIGA [10], details the development of the physical, chemical and engineering aspects that were accomplished in the early development, from 1956, of TRIGA:

"It was therefore felt that if General Atomic, as its initial contribution to reactor technology, could design and develop a novel reactor that was completely and inherently safe, such a design might not only be a technical achievement, but also facilitate the training of scientists and engineers and increase popular confidence in the safety of nuclear reactors to such a degree that progress in the field would be greatly accelerated. If this design had the added features of simplicity and economy, so much the better."

Even in the early days of nuclear reactor technology, Teller's team, which conceptualized the TRIGA reactor design, recognized that safety had to be a prime consideration, and in many cases, safety requirements were met by placing the reactor in remote locations or by enclosing it in specially designed containment buildings. This, however, interferes with the full exploitation of the beneficial uses of this technology, in research and education, as well as in medical and industrial applications, and the needs for these applications can only be adequately met by an inherently safe reactor for reactivity insertion events that can be operated at reasonably high power levels in populated areas such as university campuses, without the expensive method of placing these reactors in containment structures. The fuel elements that are used in TRIGA reactors were developed as a part of a reactor system designed to fulfil these needs.

The uranium–zirconium hydride (UZrH) fuel elements, which have been in common use around the world since 1958, were therefore developed as a part of a reactor system to fulfil these needs. The most important feature of these fuel elements as originally conceived was that they could contain fuel as well as a moderating material in an essentially homogeneous mixture. Hydrogen was chosen as the moderator of choice and zirconium as the 'sponge' to absorb the hydrogen to create the appropriate hydrogenous medium. Since this new fuel moderator material was essential to the new, inherently safe for reactivity insertion events, reactor concept, the process to produce such materials was developed and established early on, and crucial experiments on the proposed reactor core were made at the General Atomics' facilities in La Jolla, California, in the same

facility and adjacent to where the prototype TRIGA reactor was constructed and achieved criticality in 1958. By the time a satisfactory fuel fabrication process had been developed, the experiments had shown that the construction of this prototype reactor was a reasonable next step. This first TRIGA reactor was the result (see Figs 3 and 4). The properties of the fuel moderator material continued to be studied in this first reactor, and by early 1959, two additional TRIGA reactors had been built and operated.

Early work on the development of UZrH fuel elements is described in a paper presented at the First International Symposium on Nuclear Fuel Elements, at Columbia University [11]. A brief synopsis from Technical Foundations of TRIGA [10], which was "devoted to a description of the research and development carried out to make the TRIGA a reality" is presented herein.

2.2.1. Selection of the moderator

The criteria for a 'completely' and 'inherently' safe reactor for reactivity insertion events placed emphasis on a need for the design to have a large, prompt and negative temperature coefficient. Liquid homogeneous reactors have large values for this, but General Atomics decided to investigate a solid fuel design as being preferable for other obvious reasons, particularly chemical stability to high temperatures. In order to provide for such a large, 'prompt' coefficient, it was decided to develop a solid fuel design with a moderating material homogeneously and intimately interspersed with the fuel itself. Several moderators were considered for inclusion, including beryllium, beryllium oxide, graphite, lithium hydride (LiH), sodium hydride (NaH) and zirconium (I, II) hydride ($ZrH_{1.2}$): LiH and NaH were excluded as being too reactive; the others were compared using two group diffusion calculations, assuming a 20% enriched fuel (the presence of significant amounts of ^{238}U was considered necessary to achieve a significant Doppler effect on the temperature coefficient), and a reflector. Cost and toxicity caused developers to drop beryllium and its compounds from consideration. Using zirconium as a homogeneously mixed sponge with the enriched uranium, ZrH seemed promising, as it ensured a required significantly lower critical mass and had excellent thermal conductivity as well as low cost, so further research attention was devoted to ZrH, about whose metallurgy (or neutron moderation properties) little was known in 1956.

2.2.2. Research on zirconium hydride

Theoretical calculations using diffusion models were carried out on a number of moderator and fuel systems to establish reasonable ideas for reactor designs. General Atomics' scientists studied the available diffusion models

carefully, but gave special attention to the case of hydrogen atoms bound in a zirconium lattice, where traditional models were inadequate. Thus, the Einstein model using lattice vibrational energies was developed. In addition, a model originally developed by E. Fermi was exploited to calculate temperature dependencies of moderation in such a lattice. Experimental studies were initiated on: (i) the neutronics of a ZrH moderator; and (ii) the chemistry and metallurgical properties of ZrH and its potential alloys with uranium.

General Atomics' personnel utilized well characterized neutron beams at Brookhaven National Laboratory to measure neutron scattering and thermalization in ZrH_n [3]. They found that true thermalization was much reduced compared to water, but scattering effects gave rise to energy 'peaks' at intervals of 0.13 eV in energy in emergent neutron spectra as a result of scattering from the bound hydrogen atoms. This agreed well with the model calculations and gave confidence in the choice of ZrH_n as a moderator for a safe reactor. The experiments demonstrated that the integral fuel moderator system of UZrH fuel possesses a basic neutron spectrum hardening mechanism to produce the desired characteristics of an inherently safe reactor for reactivity insertion events. This fuel also had a good heat capacity, excellent fission product retention, low chemical reactivity in water and reasonably low hydrogen equilibrium pressures at elevated temperatures.

The key experimental conclusions were:

(a)   For neutrons of energy above 0.13 eV, the moderating ability of ZrH_n is at least as good as that of free hydrogen. Below this energy, it rapidly becomes small.
(b)   Cold neutrons can speed up when passing through ZrH_n by picking up amounts of energy which are integral multiples of 0.13 eV. The higher the hydride temperature, the more readily this process will take place.
(c)   These conclusions seem to be independent of the H:Zr atomic ratio in the hydride.

The experiments and theoretical agreement made gave confidence that the 'warm neutron principle' would indeed function within a homogeneous UZrH fuel system. The theoretical question remaining was whether the negative temperature coefficient of reactivity predicted was large enough to give reactor control. Calculations using the Einstein model led to the estimate of about -3×10^{-5} ($1/k$ dk/dT) at room temperature and twice that at 250°C for the coefficient. Additional effects were anticipated from 'black' poisons and Doppler effects from the presence of significant ^{238}U in the reactor.

2.2.3. First TRIGA critical tests

The experiments and calculations described above gave General Atomics confidence to design a critical system to test the models. This is of sufficient interest to describe in more detail [9]:

"A critical facility was therefore built, in which various parameters, such as water content of the core, uranium enrichment, and H:U ratio, could be varied over a wide range. The facility consisted of a water-filled tank of square cross section into which could be immersed a large number of fuel-moderator assemblies of square cross section. The water tank was surrounded by about 16 in. of graphite on all sides except top and bottom. The fuel-element assemblies consisted of hollow aluminum cans of 1 in. × 1 in. cross section. These were loaded with strips of $ZrH_{1.45}$ of approximate dimensions 0.93 in. × 0.1 in. × 10 in. and with U^{235} and U^{238} foils of the same lateral dimensions as the hydride strips but of thickness 0.001 in., each weighing about 3 g. Graphite bars of 0.93 in. × 0.93 in. × 6 in. were placed above and below each active portion as top and bottom reflectors, and each can was sealed to prevent the entry of water."

Electrical heaters were embedded in the assemblies to allow artificial temperature adjustments to the 'just-critical' system. Measurements were made of the prompt and total (bath) temperature coefficients through long period measurements as the fuel and/or bath temperature and strip arrangements were changed. The results (from -3×10^{-5} to -6×10^{-5} $1/k$ dk/dT over a temperature range of 20–120°C) were in satisfactory agreement with predictions.

Meanwhile, other General Atomics staff were working on understanding the physical and chemical properties of ZrH_n, particularly as the H:Zr ratio changed, and as uranium was included in the solid matrix [12, 13]. Better characteristics appeared as the hydrogen content increased, so the research focused on $ZrH_{1.2}$ to $ZrH_{1.7}$. In addition to basic materials science research, the idea was to rapidly develop a viable fuel design so a prototype reactor could be constructed. The plan was to fabricate a U–Zr alloy and subject this to hydriding. Parameters explored over a range of compositions included density, tensile strength, thermal conductivity, linear expansion coefficient and specific heat. Importantly, the chemical properties of the hydrided material were shown to be such that reactivity towards air and water was markedly reduced compared to metallic zirconium, even when powdered or at a high temperature.

2.2.4. TRIGA fuel elements

For fabrication, the hydriding process itself had to be explored to establish ideal conditions (temperature and pressure) for element fabrication. Grain size for the starting material was found to be crucial to reduce the formation of cracks during the process. In the early fuel, $UZrH_{1.0}$ was fabricated by means of a well defined protocol fully described in the landmark publication Technical Foundations of TRIGA [10]. These first elements were solid cylinders, assembled with top and bottom graphite reflectors and included burnable poison disks within the aluminium alloy cladding. In later elements, this design was modified somewhat, but the basic principles have remained the same. By the early 1960s, General Atomics had extended the development to higher contents of hydrogen, increasing the H:Zr atomic ratio from a nominal 1.0 to 1.6, and replacing the aluminium cladding with stainless steel. All this further enhanced the TRIGA fuel's safety features. The fuel–metal alloy was found to be as robust and as corrosion resistant as stainless steel. The $UZrH_{1.6}$ containing fuel element even made rare cladding failure incidents of no consequence at a TRIGA reactor, and high temperatures as a result of prompt reactivity insertions were of little consequence based on the simple physical principles of this fuel.

The standard TRIGA fuel elements that were manufactured in the first years contained 8.5–12 wt% of nominally 20% enriched uranium. As TRIGA reactors with higher steady state power levels were designed, high enriched versions (from 70% to 93% enrichment) of the standard fuel elements, as well as fuel rods with smaller diameters to accommodate higher power densities, were designed and fabricated; these also contained the burnable poison erbium, as part of the homogeneous fuel moderator, to increase the core lifetime and to contribute to the negative temperature coefficient.

Later, in the late 1970s, General Atomics undertook the development of fuels containing up to 45 wt% of low enriched uranium for the long core life, as proliferation concerns discouraged the continued use of high enriched fuels. These fuels were fabricated successfully with the required hydrogen content and erbium loadings. Fuel qualification testing by irradiation and post-irradiation examination by metallographic, X ray diffraction and other techniques showed that the structural features and irradiation behaviour of these high density fuels were similar to those of the already proven 8.5 wt% and 12 wt% fuels. A special design using cladding 'dimples' was also invented to generate even higher power in pulsing (the Annular Core Pulsing Reactor, ACPR) [7]. Fuel development and manufacturing continued at General Atomics' facility in La Jolla, California, until 1996, after which it was relocated to France under a new company, TRIGA International, formed as a joint venture between General Atomics and the Company for the Study and Atomic Fuel Creation (Compagnie pour l'Etude et

la Réalisation de Combustibles Atomiques, CERCA), a current subsidiary of AREVA, at CERCA's plant in Romans-sur-Isère, France, where other types of research reactor and power reactor fuel are also manufactured.

2.3. THE EARLY TRIGA REACTORS

It was decided to limit the power level produced for the first TRIGA reactor to 10–100 kW. This was considered suitable enough for training physicists and chemists, making short lived radioisotopes, performing neutron activation analysis and creating neutron beams for neutron physics studies. Furthermore, at such a power level safety concerns would be minimal as would any required instrumentation and control (I&C) systems, and only a low excess reactivity would be required to overcome the poison buildup and the prompt negative temperature coefficient of reactivity which contributes to the inherent safety for reactivity insertion events of a $UZrH_n$ fuelled reactor. In addition to the safety response in the event of a power excursion, the advantages of the new fuel type were considered to include the compactness of the core, giving higher fluxes for the given power level compared to other reactors available at that time. It was also decided to include several experimental facilities with this first reactor — a rotating specimen rack, a pneumatic transfer system for rapid transfer of samples for studying short half-lives, and a central thimble for irradiations and the production of neutron beams. The core also included a graphite reflector surrounding the cylindrically arranged core of elements. The first reactor was installed in a tank at the grounds of the newly built General Atomics campus on the Torrey Pines Mesa, near San Diego, California, which had been named the John Jay Hopkins Laboratory for Pure and Applied Science. Other than the installation of freon cooling coils within the water tank, the original Mark I design was almost the same as that used in subsequent early, as well as more recently built TRIGA facilities. History seems obscure as to when the first actual TRIGA 'pulse' requiring a prompt reactivity insertion of greater than 1.00$ was actually fired, but as stated earlier, one of the early pulses recorded was during the inauguration of the new General Atomics campus, where Bohr, in formally dedicating the permanent campus in 1959, pushed a button remotely to pulse the reactor several hundred yards away. Before this first TRIGA was shut down permanently in 1995, it had recorded almost 13 000 pulses.

The prototype TRIGA that was installed on the permanent campus achieved initial criticality on 3 May 1958 and closely reproduced its theoretical design parameters. This design and an above ground design — dubbed the Mark II — with beam port installations were subsequently offered to customers. The second TRIGA was operated in the US government exhibit at the Second Geneva

Conference for the Peaceful Uses of Atomic Energy, 1–13 September 1958, and the third TRIGA started up at the University of Arizona on 7 December 1958 — all within the same first year. Two more TRIGA sales were announced directly at the Geneva Conference [7]. In a public ceremony at the Palais des Nations, the Italian National Committee for Nuclear Research (Comitato Nazionale per l'Energia Nucleare, CNRN) formally signed a contract for the installation of an above ground TRIGA at CNRN's new research centre in Rome, which started operations in 1960 (see Fig. 10).

Three days later, Viet Nam announced its selection of a TRIGA Mark II for the country's first reactor, which achieved criticality in 1963. And finally, the University of Lovanium, in the now Democratic Republic of the Congo, made arrangements to acquire the actual TRIGA that operated in Geneva, which subsequently was shipped from Geneva to Léopoldville (now Kinshasa) (see

FIG. 10. TRIGA Mark II in Rome, Italy (courtesy of ENEA).

Fig. 11) and became the first nuclear reactor to be installed and operated on the African continent. The first above ground TRIGA research reactors to go critical were at the ENEA Centre in Rome and the University of Illinois in 1960. Within three years, this popular design was also installed at facilities in Austria, Finland, Japan, the Republic of Korea and Cornell University, United States of America.

Following the initial Mark I prototype, and over the next several years, General Atomics installed two more prototype reactors (known as TRIGA Mark F and TRIGA Mark III) at their facility in San Diego to support their testing and development programmes. The Mark F operated from 1960 to 1993 at a steady state power level of 1.5 MW, and its licence allowed maximum transients of up to 5.50$ reactivity insertions, allowing the performance of $UZrH_n$ fuel for later higher power TRIGA research reactors to be tested under extreme operating conditions. The 2 MW steady state TRIGA Mark III was initially built at General Atomics to test thermionic fuel for space power applications and was the first TRIGA to operate with the high enriched Fuel Life Improvement Programme (FLIP) fuel design with a burnable poison. It was shut down in 1972. Both of these reactors provided the prototypes for a moveable core design. Several of these designs were immediately commissioned at other facilities, notably one at the US Armed Forces Radiobiological Research Institute, in Bethesda, Maryland, with its unique exposure facilities for radiological testing benefiting from the moveable core design. Descriptions of these and many of the subsequent facilities installed worldwide are provided in Ref. [8].

FIG. 11. Arrival of research reactor components from Geneva to Léopoldville (Kinshasa) (courtesy of General Atomics).

In addition to complete reactors, General Atomics also designed and contracted to supply $UZrH_x$ fuel assemblies to a number of existing reactors that had previously been using other fuel types. Most of these non-TRIGA reactors used fuel in the form of thin plates (which was known as the MTR), and so the design of the new TRIGA type fuel assemblies had to be adjusted to fit into the existing reactor core structure as a one for one replacement. The first of these conversions was the existing MTR type research reactor at Pennsylvania State University, United States of America. To accommodate the conversions, General Atomics designed $UZrH_x$ fuel elements with slightly different dimensions than the standard TRIGA core, which were typically held together in four rod clusters, whose dimensions were designed to fit in the older core configurations. One of the key issues was ensuring adequate spacing of the individual fuel rods to ensure sufficient water surrounding each element to provide the necessary cooling and the moderation to make the 'warm neutron' concept as effective as in the original TRIGA reactors.

In addition to the supply of fuel assemblies, related reactor core structures, systems and components, a key subsystem which General Atomics successfully developed, manufactured and supplied, both to TRIGA installations as well as to some non-TRIGA installations, was components as well as complete systems for reactor I&C. These included control rod drives for both, steady state and pulsing operations, as well as nuclear instrumentation specifically designed to record the short duration (ms) pulses from transient operations. Early versions were of course based on vacuum tube technology, and in the 1980s General Atomics was one of the first suppliers to introduce digital I&C systems based on personal computer technology. The evolution of I&C systems for TRIGA reactors is further described in Section 6.

Considerable information regarding the history and development of the various types of TRIGA reactor and fuel is now available: the IAEA maintains a web site devoted to TRIGA reactors as part of its on-line training programme[1]; the World TRIGA Network, known as the Global TRIGA Research Reactor Network (GTRRN) hosted by the Institut für Kernchemie of the Johannes Guttenberg University, Mainz, Germany, also has information and links concerning TRIGA reactors[2]; the current situation status for TRIGA is also given by General Atomics on their web site[3]; and typically each individual facility provides specific information on their installations on their respective institutions' web sites.

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[1] See https://ansn.iaea.org/Common/documents/Training/TRIGA%20Reactors%20%28Safety%20and%20Technology%29/chapter1/characteristics11.htm

[2] See http://triga-world.net/index.html

[3] See http://www.ga.com/triga

3. TECHNICAL CHARACTERISTICS OF TRIGA REACTORS

3.1. TRIGA REACTOR DESIGNS

General Atomics has developed six research and test reactor designs; all of which based on the principle of inherent safety for reactivity insertion events utilizing $UZrH_x$ fuels, and all are under the registered trademark TRIGA. Over the years, following the successful, initial demonstration and validation of the concept with the prototype, General Atomics has developed and provided larger and more complex reactor designs to accommodate increasingly complex irradiation experiments and requirements. However, these changes were still rooted on the original General Atomics concept from 1958 [4] of a walk away safe reactor. While the reactor features can (and often do) vary significantly, the common feature linking all the TRIGA type reactors is the inherently safe for reactivity insertion events thanks to the $UZrH_x$ fuel moderator matrix material.

The basic TRIGA reactor design is an open pool, light water moderated reactor, using a homogeneously mixed fuel moderator fuel element design. All six basic designs will be presented in roughly chronological order that General Atomics made them available to potential customers. Alterations and special additions were common and may be mentioned where they pertain to parameters for existing reactors, but this discussion is not exhaustive.

3.1.1. TRIGA Mark I reactor design

The first TRIGA reactor concept developed by General Atomics (patent issued in 1964 [6]) was the below ground Mark I (General Atomics chose to use the branding of Mark to distinguish one TRIGA reactor type from another). This reactor — as do all other TRIGA type reactors — used the $UZrH_x$ fuel moderator elements described previously inside a water cooled reactor tank. The underwriting principle of these reactors was to keep them as simple and easy to build and use as possible. As such, the below ground level Mark I reactor core was a fairly small, graphite reflected, open pool configuration, and cooled by natural convection flow through the core with demineralized water. The original reactor configuration is shown in Fig. 12.

The core had a fixed cylindrical geometry, approximately 0.58 m × 0.45 m in diameter surrounded by a graphite reflector. The core consisted of a lattice of fuel moderator elements, graphite dummy elements and three control rods located within 90 element positions (i.e. 90 position holes located in a circular array on the bottom and top grid plates). The two grid plates held the fuel moderator
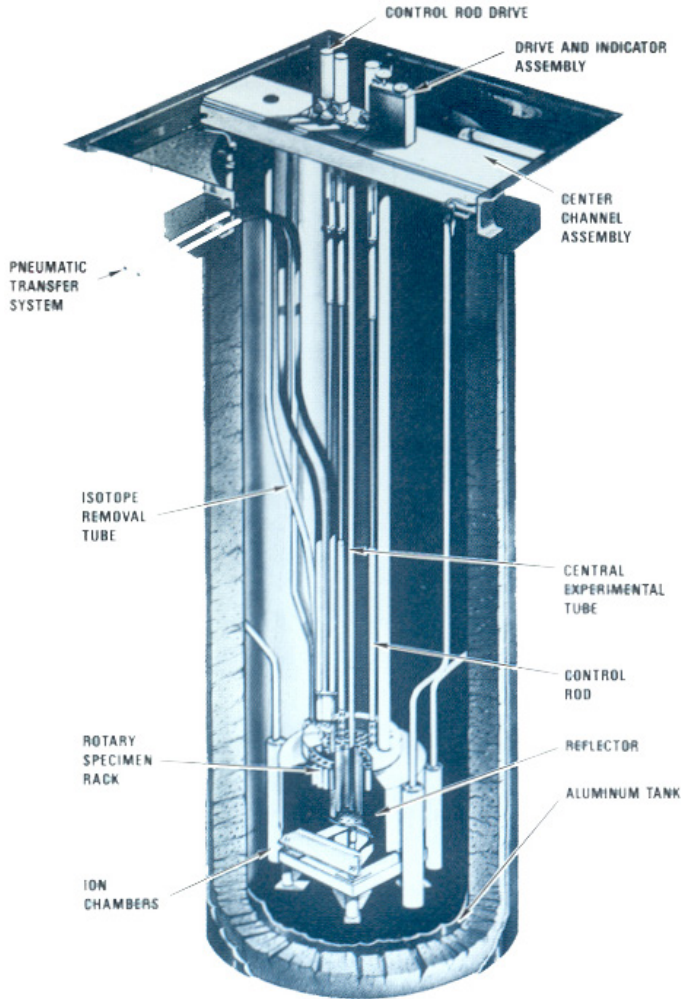
FIG. 12. Cutaway view of the Mark I core and reflector assembly (courtesy of General Atomics).

elements in a vertical position and allowed cooling water flow to flow by natural convection around the fuel elements to remove the heat generated from fission. In addition to the 90 element positions, there was a central chamber (38.4 mm diameter) located at the centre of the core and 16 foil insertion holes (8 mm diameter) inside the reactor core configuration. The central chamber was used for irradiating larger sample specimens, using an irradiation thimble, or it housed the transient control rod for pulsed operation (discussed in Sections 3.1.5 and 4.2.3).

Mark I cores rest on an aluminium reflector platform which is bolted to the bottom of a 6.3 mm thick aluminium tank with outside dimensions measuring 6.35 m × 1.98 m in diameter. The aluminium tank rests below ground and is filled with demineralized water, which provides approximately 4.87 m of water as vertical shielding [14]. The core lattice is an open configuration allowing the natural circulation of the surrounding demineralized water to provide natural convective cooling to the core during operation.

An annular graphite reflector surrounds the core. This reflector has an inside diameter of 45.7 cm, an outside diameter of 76.2 cm and is 55.9 cm tall. The graphite is sealed in a watertight aluminium can. Graphite is also used as a reflector on the top and bottom of the core. However, the graphite is contained in the fuel elements as 10.2 cm tall graphite plugs located in the top and bottom of each element. This provides an effective 10.2 cm thick axial reflector and a 30.5 cm thick radial graphite reflector.

Typical core components include a mixture of fuel moderator elements, graphite dummy elements, source elements, and three or four different types of control rod (regulating, shim, safety and pulsing rods, if applicable) that were all originally clad in aluminium. This outer cladding was soon changed to stainless steel to offer a higher temperature and strength cladding material.

The various control rod types (for fuel follower control rods, see Section 3.2.3) allow the TRIGA reactor to be operated in different modes: steady state, square and pulsed operations. Boron carbide (B_4C), either as powder (in the earlier designs) or as solid pellets (in later designs), is encapsulated in an aluminium or stainless steel cylinder as the neutron absorbing material in the control rods providing a thermally and chemically stable material matrix. Each control rod type is approximately 51 cm long with different radial diameters. The regulating rod has a diameter of 2.22 cm, while the shim and safety rods are 3.18 cm in diameter. The safety rod is completely withdrawn from the core during normal operations and is dropped into the core during a scram event to shut down the reactor under steady state operation. Both the safety and shim rods have a reactivity worth of approximately 2.6% $\delta k/k$ each. The regulating rod typically has a lower worth of approximately 0.7% $\delta k/k$.

Typically, four control rods are required for reactivity control in TRIGA reactors with thermal power levels of less than 1 MW [15]. Two or more motor driven rods are used during steady state operation. To produce pulsed operation in the reactor, a special high speed pneumatic rod drive mechanism is used instead of the standard mechanical safety rod drive. Using this pneumatic equipment, the high speed rod is removed from its position in about 0.1 seconds. For high power pulses, this rod can be located in the central grid position in the core (A-1) or may be located at another grid location. The transient rod is made from a solid matrix of borated graphite rather than the standard powdered B_4C material.

The neutron flux (power level) is monitored using up to four instrumentation channels for which neutron sensitive radiation detectors are located just outside the core. The bulk pool water temperature is typically measured with a probe submerged below the pool, and pool water conductivity is measured at the inlet and outlet of the demineralizer equipment.

3.1.2. Modifications and additions to the original Mark I design

As described in Section 3.1.1, the original prototype reactor built at General Atomics was a 10 kW Mark I type reactor using a minimum of fuelled elements. However, this was soon upgraded to steady state power levels up to 250 kW, with pulse power levels utilizing a transient rod in the central core position, and which achieved peak powers of 1000 MW using about 70 elements in the reactor core. This was the first TRIGA reactor configuration commercially available and sold to customers. Fourteen reactors with these original power level ratings were sold overall to customers in the United States of America and worldwide. The typical number of fuel elements in a Mark I core ranged from 60 to 87 elements, with the approximate numbers of fuel moderator elements needed to achieve the power levels shown in Table 1. These requirements are based on operating experience and can vary depending on fuel type utilized and operating cycles, among other things.

TABLE 1. TYPICAL NUMBER OF FUEL ELEMENTS
REQUIRED FOR VARIOUS REACTOR SIZES

Reactor power (kW)	Fuel elements required (av.)
<100	60
<600	65
<1500	80
1500	90
2000	100

3.1.3. TRIGA Mark II reactor designs

The Mark II reactor is basically an above ground version of the Mark I reactor with the addition of four horizontal irradiation ports for extended experimental capabilities [15]. A concrete shell houses the core tank and irradiation beam tubes providing both structural support and radial irradiation shielding. Except for the horizontal beam ports, the core design is identical to the Mark I, with a fixed core, graphite reflector (but having penetrations to accommodate neutron beams originating directly from the core), open pool configuration and natural convective cooling (up to thermal power levels of 2000 kW) with demineralized water.

The ability to perform irradiation experiments had become the primary function and feature of TRIGA reactors, and General Atomics wanted to enhance this capability in all its Mark II reactor designs. Therefore, the irradiation beam ports as well as graphite thermal column were added to the original TRIGA Mark I reactor core design. Some of the Mark II designs also featured a bulk irradiation tank diametrically opposite to the thermal column. Figure 13 shows a schematic of Mark II reactor layout showing thermalizing column and neutron
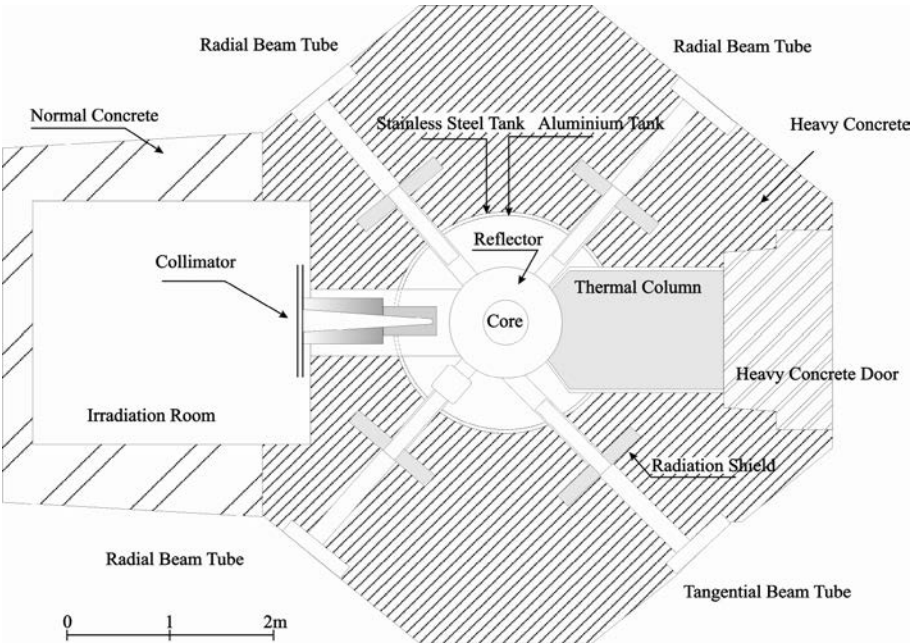
FIG. 13. Schematic of Mark II reactor layout showing thermalizing column and neutron beam ports.

beam ports. These irradiation ports necessitated that the reactor core be above ground to allow access to the beam ports for experimental studies.

Figures 14 and 15 show typical Mark II installation with and without the bulk irradiation tanks. The following components can be easily identified:

(a)   Thermal column: The thermal neutron column was a graphite filled space approximately 1 m × 1 m × 1.7 m long. The column extended through the concrete shield to the graphite reflector surrounding the core. This column provided a maximum thermal neutron flux of the order of 4.0×10^8 cm$^{-2}\cdot$s^{-1} (at 2 MW steady state power level).

(b)   Beam ports: Four horizontal beam ports (15.24 cm diameter) were positioned around the reactor core. These ports permitted the extraction of core radiation or allowed irradiation samples to be inserted next to the core for maximum exposure. Two of the beam ports extended into the reflector edge, one port penetrates the reflector to the actual core and the fourth port is tangential to the core, as shown in Fig. 13. A maximum thermal neutron flux of 3×10^9 cm$^{-2}\cdot$s^{-1} is achieved inside these ports (at 2 MW steady state power level).

FIG. 14. A typical TRIGA Mark II installation (courtesy of Johannes Gutenberg University, Mainz, Germany).

FIG. 15. TRIGA Mark II design with a bulk irradiation tank (courtesy of Kansas State University, United States of America).

In all other respects, except the irradiation beam ports and columns, the Mark II reactor is similar to the Mark I design, as described in Table 2.

TABLE 2. GENERAL MARK I AND MARK II REACTOR DATA

Parameter	Original configuration	Later configuration
Core type	Below ground, fixed core	Below ground, fixed core
Reflector	Water and graphite	Water and graphite
Operating power	Steady state = 18–250 kW Pulsed $\leq$ 1 MW	Steady state = 100 kW–2 MW Pulsed $\leq$ 6 400 MW (1.5 ms)
Neutron flux (steady state)	Thermal $\leq 1.6 \times 10^{12}$ cm^{-2}·s^{-1}	Thermal $\leq 8.0 \times 10^{13}$ cm^{-2}·s^{-1}

3.1.4. TRIGA Mark III reactor designs

The Mark III reactor is essentially a combination of the TRIGA Mark II and Mark F reactors; the Mark III and Mark F both utilize a moveable core design where the core structure is suspended from the reactor bridge mounted on top of the reactor pool, an exposure room on one side of the reactor pool and beam ports on the opposite side of the pool. Some of these design concepts originated with the Advanced TRIGA Prototype Reactor (ATPR) programme [16]. Figures 16 and 17 are schematic illustrations of the Mark III reactor showing: (i) the top reactor pool layout and (ii) the cross-section through the pool layout. This particular reactor concept takes elements from previous TRIGA reactor designs to maximize high energy neutron and gamma ray irradiation experimental capabilities [17].

The designs utilize a moveable reactor core which can be moved and operated in two principle operating positions inside the reactor tank, similar to the TRIGA Mark F design. However, on one end of the reactor tank multiple fast and thermal neutron beam columns are located to allow multiple irradiation experiments to be conducted simultaneously. A high radiation dry exposure room is located at the other principle operating position. Similar to the Mark I and II cores, an isotope production facility (rotating specimen rack) surrounds the core.

The port configuration contains four transient beam ports (20.3 cm diameter), four radial beam ports (15.2 cm diameter), a vertical thermal neutron column and a large horizontal thermal column adjacent to one of the
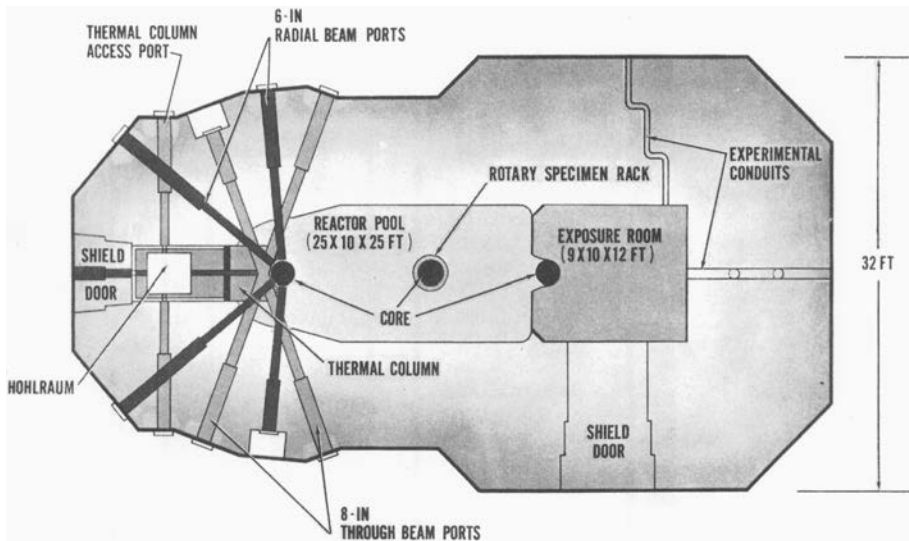
FIG. 16. Mark III reactor showing top reactor pool layout (courtesy of General Atomics).
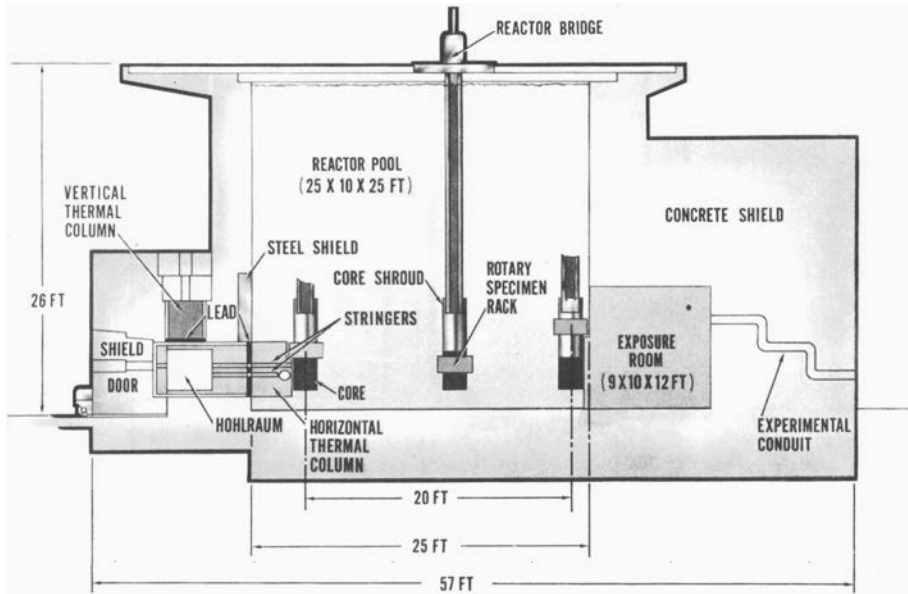
FIG. 17. Mark III reactor showing the cross-section through the pool layout (courtesy of General Atomics).

principle operating positions, as shown in Fig. 16. The walk in exposure room is 3 m × 3.7 m × 2.7 m to accommodate very large experiments [17].

The aluminium reactor pool is contained and shielded in an above ground concrete shield structure approximately 10 m × 17 m × 8 m. The core is suspended, thanks to a moveable bridge platform, near the bottom of the 7.6 m × 3.0 m × 7.6 m large reactor pool, as shown in Fig. 17. Approximately 6.1 m of demineralized water above the reactor provides vertical shielding, while the concrete structure provides radial shielding.

The Mark III design's basic safety characteristics are derived from the same types of $UZrH_x$ fuel moderator elements as the core designs of all other TRIGA reactor types, and it has an open pool configuration, is cooled by natural convection for thermal power levels up to 2 MW and is completely water reflected. However, the actual pool configuration is larger than the standard Mark I or Mark II designs to facilitate the moveable core design [17]. The original design of the Mark III core uses a standard Mark I/II core lattice arrangement, with an additional ring of elements added on the outside to increase the excess reactivity of the core, and that are loaded with $UZrH_x$ fuel elements, graphite or a combination of the two. Such a larger core design is now standard on all TRIGA reactor types.

The TRIGA Mark F and Mark III reactors were designed for steady state power levels in the range of 1–2 MW, and pulsed power levels up to 6400 MW(th) have been achieved when testing fuel at General Atomics' prototype TRIGA Mark F. Typical steady state flux levels are approximately 6.5×10^{13} cm^{-2}·s^{-1} (thermal and fast neutrons) in the central thimble. A further design enhancement which allowed more stable operation at power levels up to 2 MW in natural convection was to reconfigure the layout of the individual fuel moderator elements from a circular array of concentric rings to a hexagonally laid out array for improved thermal-hydraulic characteristics in the fuel region of the core. Figures 18 and 19 show typical installations of the moveable core designs.
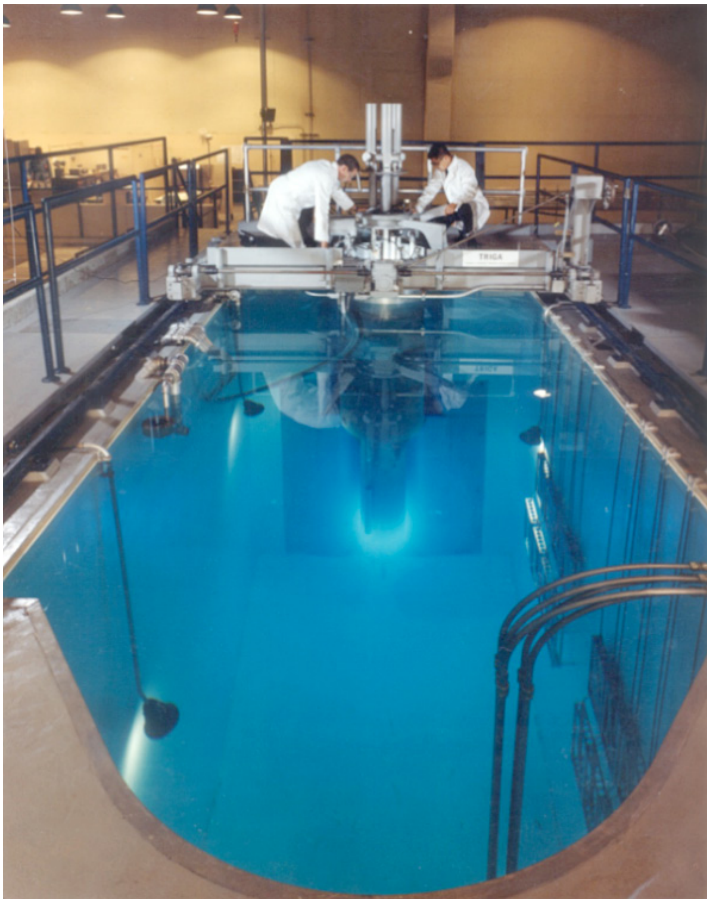
FIG. 18. Moveable core installation at Northup Grumman Corporation, United States of America (courtesy of General Atomics).

FIG. 19. TRIGA Mark III shield structure and exposure facility at University of California, Berkeley, United States of America (courtesy of General Atomics).

3.1.5. Annular core pulsed reactor design

The ACPR concept was designed to take maximum advantage of the pulsing characteristics of the TRIGA core. The function of the ACPR is to study the transient behaviour of materials to intense radiation for short time durations. They employ specially designed 'dimpled' cladding for the $UZrH_x$ fuel elements, permitting higher peak fuel temperatures, while retaining the inherent safety and simplicity of natural convection cooling in an open pool configuration.

The large radiation levels are accomplished using a large, dry central test cavity that can accommodate large samples (22.9 cm). This open cavity provides a high flux area where a maximum neutron flux ($\sim 2 \times 10^{17}$ cm$^{-2} \cdot$s^{-1}) can be generated at a peak power level of 22 000 MW(th) [18] with a time duration of approximately 1.2 ms (see Table 3).

The reactor can be operated at normal steady state power levels of 500 kW and transient levels of 22 000 MW(th) [21]. A typical core configuration for an ACPR design is shown in Fig. 20.

TABLE 3. TECHNICAL SPECIFICATIONS FOR THE ACPR (PITEŞTI, ROMANIA)

Technical specification	Measurement units		
Steady state power	500 kW Wide steady state power range from mW to MW Suitable for tests and calibration measurements for very low power levels		
Pulse operation			
Max. pulse power	20 000 MW		
Min. period	1.2 ms		
Pulse width	4.6 ms 1/2 pulse		
Fuel type	12 wt% $UZrH_x$ fuel		
Fuel enrichment	19.9 wt% ^{235}U		
Cladding material/diameter	Dimpled stainless steel / 3.56 cm		
Rods number	146 + 6 fuel followers		
Max. thermal flux	1.0×10^{17} cm^{-2}·s^{-1} (1.0×10^{15} n·cm^{-2} fluence per pulse)		
Max. fast flux	1.0×10^{17} cm^{-2}·s^{-1} per pulse		
Control rods			
General description	Type	Fuel follower type	
	Poison material	Natural B_4C	
	Number	6	
	Rod drive	Rack and pinion, electromagnetic connection with the control rod	
Transient Rods	Number	2 fast transient rods 1 adjustable transient rod	
	Type	Air follower	
	Poison material	Natural B_4C	
	Rod drive	Fast: pneumatic Adjustable: rack and pinion drive	
Pool			
Pool dimensions	10 m depth, 5 m width, 9 m length		

Source: References [19, 20].

FIG. 20. The TRIGA ACPR core at Tokai Mura, Japan, showing the dry central irradiation cavity (courtesy of General Atomics).

3.1.6. High power TRIGA reactor designs

The multipurpose reactor (MPR) design is the highest power TRIGA reactor concept ($\geq$ 5 MW steady state power) and departs from many of the traditional core configurations from previous core designs. At this time, the MPR-16 reactor in Romania is the only operating reactor in this class of TRIGA reactors (see Fig. 21). The reactor continues to use the primary TRIGA fuel UZrH fuel moderator matrix concept (i.e. inherently safe for reactivity insertion events). However, the core configuration is more compact and thus has higher power densities to produce steady state power levels from 5 MW to as high as
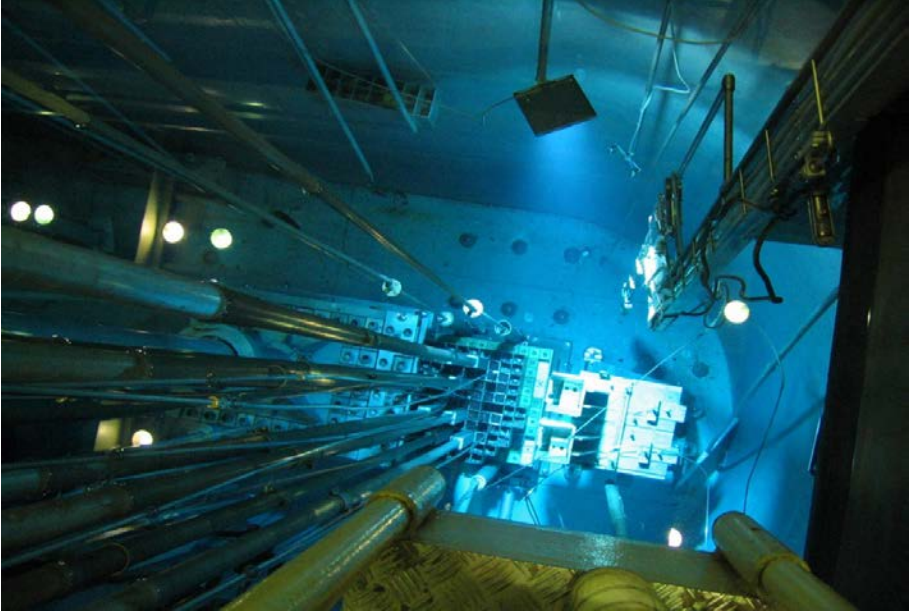
FIG. 21. MPR-16 steady state core configuration at TRIGA reactor in Piteşti, Romania (courtesy of Institute of Nuclear Research, Piteşti, Romania).

20 MW for continuous irradiation experiments [22]. (Cores with steady state power levels above 3 MW are not considered pulsing cores.)

To achieve these higher power densities, the core configuration is square shaped, compact and uses square shaped fuel rod 'bundles' inside the rectangular core assembly, as in Table 4. Hence, some of the specifications in Table 4 may be unique to that reactor rather than typical of the MPR design. These 8.9 cm^2 cross-section fuel rod bundles contain square arrays of 13 mm diameter UZrH$_x$ fuel moderator pins [8]. The bundles can contain 16 (4 × 4), 25 (5 × 5), 36 (6 × 6) or 49 (7 × 7) fuel pins depending upon the desired power level of the reactor.

The higher power densities achieved through these compact fuel bundles requires the core to use forced convection cooling via a primary circuit with four pumps and three heat exchangers to eliminate the heat from the centre of the reactor. The forced convection cooling still occurs at ambient pressures and does not require sophisticated pumping/pressure systems. The water from the reactor pool is simply forced into the reactor assembly by pumps through the top of the core grid rather than using natural convection as previous designs have used in the past. The MPR-16 and ACPR in Piteşti, Romania, have the unique characteristic of being operated in the same open pool (see Fig. 22).

TABLE 4. TECHNICAL SPECIFICATIONS FOR THE MPR-16 REACTOR (PITEŞTI, ROMANIA)

Technical specification	Measurement units
Nominal power	14 MW (steady state) Tested and licensed for 16.5 MW operation
Fuel (all values are nominal)	
Type	LEU 45 wt% $UZrH_{1.6}$–Er
Fuel enrichment	19.97% ^{235}U (originally commissioned with HEU — 93% ^{235}U)
Clad material	Incoloy-800
Clad diameter	13.72 mm
Fuel length	564 m
Number of bundles	29
Bundle fuel pin number	25
Bundle dimensions	8.9 cm square cross-section
Control rods	
Material	Hot pressed compacts of B_4C aluminium clad, natural boron; square cross-section identical with fuel bundle
Number	8
Control rod drive	Electrical motor with rack and pinion Electromagnetic connection with the control rod
Reflectors	
Material	Beryllium; square cross-section identical with fuel bundle
Number	20 with 3.3 cm central hole 24 without hole
Flux levels	
Max. thermal flux	2.6×10^{14} cm$^{-2}\cdot$s^{-1}
Max. fast flux	2.0×10^{14} cm$^{-2}\cdot$s^{-1}
Pool	
Pool dimensions	10 m depth, 5 m width, 9 m length, aluminium tank, light water

Source: References [19, 20].
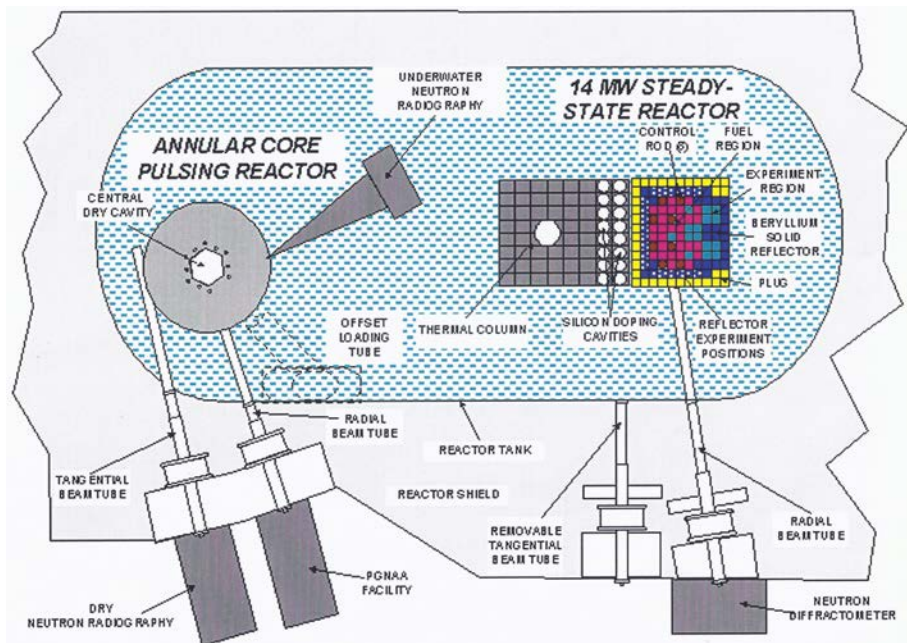
FIG. 22. Both the MPR-16 steady state reactor and ACPR in Piteşti, Romania, are operated in the same open pool (courtesy of Institute of Nuclear Research, Piteşti, Romania).

3.2. TRIGA FUEL DESIGNS

The TRIGA fuel element design evolved simultaneously and with some independence from the TRIGA reactor design. All TRIGA type fuels are inherently safe for reactivity insertion events by virtue of the $UZrH_x$ matrix; however, there are several TRIGA fuel types that differ in cladding, enrichment, weight percent uranium, size and burnable poisons.

Early fuels used type 1100F aluminium cladding, which was later discontinued as the thermal power of the reactors increased, in favour of type 304 stainless steel and Incoloy-800 clad designs. A FLIP fuel type developed and tested with variations in uranium enrichments (from 70–93 wt% ^{235}U) and weight percent uranium loadings and an erbium additive to extend core lifetime lead to the development, qualification and utilization of high density (20–45 wt% uranium) fuels using low enriched uranium (LEU) and allowed the elimination of high enriched uranium (HEU). More than 20 fuel element variations were also developed to perform specific functions in the core. Starting with a basic single rod fuel moderator element, the first variation added thermocouples within the metal fuel matrix to support the monitoring of fuel temperature during reactor

34

operation. Fuel follower control rods (FFCRs) are also available. Other fuel variations, such as fuel for ACPRs, TRIGA fuel rod cluster designs for converting reactors with plate type fuel clusters without changing the core grid and smaller diameter 1.37 cm fuel rod clusters for higher power (>5 MW) TRIGA reactors were designed and marketed for specific reactor designs dictated by user needs.

Each fuel element has its own, unique and discreet serial number assigned by the manufacturer, engraved permanently on the top end fitting that are traceable to its origins during the fuel fabrication process. Single rods for cluster assembly have their serial number engraved on the bottom fitting. Graphite dummy rods that are used for reactivity shimming in TRIGA reactor cores also carry unique serial numbers assigned by the manufacturer.

3.2.1. The basic TRIGA fuel moderator elements

The $UZrH_x$ fuel rod is a cylinder, with the fuelled section approximately axially centred in the cladding tube, with a graphite moderator slug at each end. The fuel elements contain uranium with various loadings as a fine metallic dispersion in a ZrH matrix. The H:Zr ratio is nominally 1.6, although the very early fuels previous to 1960 used a H:Zr ratio of 1.0. The equilibrium hydrogen dissociation pressure is governed by the fuel composition and temperature. For $ZrH_{1.6}$, the equilibrium hydrogen pressure is 1 atmosphere at about 760°C. The single phase, high hydride composition eliminates the problems of density changes associated with phase changes and with thermal diffusion of the hydrogen. Several 10 000 pulses (transients) have been performed with various $UZrH_x$ fuels manufactured by General Atomics. The ultimate safety limit for TRIGA fuel is 1150°C due to potential stress failure of the stainless steel cladding and to dissociated hydrogen pressure. In 1969, during a 5.50$ reactivity insertion event, the General Atomics Mark F reactor reached a power level of 6400 MW, and the maximum fuel temperature was approximately 660°C. The ultimate safety limit for TRIGA fuelled reactors is the fuel temperature. This limit stems from the outgassing of hydrogen from $UZrH_x$ fuel and the subsequent stress produced in the fuel element cladding material. The strength of the cladding as a function of temperature therefore sets the upper limit on the fuel temperature. This safety limit is about 1150°C when the cladding temperature is below 500°C for $UZrH_{1.65}$ and precludes the loss of cladding integrity. When the cladding temperature equals the fuel temperature, the fuel temperature limit is about 950°C. There is also a long term operational fuel temperature design criterion of 750°C based on consideration of irradiation and fission product induced fuel growth and deformation. This time a temperature dependent fuel growth occurs over the burnup life of the fuel. A maximum temperature of 750°C has therefore been used as the design basis operating temperature because lower average fuel

temperatures result only in insignificant calculated fuel growth from temperature dependent irradiation effects.

The early fuels had samarium burnable poison discs placed between the fuel rod and the graphite to minimize reactivity changes from [149]Sm, fission product build up and [235]U burnup (see Fig. 23). As reactor power was increased, later fuels abandoned the use of such a burnable poison disc in favour of the use of erbium to minimize reactivity changes from [235]U burnup in the high density fuels. This also enhances the prompt negative temperature coefficient due to the [167]Er resonance peak and is mixed homogeneously in the fuel moderator UZrH material. All fuels also now include a molybdenum disc, which is now placed between the lower graphite reflector piece and the fuel meat to prevent bonding at higher temperatures between the fuel meat and the lower graphite reflector. There is no bonding material between the cladding and the fuel rod. While the original core and the early (aluminium clad) fuel elements had a single 35.56 cm $UZrH_x$ fuelled meat, this design evolved to a 38.1 cm fuelled section in three equal lengths. Furthermore, the $UZrH_x$ fuel meats now also have a central 0.64 cm cavity (for improved hydriding), with a solid zirconium rod inserted in this cavity upon element assembly.

Fixtures are heliarc welded to the top and bottom ends of the cladding tube, encapsulating all of the internal pieces. The bottom end fixtures serve to guide the fuel elements into the bottom support plate of the reactor core. The top end fixtures provide an end surface that the coupling end of a handling tool can grip. Both fixtures are designed to facilitate the flow of cooling water around the fuel element. Typical end fittings are illustrated in Fig. 23. Modern streamlined end fittings were introduced in the 1970s to eliminate problems with 'chugging' (the movement of elements associated with water turbulence phenomena in the core) in natural convection cooled reactors at power levels approaching 2 MW(th) owing to hot cooling water flow around the element in the reactor creating voids at the fuel element surface.

3.2.2. Instrumented TRIGA elements

Each reactor typically has one or more thermocouple instrumented elements in the core to monitor fuel temperature — and scram the reactor if so configured — during operation. Such an instrumented element (see Fig. 23) is similar to the standard fuel moderator elements for that core, but has three chromel–alumel thermocouples inserted partway into the bottom, middle and top fuelled sections of the $UZrH_x$ fuel element. The top fitting of the instrumented element is adapted for a thermocouple lead out tube. The thermocouples pass through the lead out tube integral to the top end fixture and through additional watertight tubing (typically welded or soft soldered at the juncture) to above the water surface. This
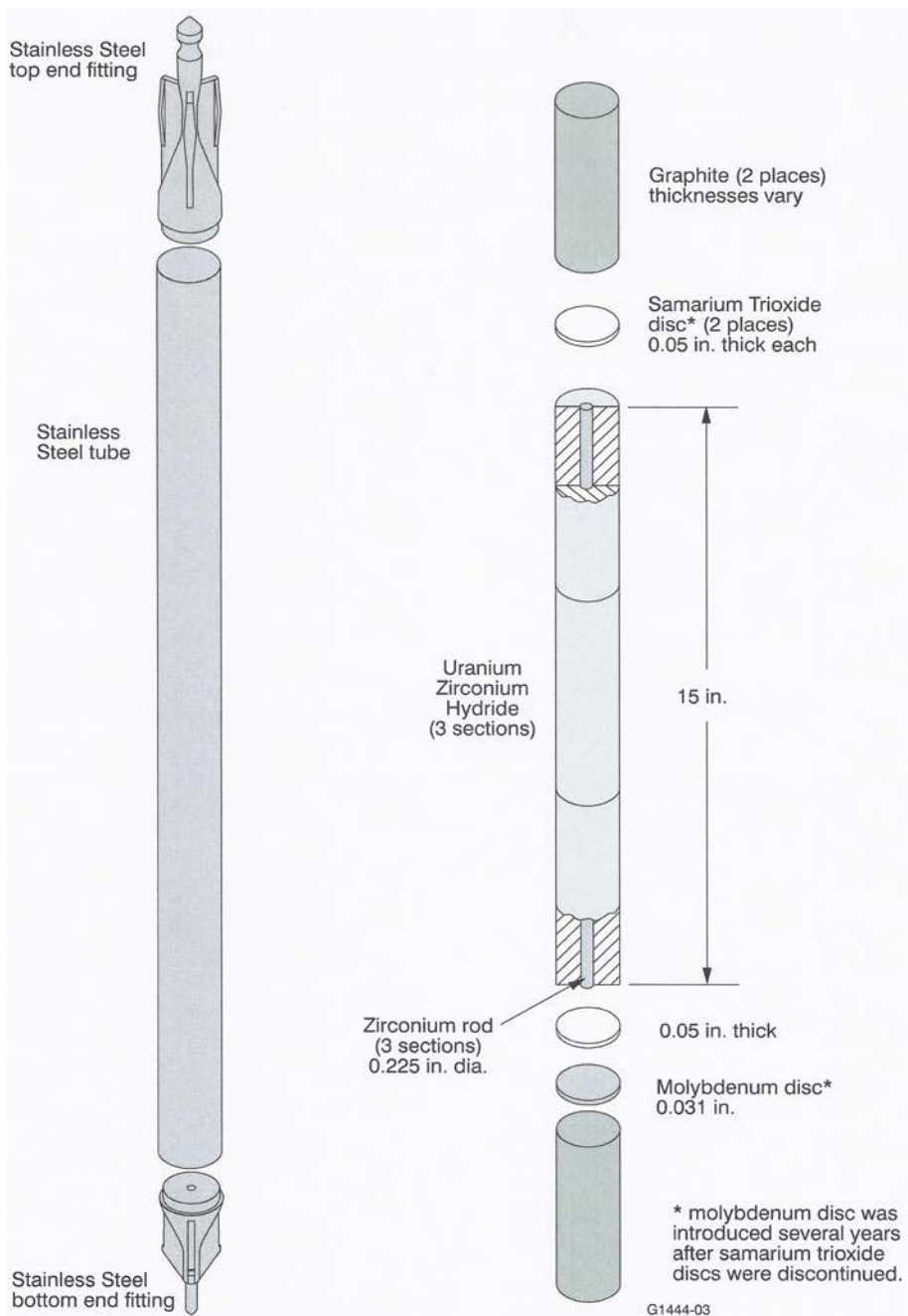
FIG. 23. Contents of a typical stainless steel clad fuel element (courtesy of General Atomics).

additional tubing is removed prior to shipment or storage of the spent fuel. A stub end of the lead out tube could affect the end of life element length, depending on the manner of the tube's removal.

3.2.3.   Fuel follower control rods

The cylindrical control rods with neutron poison material that are used in TRIGA type reactors are inserted in various core positions in lieu of a cylindrical fuel element for achieving reactivity control. When withdrawn from the core, control rods with follower sections that extend below the bottom of the core when the control rod is inserted can have solid or hollow aluminium air follower sections or sections with $UZrH_x$ fuel meats to increase the reactivity worth of the core by inserting a fuelled section into the core active region when the poison section of the core is withdrawn. TRIGA reactors that operate at thermal powers of 1 MW or higher are typically equipped with four to six FFCR elements, depending upon the reactor size and features. Reactors lower than a 1 MW power level could also use FFCRs, anywhere from zero to four FFCR elements, depending on the design. The same FFCR design can be used with all control rods (i.e. safety, shim and regulating rods).

FFCR elements have four internal sections: two air voids (one at the top and one at the bottom), a B_4C neutron absorber (top middle) and a $UZrH_x$ fuelled section (bottom middle), which has a standard 38.1 cm long $UZrH_x$ fuelled section, identical to that in the 'standard' fuel elements. FFCR elements have been fabricated for both 'standard' and ACPR TRIGA reactors — note that the FFCR elements for ACPR reactors are substantially longer than the others. The 'standard' FFCR elements have been fabricated with any of the uranium loadings typical of the stainless steel clad TRIGA elements (8.5 wt% uranium, 19.7% enriched in [235]U; and FLIP-HEU, or any of the high density 20% uranium — 45% uranium loadings). The ACPR FFCR elements have a slightly shorter fuelled section (12 wt% uranium, 19.7% enriched in [235]U) but a much longer bottom air void.

3.2.4.   Fuel design for TRIGA conversion reactors

Conversion reactors are reactors that originally used thin plate fuels with a particle dispersion fuel meat matrix (known as MTR fuel), but were later converted to use the inherently safe design for reactivity insertion events of the TRIGA $UZrH_x$ fuel moderator material. In most cases, the converted reactors have retained their existing core grid structure and in some cases their existing

reactor control systems. Plate type reactors usually have a rectangular core configuration and long, thin, rectangular plates that were stacked together inside a larger fuel bundle. Gaps between the plate faces supplied a maximum surface area for the cooling of the plates within the element bundles.

To accommodate the physical geometry required to convert a plate type fuel bundle, General Atomics created fuel cluster assemblies with the same outside dimensions as required for the reactor's core grid, but replaced the plate fuel with four $UZrH_x$ rods within the new assembly. These four rod bundles were then added a few assemblies at a time to the existing plate type core until the core was entirely comprised of $UZrH_x$ TRIGA fuel moderator assemblies.

The resulting TRIGA conversion reactors are capable of steady state power levels of 1–3 MW as well as pulsed power operation. (Conversion reactors may be capable of pulsed power operation, but their licences have not generally been revised to accommodate this function.) Thirteen former plate type reactors have been converted to $UZrH_x$ TRIGA fuel moderator rods (seven US domestic reactors and six foreign reactors). Also, due to the advantage in core life expectancy, many conversion reactors generally used the high enriched TRIGA FLIP rods in their assemblies for initial criticality rather than rods with the original 8.5 wt% or 12 wt% fuel loading. All conversion reactors originally converted with HEU fuel have now been converted from high to low enriched fuel using the high density fuels $UZrH_x$–Er (uranium > 20 wt%). More detailed information about TRIGA conversion is provided in Section 5.

3.2.5. Cluster assemblies

Conversion cluster assemblies were used to replace MTR plate type fuel assemblies with clusters of (three or) four TRIGA rods. The rods in the TRIGA conversion cluster assemblies are slightly smaller in diameter than the basic TRIGA fuel moderator elements and assembly end fittings are compatible with the MTR type reactors, as shown in Fig. 24. The bottom end fitting of the cluster assembly can have either a round or square cross-section. Most conversion clusters hold four rods, but a three rod variation offers a sample irradiation tube in place of a fourth rod.

TRIGA reactor designs have now been extended from the early low to medium power levels, typically less than or equal to 1 MW(th), to operate at power levels in the tens of megawatts. This required the development and qualification of a new generation of TRIGA fuel assemblies for such higher power (≥ 5 MW(th)) TRIGA type research reactors. These fuel assemblies use fuel rods with the same $UZrH_x$ fuel meat characteristics providing the same
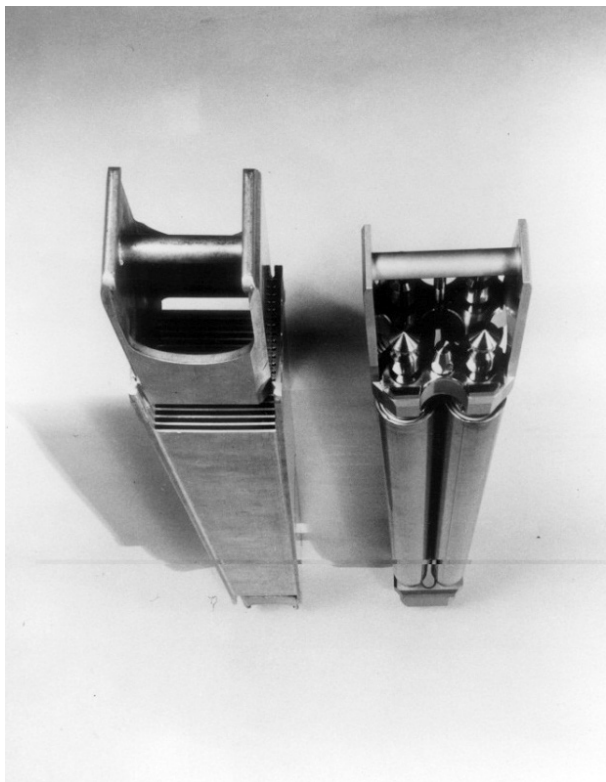
FIG. 24. MTR to TRIGA conversion fuel assembly (courtesy of General Atomics).

inherent safety characteristics for reactivity insertion events, but are of a much smaller diameter of 1.27 cm and use Incoloy-800 for cladding instead of 304 stainless steel. They are assembled in square clusters of 16 ($\leq$ 10 MW(th)) or 25 (10–20 MW(th)) fuel rods, with separator grids provided in the assembly shroud to ensure proper separation and coolant flow. There are four fuel meats in these smaller diameter rods for a total fuelled length of 55.88 cm, and they do not have the zirconium rod in the middle. These Incoloy clad rods also have springs instead of the graphite moderator pieces above and below the fuel within the cladding to allow for radiation induced axial expansion of the fuel meats.

Table 5 summarizes some of the general specifications for the $UZrH_x$–Er fuels that comprise most of the fuel that has been used, or is currently being used, in TRIGA type reactors.

TABLE 5. SUMMARY OF NOMINAL SPECIFICATIONS FOR UZrH–ER TRIGA LEU FUELS

Specification	Comment
Uranium content (wt%)	8.5, 12, 20, 30 or 45
Uranium enrichment (wt%)	19.7 ± 0.2
Uranium homogeneity	Manufacturing process produces a homogeneous distribution of uranium
Burnable poison	Erbium: Nominal range is 0.5–1.6 wt%, depending on uranium content, core reactivity and burnup requirements The low uranium density fuels (i.e. fuel with 8.5 wt% and 12 wt%) has no erbium
Erbium homogeneity	Manufacturing process produces a homogeneous distribution of erbium
H/Zr ratio	Nominal value is 1.60 (range 1.57–1.65)
Cladding material	304 stainless steel or Incoloy-800 with a $B_{eq} < 8$
Cladding thickness	Nominal element OD (cm): 3.81, 1.27 Nominal thickness (mm): 0.51 0.41

Note: OD — outer diameter.

3.3. SAFETY ASPECTS OF TRIGA REACTORS

Besides being versatile and having a wide scope of application, TRIGA reactors have unique user friendly safety features (i.e. a large negative temperature coefficient of reactivity (10^{-4} $\Delta k/k/°C$) and void coefficient due either to the under moderation of the core or in some cases) to the increase of neutron temperature by the crystalline effect of the moderator used (ZrH). This negative temperature coefficient limits the power excursion during a sudden insertion of reactivity. Since this coefficient is prompt due to the intimate mixing of the fuel (uranium) and moderator (zirconium, hydrogen), a large insertion of reactivity (i.e. 3$) can pulse the power of the reactor to a very high power level up to 2000 MW, while

the self-limiting property of the prompt negative temperature coefficient quickly (i.e. 40 ms) brings down the power surge. Pulse operation of a research reactor allows a number of special experiments such as activation of short lived nuclides (i.e. ^{28}Al, ^{20}F, ^{8}Li, ^{16}N and ^{19}O) with half-lives in the second and minute range and radiation pulses on organic, medical and biological materials can be studied.

In summary, it can be stated that in nearly 60 years of operations with TRIGA research reactors, no serious incidents have occurred in any of the TRIGA research reactors that have been installed worldwide.

3.4. ADVANTAGES AND DISADVANTAGES OF TRIGA REACTORS

3.4.1. Advantages of TRIGA reactors

The attractive features of TRIGA type research reactors include:

(a) Low capital cost: The exact cost information for operating reactors is not relevant today, since most of them were installed a long time ago. Best estimates for the less than 30 kW reactors range from around US $400 000 for the simplest reactor to about US $1–1.5 million at the time they were installed. This compares to a price of about US $5 million for a different reactor type of several hundred kilowatts of power. The lower end of the price range depends on the amount of local design and construction performed. Selection of additional options such as experimental facilities and instrumentation, among others, could significantly influence the cost.

(b) Low operating costs: The annual operating cost of TRIGA reactors is represented in the range of hundreds of thousands of US dollars. However, because of differences in economic conditions, it may be more reasonable to express this in terms of personnel requirements. This usually includes, as a minimum, one operator and one supervisor and other services such as a maintenance group, which can usually be shared with other laboratories, and repair, maintenance and radiological protection services, already available at the institute in which the reactor is located. Experience shows that these reactors are very reliable and relatively free of problems. The failure rate of various reactor components (pumps, filters and motors) is typically less than one per year and can be kept very low with a proper maintenance programme. The annual cost for consumable supplies, such as filters, irradiation capsules and protective clothing, is estimated at US $10 000–15 000.

(c) Low burnup: Because of the low operating power and large core lifetimes of the UZrH$_x$ fuel type, the fuel burnup is extremely small and most reactors

in this class have close to 'lifetime' cores. For example, a 250 kW TRIGA reactor with about 30 hours per week operation would run about 10 years without refuelling, and can be extended further with core management practices such as fuel shuffling. Many low power reactors have lifetime cores. The low burnup and limited radioisotope production capability result in minimal radioactive waste issues.

(d) Simple to operate: This class of reactors of simple design is easy to operate and requires a relatively short period of operator training. Higher power research reactors, designed for operator training, are more complex and require correspondingly more detailed training.

(e) Safety: TRIGA type reactors are inherently safe for reactivity insertion events due to the large, prompt negative temperature coefficient of the $UZrH_x$ fuel. Low power (<1.5 MW) TRIGA reactors are also able to withstand a loss of coolant accident owing to the low power density and the high fuel melting point. To mitigate the effects of a loss of coolant in the higher power reactors, they are equipped with a simple emergency core cooling design. Occupational radiation doses are relatively low. As with all reactors, however, any fuel manipulations needs to be carefully supervised.

(f) Less restrictive containment and siting requirements: Because of the low fission product inventory, the demonstrated fission product retention characteristics of the $UZrH_x$ fuel matrix, inherent safety in reactivity transient events and in loss of coolant accidents, the siting and containment requirements for this class of reactors is less restrictive. Most of the reactor facilities discussed above were housed in normal buildings (with controlled access) and in populated areas. Even the higher power reactors are housed in confinement buildings, not containment structures. However, the matter of siting and confinement or containment needs to be left to the national authorities where the reactors will be located.

(g) Versatility: These reactors are very well suited for training activities because of the flexibility of operation (startup time around 3 min). Such a flexibility to start up, shutdown or change operating power is usually not found in larger reactors.

3.4.2. Disadvantages of TRIGA reactors

TRIGA reactors, while providing significant flexibility in operations because of their inherent safety, have some limitations, primarily owing to their low power. These include:

(a) Limited production of radioisotopes: Many of the early TRIGA reactors were built as low power (100–250 kW) reactors built primarily for education

and training, and where the production of large quantities of radioisotopes is limited to those with short half-lives, for use in calibration or radiotracer applications, teaching or academic research, and in any case for local uses only. However, the production quantities can be increased by the use of in-core irradiation locations where the neutron flux is higher. While routine production of commercial quantities of radioisotopes such as ^{99}Mo requires higher power TRIGA reactors, typically 5 MW or above, limited quantities also can be produced in medium power (1–2 MW) reactors.

(b) Limited use of neutron beams: Performance of neutron beam research such as neutron scattering experiments can be limited because of the lower leakage flux available to the beam tubes from the very low power reactors (<100–250 kW). This is also true for fast neutron studies. However, TRIGA reactors have been efficiently used to perform beam tube experiments which could then be transferred and installed at high power research reactors. Examples of successful use of neutron beams at low power TRIGA reactors are at the Atominstitut, in Vienna, Austria (250 kW), and the Johannes Gutenberg University Mainz, Germany (100 kW). A typical example was the development of a neutron interferometer at the TRIGA reactor in Vienna in 1974, which was then transferred to the Grenoble high flux reactor for full scale demonstration and academic exploitation. Similarly, other lower power TRIGA reactors, such as the Johannes Gutenberg University Mainz, have effectively utilized their neutron beams for performance of a variety of beam tube experiments thanks to the installation of an ultra cold neutron source (more details can be found in the paper by Eberhardt and Geppert on the CD-ROM accompanying this publication).

(c) Limited use for materials and fuels testing applications: It is clear that the majority of TRIGA reactors, due to their low power, are not well suited for many of the material irradiation and fuel testing experiments where neutron fluxes greater than 10^{14} cm$^{-2} \cdot$s^{-1} are typically required. Even so, facilities have designed unique ways to conduct limited testing of fuels by utilizing the ability to perform such irradiations in-core. An example is the TRIGA High Temperature In-Core Furnace at the 250 kW TRIGA operated by General Atomics, where testing could be performed on reactor fuel samples to characterize fission product release behaviour at temperatures as high as 1800°C. Low power TRIGA reactors have also been extensively used to test various nuclear instruments or to examine the hardness neutron irradiation of various electronic components.

4. AREAS OF APPLICATIONS

The IAEA Research Reactor Data Base (RRDB)[4] includes information on 247 operating research reactors ranging in power from zero to several hundred MW(th). Among those operating research reactors, there are 38 TRIGA reactors operating with licensed powers in the range of 100 kW to 16 MW. Five operate in the power range of more than 1 MW to under 5 MW, and power levels between 250 kW and 1 MW are the most common.

In early 2015, the TRIGA research reactors worldwide numbered:

— Total constructed: 69 in 23 countries;
— In operation: 38;
— Shutdown or decommissioned: 31.

TRIGA reactors are versatile, multipurpose facilities covering diverse areas of utilization and applications with an original geographical worldwide distribution in 23 countries (see Fig. 25 and Table 6) [23].

FIG. 25. World map of TRIGA reactors (courtesy of General Atomics).

TABLE 6. LIST OF OPERATIONAL TRIGA REACTORS

No.	Country	Facility name	TRIGA type	Power		First criticality
				Steady state (kW)	Pulsing (MW)	
1	Austria	TRIGA II Vienna	Mark II	250	250	1962-03-07
2	Bangladesh	TRIGA Mark II	Mark II	3 000	3 900	1986-09-14
3	Brazil	IPR-RI	Mark I	100	n.a.[a]	1960-11-06
4	China[b]	Thor	Conversion	2 000	n.a.[a]	1961-04-13
5	Colombia	IAN-R1	Conversion	30	n.a.[a]	1997-08-29
6	Finland	FIR-1	Mark II	250	250	1962-03-27
7	Germany	FRMZ	Mark II	100	250	1965-08-03
8	Indonesia	TRIGA Mark II, Bandung	Mark II	2 000	n.a.[a]	1964-10-19
9	Indonesia	Kartini-PTAPB	Mark II	100	n.a.[a]	1979-01-25
10	Italy	Lena, TRIGA II Pavia	Mark II	250	250	1965-11-15
11	Italy	TRIGA RC-1	Mark II	1 000	n.a.[a]	1960-06-11
12	Japan	NSRR	ACPR	300	22 000	1975-06-30
13	Morocco	MA-R1	Mark II	2 000	n.a.[a]	2007-05-02
14	Mexico	TRIGA Mark III	Mark III	1 000	2 000	1968-11-08
15	Malaysia	TRIGA Puspati (RTP)	Mark II	1 000	1 200	1982-06-28
16	Romania	TRIGA II Piteşti – SS Core	MPR	14 000	n.a.[a]	1980-02-02
17	Romania	TRIGA II Piteşti – Pulsed	ACPR	500	22 000	1980-02-02
18	Slovenia	TRIGA Mark II Ljubljana	Mark II	250	n.a.[a]	1966-05-31
19	Thailand	TRR-1/M1	Conversion	2 000	1 200	1977-11-07
20	Turkey	ITU-TRR, Tech. Univ.	Mark II	250	250	1979-03-11
21	USA	MUTR Univ. Maryland	Conversion	250	n.a.[a]	1974-12-01

TABLE 6. LIST OF OPERATIONAL TRIGA REACTORS (cont.)

No.	Country	Facility name	TRIGA type	Power Steady state (kW)	Pulsing (MW)	First criticality
22	USA	OSTR, Oregon State Univ.	Mark II	1 100	3 200	1967-03-08
23	USA	UC Davis/ McClellan N. Research Center	Mark II	2 000	1 200	1990-01-20
24	USA	TRIGA II Univ. Texas	Mark II	1 100	1 600	1992-03-12
25	USA	TRIGA Univ. Utah	Mark I	100	n.a.[a]	1975-10-25
26	USA	NRAD	Conversion	250	n.a.[a]	1977-10-12
27	USA	Kansas State TRIGA Mark II	Mark II	250	250	1962-10-16
28	USA	GSTR USGS Geological Survey	Mark I	1 000	1 200	1969-02-26
29	USA	Univ. California, Irvine	Mark I	250	250	1969-11-25
30	USA	AFRRI TRIGA	Mark F	1 000	3 300	1962-01-01
31	USA	Dow TRIGA	Mark I	300	n.a.[a]	1967-07-06
32	USA	RRF Reed College	Mark I	250	n.a.[a]	1968-07-02
33	USA	UWNR Univ. Wisconsin	Conversion	1 000	2 000	1967-03-26
34	USA	WSUR Washington St. Univ.	Conversion	1 000	2 000	1967-03-13
35	USA	PSBR Penn St. Univ.	Conversion	1 000	2 000	1965-08-15
36	USA	Ann. Core Res. Reactor (ACRR)	ACPR	4 000	22 000	1967-06-01
37	USA	Aerotest Operations	Conversion	250	n.a.[a]	1965-07-09
38	USA	NSCR Texas A&M Univ.	Conversion	1 000	2 000	1968-01-01

[a] n.a.: not applicable.
[b] In Taiwan, China.

These reactors all operate with UZrH homogeneous solid fuel and following the HEU to LEU conversion programme, all now operate with an enrichment of less than 20%. All are light water cooled and use graphite or water as a reflector. Many TRIGA reactors are capable of operating also in a pulse mode (see Section 3) and most have a rotary specimen rack (called a Lazy Susan) containing 40 rotating irradiation positions between the core and the reflector. Most of them have neutron beam channels, so experiments can be designed outside the reactor shielding (see Fig. 26).

Compared to higher power research reactors, TRIGA reactors with a power level below 2 MW usually have neutron flux densities around 10^{13} cm^{-2}·s^{-1} in the central irradiation position. They offer a number of distinct advantages to a variety of users and for these reasons would be generally described as user
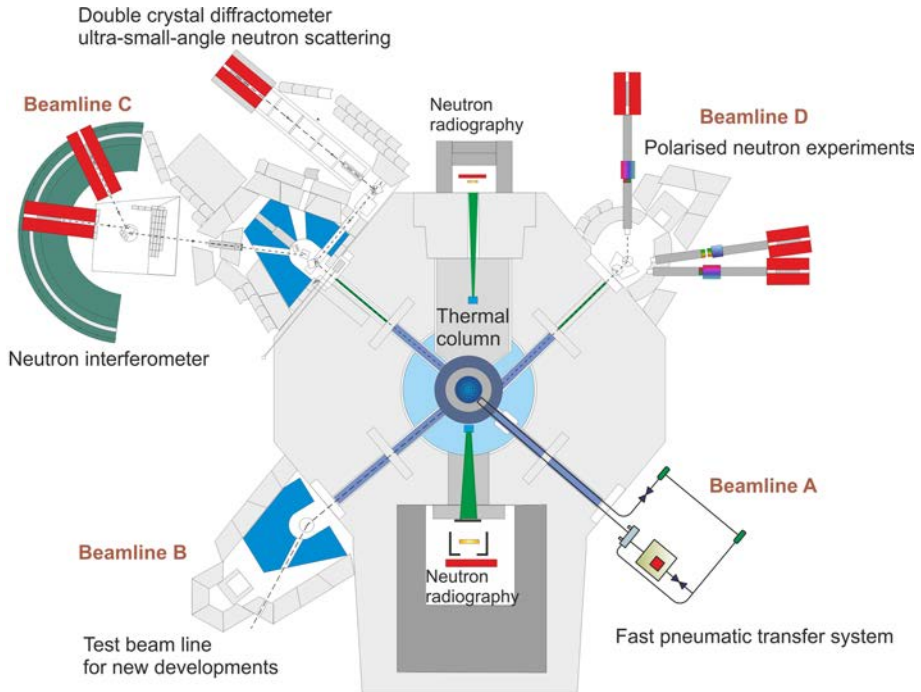
FIG. 26. Example of diverse experimental setups at the TRIGA Mark II reactor (horizontal cross-section) in Vienna, Austria (courtesy of Atominstitut, Vienna).

friendly (see also Section 3.4.1). Among these attractive features of TRIGA reactors, the following warrant a mention:

(a)   High degree of passive safety: The reactor will shut down on its own owing to a large, prompt negative temperature coefficient, and this is of paramount importance for training related applications.
(b)   Ease of operation: A minimum number of restrictions are imposed on the students and instructors (e.g. the control console can be operated safely by inexperienced students after a short instruction time under the supervision of a licensed operator).
(c)   Ease of maintenance: Equipment is arranged to provide easy access for maintenance, and components are selected for life and minimum maintenance.
(d)   Ease of experiments for students and instructor: A wide variety of training and research experiments for students can be performed [24].

Owing to the fact that flux levels at the sample irradiation locations are low and if the irradiation times are shorter, the typical irradiated samples are not highly radioactive, which reduces handling restrictions and associated administrative procedures. Therefore, the actual insertion of samples for irradiation in a reactor core positions can either be done by reactor staff or even by the experimenters themselves. The experience with irradiations in TRIGA reactor shows that there also a great variety of sample types and containments which can be employed: simple, inexpensive, plastic irradiation capsules can be used and re-used many times; organic matter can be safely irradiated, including liquids, even highly volatile, combustible or explosive organic or petroleum nature, wet tissues, excreta from humans or animals (without the need for drying or tedious sample preparation). Finally, quite large material quantities can be irradiated in the reactor without producing an excessive amount of radioactivity. Generally, this will be the case when fluxes of the order of 10^{12} cm$^{-2} \cdot$s^{-1} and irradiation times of less than one hour are applied.

4.1. EDUCATION AND TRAINING

Education and training is one of the most important uses of low power research reactors and TRIGA reactors are no exception. Generally, a reactor is rarely chosen (e.g. power, type of design and fuel) for a single application (e.g. physics, radioisotope production and training), but it appears that practically all low power research reactors in operation are used, at least partially, for training purposes. This interest in training activities concerns not only the States with an

active national nuclear programme. Many other States run education and training programmes with an important contribution from their research reactor facilities.

Several types of training are possible, depending on the type of trainees (students, physicists, operating personnel and radioprotection personnel), on the degree of information to be transmitted (theory and practice), on the length of the training courses (from a few days to several months) and, of course, on the characteristics and potential of the available research reactor (e.g. its power, installation characteristics, safety regulations and accessibility).

4.1.1. Training programmes

It is difficult to give an outline of the training programmes run by TRIGA reactors without taking into account the trainees involved or mentioning the possibilities and limitations offered by each type of reactor [25, 26]. Therefore, the following classification can be distinguished together with an associated list of training capabilities:

(a) Training of physics students:
— Nuclear radiation measurements such as activity, dose, half-life, energy, reaction with materials and activation analysis;
— Reactor theory in neutron transport by using spectrometers, neutron choppers, foil activation dosimeters;
— Reactor kinetics and reactor dynamics;
— Reactor operation and control, also by using associated computers;
— Criticality and power increase of the reactor;
— Relative and absolute flux measurements;
— Reactivity measurements;
— Control rod calibration;
— Temperature coefficient measurement;
— Poisoning effect measurement;
— Neutron spectrum measurement;
— Void coefficient determination;
— Radiation protection and shield measurement;
— Neutron radiography;
— Neutron activation analysis (NAA);
— Radioisotope determination.
(b) Training for non-physics students and researchers. TRIGA reactors can also be used for a whole series of related applications, involving the field of basic or applied research such as:
— Archaeometry;
— Biological applications;

— Chemical applications;
— Earth sciences;
— Environmental sciences;
— Medical applications;
— Metallurgy;
— Industrial applications;
— Forensic applications;
— Radiation detector.

(c) Training of reactor operating personnel. For this type of trainees, the emphasis is placed on procedures and operations involved in reactor operation and safety, in particular:
— Fuel loading and unloading;
— Approach to criticality;
— Effects of prompt and delayed neutrons;
— Poisoning effects (samarium and xenon);
— Temperature effects;
— Reactivity effects;
— Load variations;
— Instrumentation and calibration;
— Flux and power measurements;
— Reactor kinetics and dynamics;
— Radiation protection;
— Radiochemistry.

4.1.2. Other training equipment associated with TRIGA reactors

Among the numerous training course experiments using TRIGA reactors, the following additional possibilities are available.

4.1.2.1. Simulators

The use of reactor simulators has become progressively important. This equipment is flexible, adaptable, simple to implement, relatively cheap and provides training means which are greatly appreciated in the nuclear power field for training both nuclear engineering students and future power plant personnel. There is a large amount of literature in this field dealing with the different types of simulator and their main assignments (e.g. basic principle simulators, full scope representation, accident and function simulators). However, no full scope

TRIGA simulators have been developed, except the one available on the IAEA web site[5].

4.1.2.2. Computer assisted teaching

This is a more general topic as computer assisted teaching becomes dominant in all areas. Furthermore, in the nuclear field computer support is very important, as it may clarify many interconnections between reactor operation, temperature, reactivity and thermal-hydraulics. A number of simple teaching programmes are available on the market. Computer assisted courses have been developed by many international and national organizations as well as specific academic institutions. Such courses may be obtained from the IAEA free of charge[6] (see also Ref. [27]).

4.2. TRIGA REACTOR UTILIZATION FOR ACADEMIC PURPOSES

Many university owned TRIGA reactors are used not only for specific training but also for teaching and research purposes.

4.2.1. Teaching

At universities with a nuclear engineering programme, a reactor can be used to supplement theoretical courses with practical exercises. Typical courses could include:

— Reactor theory;
— Reactor dynamics;
— Reactor physics;
— Reactor I&C;
— Radiation measurements;
— Radioisotope techniques.

4.2.1.1. Reactor theory

In reactor theory courses, the topics cover the interaction of neutrons with matter, neutron fields, criticality calculations by use of diffusion and

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[5] See https://www.iaea.org/NuclearPower/Simulators/index.html
[6] See http://clp4net.iaea.org/ and http://elearning.iaea.org/m2/

transport calculations, homogeneous and heterogeneous reactor calculations and perturbation theory. This includes thermal, epithermal and resonances flux measurements, cross-sections, void coefficient, diffusion length, Albedo, Fermi-age and crystal spectrometers.

4.2.1.2. Reactor dynamics

A typical course in reactor dynamics would cover the time behaviour of reactors. This includes: the kinetics of zero power reactors, the reactivity effects of temperature, burnup, breeding, poisoning as well as the computer simulation of power stations and their control. In the case of such courses, the reactor provides a criticality experiment, control rod calibration, reactivity temperature coefficient, fuel reactivity values, reactor period measurement, subcritical multiplication, inhour equation, fission and ionization chambers, and self-powered neutron detectors.

4.2.1.3. Reactor physics

The following course determines the most important reactor physics parameters and could be carried out both in theory (classroom instruction) followed by practical exercises at the reactor itself. The main topics are:

— Thermal and fast neutron flux measurements;
— Void and temperature coefficient;
— Control rod calibration;
— Power calibration;
— Approach to criticality;
— Reactor period;
— Fuel reactivity values.

4.2.1.4. Instrumentation and control

Such a course covers the design and requirements for a reactor's I&C system, including nuclear channels, fuel temperature channels, interlocks, alarm and scram systems, and neutron detectors. This would require a fully operational I&C system, operational reactor, operational neutron detectors, circuit graphs and block diagrams.

4.2.1.5. Radiation measurement courses

Such a course includes the atom model, radioactivity and radioisotopes, alpha, beta and gamma, as well as neutron radiation, interaction between radiation and matter, radiation detection and dosimetry, standard measurements like activity, lifetime, energy and dose, and statistical computation. The reactor serves as a source for thermal and fast neutrons, production of weak and short lived isotopes for measuring purposes, shielding experiments in experimental channels of thermal column, and measurement of the radiation field in the vicinity of the reactor.

4.2.1.6. Radioisotope or other technique courses

The application and use of radioactive isotopes in science and technology are dealt with in radioisotope technique courses. This includes production and measurements of radioactive isotopes, their application as traces, and radiographical measuring techniques of material properties, basics of radiography as well as NAA.

4.2.2. Academic research

There are mainly three research reactor based techniques in basic and applied research with growing significance:

— NAA;
— Non-destructive testing by neutrons;
— Production of short lived radioisotopes.

Practically all fields of natural sciences and technology, especially medicine, biology, chemistry, geology and metallurgy, can be served by these procedures.

4.2.2.1. Neutron activation analysis

The technique is used as a tool in several R&D fields and in materials characterization for industrial processes or as useful reference technique. NAA is a non-destructive, highly precise and accurate analytical technique capable of determining small quantities of up to 30 elements in almost all types of sample matrix. One of the advantages of this method, compared to standard chemical analysis methods, is the sample preparation for the analysis. Namely, it consist only in weighting the sample but not performing its dissolution or other chemical

treatment, which can sometimes lead to sample contamination. When the samples are analysed as they are and without special preparations, the technique is called instrumental neutron activation analysis (INAA). In some cases, some samples might require radiochemical separations after irradiations (radiochemical neutron activation analysis, RNAA). Typically, samples are investigated by irradiating them for short periods in a reactor, from tens of seconds to tens of minutes (sometimes repeatedly) and then by measuring the induced activity [28]. In this way, some of the material is transformed into radioactive isotopes. The samples are investigated by gamma spectrometry after being removed from the reactor, in view of gamma energy, intensity and, in certain circumstances, also decay constants. Out of these results, a computerized analysing system can determine quantitatively the composition of the sample.

The reactor serves as a source of neutrons for irradiating the samples. In addition, the technique requires a sample preparation laboratory, a high quality multichannel analyser with semiconductor detectors and a computer with specific software.

4.2.2.2. Non-destructive testing by neutrons

This diagnostic technique is used to improve the knowledge of materials structures and defects in various objects. Neutron radiography is a modern investigation procedure similar to X ray analysis but using neutrons as a probe. Presently modern radiography uses charged-coupled device cameras and therefore a digital image is obtained for further analysis and interpretation. Present applications are corrosion studies in aluminium (aircraft industry), behaviour of moisture in building materials, distribution of hydrogen in various objects or devices, performance of lithium batteries, oil or lubricants in mechanical equipment, and nuclear fuel behaviour research, among other things [29, 30]. This method can also be extended to the computed tomography with neutron beams in order to obtain 3-D information about spatial location from 2-D projections. Again, the reactor serves as a source of collimated neutrons, extracted through neutron beam channels.

4.2.2.3. Production of short lived radioactive isotopes

Radioisotopes are a powerful tool in research. TRIGA reactors, owing to their limited power, can produce only isotopes with short half-lives of less than a few days. Such radioisotopes are commonly used for tracer research at universities and hospital laboratories.

Short lived radionuclides such as ^{82}Br, ^{18}F, ^{116m}In, ^{128}I, ^{56}Mn, ^{42}K and ^{24}Na are the radionuclides which generally are not possible to be shipped owing to

their fast decay. However, their use in demonstration experiments, teaching and research remains extremely important. The use of short lived radionuclides produced by TRIGA research reactors is particularly suitable for teaching purposes because of their fast decay. The preparation of short lived radioactive sources for a variety of applications in radiation analysis, instrumentation development, calibration and use as radiotracers is possible thanks to their local production [31].

4.2.3. Other uses of TRIGA reactors

Applications of TRIGA research reactors are varied and cover most of the products and services research reactors can possibly provide [31, 32]. Some selected examples of using extracted neutron beams are explained in more detail in the following. Cold, thermal or epithermal neutron beams extracted through reactor beam tubes can efficiently be used to study:

— Neutron scattering (small angle neutron scattering, diffraction, residual stress measurements, neutron interferometry, neutron reflectometry and neutron spin echo) [33];
— Neutron radiography (or neutron imaging) [34];
— Large sample NAA;
— Prompt gamma neutron activation analysis (PGNAA) [35];
— Boron neutron capture therapy (BNCT) [36–38].

Neutron scattering allows the studies of the structural and the dynamics of atomic arrangements in materials. Neutrons are scattered by atomic nuclei, are sensitive to the atomic magnetic moment, and have scattering and absorption cross-sections independent of atomic number and mass. The neutron techniques have a complementary role to X rays, scattered by the electrons in atoms. The main applications are related to studies of transformations and fundamental properties of materials. Neutron scattering studies are applied to earth, energy and environmental sciences.

The sensitivity of light elements to the absorption of neutrons compared with X ray imaging provides neutron imaging an advantage in both 2-D and 3-D visualizations. During the last two decades, digital neutron imaging and data processing systems have been introduced. As a result of these innovations, neutron imaging applications have been extended to the determination of hydrogen in electrochemistry analyses of fuel cells, non-destructive studies of cultural heritage objects and biological samples, applications in geology and soil physics, applications with nuclear fuel and its cladding, and investigations of various aspects of materials research.

Large sample NAA is capable of direct analysis of samples with masses of hundreds of grams to several kilograms. Though the principles and physics of large sample activation analysis are thoroughly understood, the method is still not as versatile or applicable as, for example, normal small sample NAA. Examples of such applications can be found in studies on electronic waste, complete archaeological and cultural artefacts, high purity materials, and materials of irregular shape or homogeneity.

PGNAA is a very widely applicable technique for determining the presence and amount of many elements simultaneously in samples ranging in size from micrograms to many grams. PGNAA uses the prompt gamma rays emitted during the process of neutron capture. It is a non-destructive method, and the chemical form and shape of the sample are relatively unimportant. It is used to determine the elements of low Z (i.e. low numbers of protons in the nucleus) or with high neutron absorption such as boron, cadmium, carbon, chlorine, gadolinium, hydrogen, nitrogen, phosphorus and samarium.

When ^{10}B absorbs a neutron, it emits an alpha particle, which is highly ionizing and has a range in tissue about equal to the diameter of a cell. The applicable method in BNCT is thus to inject a tumour with a borated compound that irradiates it with thermal neutrons. The majority of neutron capture therapy research has focused on malignant melanomas and brain tumours, particularly glioblastoma multiform. For many years, the TRIGA reactor in Finland was used exclusively for the treatment of patients using the BNCT technique (see Fig. 27).

4.3. INDUSTRIAL APPLICATIONS OF TRIGA REACTORS

4.3.1. Specialized training

As summarized in Section 4.2, TRIGA reactors are an excellent tool for academic training. However, they can also be applied to commercial training for industrial staff (i.e. the training of nuclear power plant staff, regulatory staff, radiation protection staff and medical staff). The respective courses need to be adapted individually to the trainees' interests and requirements. These courses can then become an important commercial income of the TRIGA reactor facility, as is reported in Ref. [25].

4.3.2. Neutron radiography

TRIGA reactors equipped with beam tubes and capable of operating at a power greater than 100 kW may be used for neutron radiography (or neutron imaging). (See also Section 4.2.3 for the description of the technique.) Examples

FIG. 27. Entrance to the patients' radiation room at TRIGA facility in Finland (courtesy of VTT Espoo).

of applications are related to studies of different materials such as ceramics (e.g. defects), lubricating and oils, carbon fibre composites, characterization of building and shielding materials, and quantitative analysis of nuclear fuel pin or cladding structures. Specific applications regard studies on corrosion in mechanical structures, hydrogen fuel cell, boron carriers related to neutron therapy and fluid dynamics [32, 39]. Neutron radiography at TRIGA reactors has successfully been applied in industry (i.e. for early corrosion detection in aluminium, for bonding between metal and rubber, and water penetration in building materials). One particular TRIGA reactor, in Sacramento, United States

of America, was designed with four special neutron radiography rooms and successfully used to detect early corrosion of jet fighter wings [40].

4.3.3. Neutron activation analysis

Commercial application of NAA can be performed in the areas of the mineral and ceramic industry, the petrochemical industry, food, agriculture and the environment. In practice, in all cases of NAA the technique is used to detect trace elements or pollutants in the samples supplied by industry or medical centres. Another related application is in forensics, where traces of barium can easily be detected in criminal investigations [28, 41].

4.3.4. Radioisotope production

Radioisotopes find extensive applications in several fields, including medicine, industry, agriculture and research. Radioisotope production to service different sectors of economic significance constitutes an important ongoing activity of many national nuclear programmes. Radioisotopes require further processing in almost all cases to obtain them in a form suitable for use. Specifications for final products and testing procedures for ensuring quality are also an essential part of a radioisotope production programme.

4.3.5. Gemstone irradiation

Another commercial application which can be easily applied at TRIGA type reactors is gemstone coloration. The colour of crystals and semi-precious stones can be enhanced by epithermal and fast neutron irradiation; this could, however, lead to the activation of trace elements contained in the crystal. Therefore, thermal neutrons should be shielded to reduce undesired activation of stones. Careful radiation control has to be carried out before releasing the coloured gemstones.

5. HEU–LEU CONVERSION OF TRIGA REACTORS

5.1. CONVERSION OF US TRIGA REACTORS

This section primarily deals with the HEU to LEU conversion of TRIGA type reactors under the US Government's Global Threat Reduction Initiative (GTRI). Early in the TRIGA programme, however, General Atomics developed

UZrH$_x$ fuel assemblies to permit the conversion of existing plate type (MTR) HEU reactors to TRIGA type cores as described in Section 3. Some of these were HEU to HEU conversions — with later conversions to LEU fuel — or were originally done as HEU to LEU conversions (see Table 6, Section 4). Such conversions, when performed outside the United States of America, also involved the technical support of IAEA experts. One such conversion is briefly described here.

5.1.1. Role of the United States Nuclear Regulatory Commission

Code 10 CFR 50.64 of the United States Code of Federal Regulations requires non-power reactors to replace all HEU fuel (i.e. fuel with more than 20% enrichment) with LEU fuel (i.e. fuel with <20% enrichment) acceptable to the United States Nuclear Regulatory Commission (NRC), provided Federal Government funding is available for the HEU to LEU conversion [42]. The regulation also requires provision of supporting safety analysis to demonstrate that the converted reactor will operate safely and predictably.

Five TRIGA research reactors in the United States of America have undergone HEU to LEU conversion since 2004:

— Neutron Radiography Facility (NRAD), at Idaho National
  Laboratory (INL);
— Oregon State University;
— Texas A&M University;
— University of Wisconsin;
— Washington State University (WSU).

The INL TRIGA reactor falls under the regulatory oversight of the United States Department of Energy (DOE), and as such is not representative of the course of activities at NRC regulated research reactors. This publication focuses on the conversions that were carried out at the university research reactors in the United States of America. The conversion timeline and processes of conversion were similar at the five universities mentioned above. This narrative will use the sequence of events at WSU as a representative example.

A case history of the HEU to LEU conversion of the Washington State University research reactor (WSUR) is described as an example. The licence to convert was issued by the NRC on 4 September 2008 and was to become effective 20 days after the date of publication in the US Federal Register. The HEU fuelled core (Core 34A) was shut down on 19 September 2008 and core disassembly commenced. The first LEU fuel assembly was placed in the reactor

at 3:53 p.m. on 29 September 2008, thus accomplishing the conversion before the milestone date of 30 September 2008.

5.1.2. History and purpose of the conversion programme

The Reduced Enrichment for Research and Test Reactors (RERTR) programme was initiated by the DOE in 1978. The objective of the RERTR programme is to convert research and test reactors which use HEU fuel to LEU fuel. The RERTR programme is managed by INL of the DOE.

The WSUR was converted from MTR fuel to 8.5%, nominal 20% enriched (8.5/20) LEU TRIGA fuel in 1967 (see Fig. 28). A portion of the LEU fuel was replaced in 1976 with HEU fuel as part of the FLIP, and the reactor was operated with both FLIP and LEU fuels. The FLIP fuel had a nominal ^{235}U enrichment
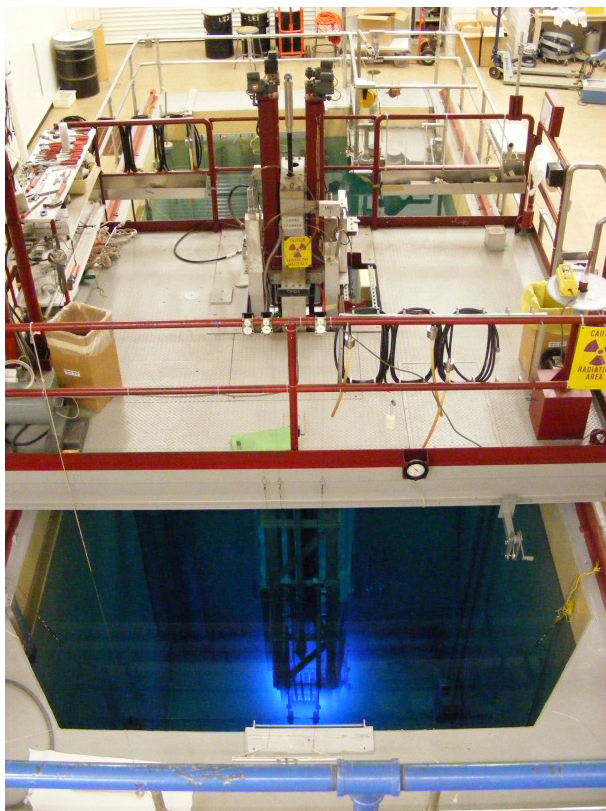
FIG. 28. Washington State TRIGA Conversion Reactor: Bridge and pool with its operational core (courtesy of Washington State University, United States of America).

of 70%. The reactor was operated until 2008 with various core configurations, rearranging the FLIP and LEU fuels, graphite reflectors and irradiation positions according to the research needs. The final HEU/LEU core configuration before conversion, referred to as Core 34A, consisted of 51 FLIP fuel rods in the central region of the core and 68 standard LEU (8.5/20) TRIGA fuel rods.

5.1.3. Principal parties in the conversion activities

In addition to WSU, the conversion was carried out with the cooperation of INL, Argonne National Laboratory, General Atomics, Brookhaven National Laboratory and the NRC. Fresh 30/20 fuel receipt and spent nuclear fuel shipping was done with the cooperation of INL, Secure Transportation Services and NAC International. The role of each organization is discussed separately.

5.1.4. Series of events

Conversion of the WSUR was carried out through a series of activities which included:

— Preparation of a safety analysis report (SAR);
— Generation of requests for additional information (RAI) by the NRC;
— WSU's responses to the RAI;
— Generation of a safety evaluation report (SER);
— Shipping fresh fuel from the fuel fabrication facility in France and receipt of the fuel at WSU;
— Generation of the HEU to LEU conversion order by the NRC;
— Defuelling and refuelling the reactor;
— Startup;
— Power calibration;
— Shutdown margin measurements.

5.1.5. Technical documentation

An SAR was submitted to the NRC in June 2002 as part of a request for facility licence renewal. The 2002 SAR described the reactor as it existed at that time, with a core containing both HEU and LEU fuels. The 2002 SAR was still under review in 2007 when the preparation began for the conversion of the reactor from the HEU/LEU mixed core to all LEU fuel. A supplemental SAR, primarily written by General Atomics, was prepared and submitted to the NRC in August 2007. The 2007 SAR presented analysis of the mixed HEU/LEU core

(Core 34A) along with predicted behaviour for the proposed all LEU fuelled reactor (Core 35A).

The core configuration, in a geometrical sense, was the same for Core 34A and the proposed Core 35A due to the planned substitution of the FLIP fuel with 30/20 fuel on a rod by rod basis. Consequently, the geometric component of the MCNP (Monte Carlo N-Particle) input deck was the same for both cores, and differed only in the compositions of the ZrH_x/U fuel matrix, with the uranium burnup, plutonium buildup, burnable poison (erbium) depletion and fission product buildup accounted for in the FLIP fuel model.

A MCNP model for the HEU/LEU core had not been developed before commencement of the conversion activities. Core analysis was done by General Atomics, using MCNP to model power distribution and the BLOOST code to model pulsing behaviour. Values of the prompt negative temperature coefficient, heat capacity, shutdown margin and core excess reactivity were also modelled by General Atomics. The model provided power distribution on a fuel assembly by fuel assembly basis. At full power, the reactor operates with the central control rod (referred to as the pulse rod or transient rod) fully withdrawn from the reactor core. The resulting core vacancy (i.e. a single rod water hole) leads to an increase in neutron thermalization in the vacancy and causes local power peaking in adjacent fuel rods. The thermal flux peaking together with the central core location of the rods combines to act as a limiting factor for placement of the temperature sensing instrumented fuel elements (IFEs).

Argonne National Laboratory was engaged to determine thermal-hydraulic characteristics and used RELAP to model expected fuel temperatures. The limiting safety system setting (LSSS) that is specified in the WSUR technical specifications stipulates that the fuel temperature scram set point be no higher than 500°C. The intent of the LSSS value is to protect the safety limit (i.e. the fuel temperature limit of 1150°C for 30/20 LEU fuel and 1050°C for 8.5/20 LEU fuel). The criteria that were established for determining allowable core positions for the IFE were that the IFE would reach the fuel temperature scram point of 500°C before onset of departure from nuclear boiling (DNB), yet would not reach the 500°C scram set point before the reactor reached full power. For example, placing an IFE in the rod position exhibiting the highest temperature due to the amount of power peaking in that position would result in the IFE centreline fuel temperature reaching 500°C at approximately 900 kW of reactor power (i.e. before reaching full licensed power of 1000 kW). Placing the IFE in the highest power core location or another nearby location with similar power factors would not compromise reactor safety. However, it would prevent operating the reactor at full power, and thus act as an unnecessary operationally limiting situation. On the other hand, there are core locations within the 30/20 region in the central part of the reactor core which would be inappropriate for an IFE. Placement of an

IFE in a core position with relatively low power, compared to the power level in the highest power fuel rod, would permit reactor power to increase beyond the licensed power level before reaching the 500°C scram set point. In addition, power peaking in the highest power fuel rod could conceivably approach an unacceptable temperature margin between operational coolant temperature and the temperature at which DNB commences.

Reactor pulsing for experimental purposes is carried out from time to time, making pulsing behaviour of considerable interest. All WSUR pulses going back to 1967 have been recorded in logbooks. The records include measurements of peak pulse power, energy release and peak fuel temperature in the IFE. The WSUR pulse behaviour was modelled by General Atomics using the BLOOST code, and was compared with historical pulsing data for Core 34A. It was found that BLOOST consistently over predicted peak power but was in reasonable agreement with observed pulse energy release. This may be due to a shortcoming in the space-independent model for the mixed HEU/LEU core or due to peak power being limited because of the relative amount of time required for transient rod movement versus pulse power rise time.

The shutdown margin for Core 35A was calculated with 'all rods in' (i.e. all control rods completely inserted into the core). It was possible to correlate the calculated values with experimentally measured shutdown margin values. The core excess was calculated under an 'all rods out' condition. Integral reactivity curves were determined for each control rod by the timed power increase method. The MCNP model and experimental data were in reasonable agreement in each case.

5.1.6. Request for additional information process

The NRC subcontracted a review of the 2007 SAR to subject matter experts at Brookhaven National Laboratory. A series of RAI regarding the 2007 SAR were generated; the RAI were forwarded to WSU by the NRC. Responses to the RAI were prepared by Argonne National Laboratory, General Atomics and WSU. An SER was generated by Brookhaven National Laboratory after resolution of the RAI. The SER provided an examination of the technical bases contained in the SAR and in the RAI. The NRC issued the conversion order after satisfying closure of all RAI.

5.1.7. Startup of low enriched uranium core

A goal of the HEU to LEU conversion was to exchange the FLIP fuel with 30/20 LEU TRIGA fuel on a rod by rod basis without causing significant adverse impact on reactor performance. The 30/20 fuel was qualified by the NRC

as acceptable for use in TRIGA fuelled reactors, was commercially available through CERCA (a subsidiary of AREVA), and was suitable in the sense that its performance characteristics are very similar to FLIP in fuel lifetime and reactivity. New graphite reflectors had been obtained to replace the original reflectors, and conversion was an appropriate time to replace the reflectors. Subsequent loading of the fuel in the reactor was carried out while measuring the approach to criticality; criticality was reached on 7 October 2008. Power calibration and control rod calibration were carried out to determine core excess (limited to 5.6% $\Delta k/k$), and shutdown margins; the core became steady state operational on 17 October 2008. Steady state operation resumed, and continued during which time samarium and other fission products accumulated in the reactor to a steady state level at approximately 370 MW·h of operation of the entire LEU core. Pulse testing was carried out in January and February 2009, and was completed on 6 March 2009. Examination of pulse testing data was conducted and Core 35A was declared fully operational on 15 April 2009.

5.1.8. Performance characteristics of mixed HEU/LEU versus LEU core

The LEU core behaves similarly to the mixed HEU/LEU core. For example, power peaking is similar, as evidenced by comparable values for indicated fuel centreline temperature in the IFE. An epithermal beam port exhibited a reduction of intensity by about 5%, apparently due to inelastic scattering of the higher uranium loading of the 30/20 fuel. Reactor pulsing behaviour with LEU fuel shows similar peak energies for given amounts of transient reactivity insertion, although a reduction in energy release. The reactor is occasionally used in pulse mode for experimental purposes; the reduction of energy release has not posed problems for experimental irradiations.

5.1.9. Fresh and spent fuel shipping

A licence modification request to increase the possession limit for [235]U was made to the NRC on 23 January 2008. An increased possession limit was necessary to accommodate the fuel inventory arising from receipt of fresh 30/20 fuel and the older 8.5/20 LEU fuel that was in storage after introduction of the FLIP fuel in 1976. The fresh fuel was sent in two separate shipments from the CERCA fuel fabrication facility in Romans-sur-Isère, France, to the Nuclear Radiation Centre at WSU in Pullman, Washington. A receipt inspection for each fuel rod was carried out immediately upon removal of the fresh fuel from the shipping casks. Serial numbers were verified, the fuel rods were visually inspected and photographed, and dimensions measured. All of the fuel rods passed the inspection.

The HEU fuel that was removed from the reactor in September 2007 was transferred to a spent fuel storage area in the reactor pool, where it remained while making arrangements for shipping. The used HEU fuel was shipped in August 2009 from WSU to a fuel storage facility at INL.

5.2. NON–US TRIGA HEU TO LEU REACTOR CONVERSIONS

In addition to the US TRIGA conversions — both TRIGA HEU – TRIGA LEU and MTR HEU – TRIGA LEU (first part of Table 7) — several TRIGA HEU to LEU conversions have been performed with research reactors outside the United States of America. In addition to the MTR HEU – TRIGA LEU conversions outside the United States of America that are listed in the second part of Table 7, three other TRIGA reactors originally fuelled with TRIGA HEU fuel assemblies have been converted to high density TRIGA LEU fuel:

(a) The TRIGA MPR-16 reactor at the Institute of Nuclear Research, Piteşti, Romania;
(b) The TRIGA Mark III reactor at the National Institute for Nuclear Research, Mexico City, Mexico;
(c) The TRIGA Mark II reactor at the Atominstitut, Vienna, Austria.

The first has been described in detail in Section 3 and will not be discussed further. The experience of, and lessons learned from, the second reactor conversion is very similar to the case study for WSU above, with the exception that NRC involvement was replaced with that of the National Commission for Nuclear Safety and Safeguards (Comisión Nacional de Seguridad Nuclear y Salvaguardias, CNSNS) as the cognizant regulator and substantial assistance was provided by the IAEA. Briefly described below is the MTR HEU – TRIGA LEU conversion of the research reactor at the Institute of Nuclear Science and Alternative Energies (Instituto de Ciencias Nucleares y Energias Alternativas, INEA) in Bogota, Colombia, and a mixed HEU/LEU core to a full LEU core conversion of the TRIGA Mark II reactor at the Atominstitut, Vienna, Austria.

5.2.1. Institute of Nuclear Science and Alternative Energies

Since 1965, the Institute of Nuclear Affairs (IAN) — and subsequently the Colombian Institute of Geology and Mining (INGEOMINAS) — has operated a small 10 kW(th) research reactor, known as the IAN-R1 reactor, which was subsequently upgraded to 30 kW(th) in 1980. This reactor was provided to Colombia under the US programme Atoms for Peace, and had been fuelled

TABLE 7. RESEARCH REACTORS CONVERTED FROM MTR TO UZrH$_X$ TYPE FUEL

Facility	Country	Original TRIGA fuel type (nominal wt% U–^{235}U enrichment)[a]	Steady state power after conversion (kW)	Original TRIGA conversion startup
Pennsylvania State University		12–20	1000	1965
University of Maryland		8.5–20	250	1974
University of Wisconsin		8.5–20	1000	1967
Washington State University		8.5–20	1000	1967
Idaho National Laboratory	USA	8.5–70[**]	250	1977
Texas A&M University		8.5–70[**]	1000	1968
Aerotest Operations, San Ramon, California		12–20	250	1965
Puerto Rico Nuclear Center[b]		8.5–70	2000	1972
Office of Atoms for Peace	Thailand	8.5–20	2000	1977
Institute of Nuclear Science and Alternative Energies	Colombia	12–20	100	1997
University of Frankfurt[b]	Germany	8.5–20	1000	1977
National Tsing Hua University[b]	China[c]	8.5–20	1000	1977
Philippines Nuclear Research Center[b]	Philippines	30–20	2000	1988

[a] Reference to "nominal 20%" for the ^{235}U enrichment means ^{235}U enriched to just under 20%, or approximately 19.7 wt% ^{235}U.
[b] Decommissioned.
[c] In Taiwan, China.

with MTR HEU fuel, enriched nominally to 93% ^{235}U. With the cooperation of the IAEA, a gradual reactor upgrade programme was undertaken, beginning in 1987. The IAN changed its structure and also its name into Institute of Nuclear Sciences and Alternative Energy (INEA) in 1992. In 1994, the IAEA and INEA entered into a tripartite contract with General Atomics to perform an HEU to LEU conversion of the R-1 reactor, including: preparing the SAR; manufacturing TRIGA type LEU (19.7% enriched) fuel assemblies to replace the original MTR HEU fuel plate assemblies; operator training; upgrading the reactor power to 100 kW(th); carrying out additional upgrades to the I&C system, auxiliary reactor systems; and commissioning the converted, upgraded reactor as a TRIGA conversion reactor that was successfully completed in 1997.

The original HEU core was a flat plate design consisting of a 4 × 4 array of fuel elements surrounded by 20 graphite reflector elements. The TRIGA LEU core used the same core geometry, and replaced the HEU plate assemblies with TRIGA 12 wt% uranium, LEU four rod fuel cluster assemblies, including standard four rod and three rod control rods, and in core experimental fuel assemblies. The final core configuration in 1997 is given in Fig. 29.

As mentioned earlier, this conversion was carried out under a tripartite arrangement between the IAEA, the reactor owner (INEA) and the reactor vendor (General Atomics). Under such an arrangement, not only were all contractual arrangements supported by the IAEA, but significant technical support activities
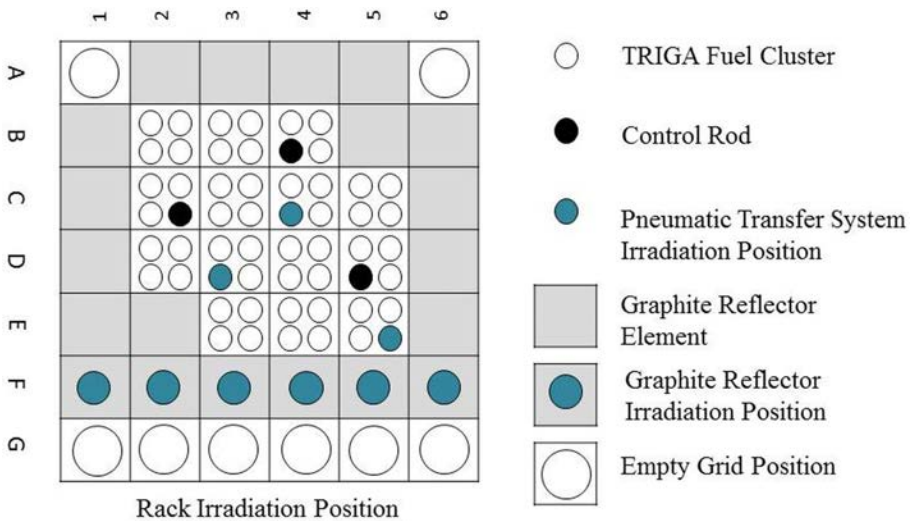
FIG. 29. Core layout of INEA MTR HEU to TRIGA LEU conversion (courtesy of General Atomics and INEA, Colombia).

were provided by cognizant experts at all stages of the conversion and upgrade process, including overall project management support, review and acceptance of the SAR, inspection and acceptance of as manufactured fuel assemblies, installation of auxiliary equipment, and fuel loading and commissioning operations by the reactor vendor.

The reactor shielding, which was designed for the original 10 kW operations, was not upgraded during the conversion and power upgrade to 100 kW with TRIGA LEU fuel, relying on administrative controls to achieve the desired radiation safety during operations at 100 kW. Currently, the Colombian regulator has limited the operations to 30 kW for ensuring occupational radiation safety.

5.2.2. TRIGA Mark II reactor at the Atominstitut

The TRIGA reactor in Vienna operated for many years with a mixed core using aluminium clad and stainless steel clad LEU fuel and a few stainless steel HEU fuel elements. In view of the US spent fuel return programme, the average age of these fuel elements and the Austrian position not to store any spent nuclear fuel on its territory, negotiation started in April 2011 with the DOE and the IAEA. The sensitive subject was to return the old TRIGA fuel and to find a solution for a possible continuation of reactor operations for the following decades. As the TRIGA reactor in Vienna is the closest nuclear facility to the IAEA headquarters, there was great interest at the IAEA to have an operating research reactor nearby, as historically close cooperation exists between the IAEA and the Atominstitut.

Negotiation involved Austrian ministries, the IAEA, the European Supply Agency and the DOE, and concluded that Austria would return 91 spent fuel elements to INL, while INL would provide 77 very low burnup stainless steel clad LEU elements for further reactor operation of the TRIGA reactor in Vienna. The title to these 77 new fuel elements will be transferred to European Atomic Energy Community (Euratom) in accordance with Article 86 of the Euratom–US Treaty. The fuel exchange with the old core returned to the INL, and the new core transferred to Vienna was carried out in one shipment in late 2012 through the ports of Koper, Slovenia, and Trieste, Italy.

The administrative, logistic and technical preparations of the fuel exchange took more than one year, as this fuel exchange was unique worldwide and was the first of its kind between Austria and the United States of America. The whole exchange procedure, from the arrival of the 'new' fuel to the return of the spent fuel, was performed successfully in two weeks in November 2012. Detailed information on the fuel swap can be found in the individual contribution by Villa et al. on the CD-ROM accompanying this publication.

5.3. LESSONS LEARNED

It is important to emphasize the value of active project management by an experienced person who is specifically assigned to the HEU to LEU conversion project, who carries responsibility and accountability and is not subject to competing demands for time. All project participants in the conversion programme need to take ownership of their respective roles in the sense that they will provide the dedicated effort at each stage and for each activity in the conversion.

As an example, the conversion of the IAN-R1 Reactor, in Colombia, was carried out under an IAEA contract. After the original HEU fuel had been removed and the replacement TRIGA type fuel had arrived at the facility, the reloading of the new fuel — including the critical experiment and other startup related experiments — was carried out in a relatively short of period within ten working days in August 1997. This demonstrates that with proper advanced planning and close cooperation between the involved organizations, a TRIGA HEU to LEU conversion can be efficiently carried out.

There were no video or photographic records of the 1997 conversion from MTR to TRIGA fuel at IAN-R1, nor were any available for the subsequent transport operations of the MTR fuel. While loading and unloading a TRIGA reactor core is not a routine undertaking at the facilities, it is done frequently enough (e.g. for annual fuel inspections) that with the proper training of operations staff and the presence of General Atomics experts during the process, there was adequate, combined institutional experience to carry out the procedure without difficulty. However, shipping spent nuclear fuel had not been done at any time in the tenure of the staff employed at that time. Consequently, there was much careful planning and rehearsal for each step. The process of moving the spent nuclear fuel from the pool and transferring it to a shipping cask, however, was recorded in high resolution video for future reference.

For the WSU conversion, one staff member from the WSU Nuclear Radiation Centre was sent to another university reactor facility to observe the activities that were carried out for the spent nuclear fuel shipping campaign. Personal observation proved to be invaluable, as it compensated for the lack of institutional knowledge at WSU.

6. AGEING MANAGEMENT OF TRIGA REACTORS

Since the early days of TRIGA reactor operation, a number of modifications, reconstructions and improvements of various reactor systems have been carried out in most of the TRIGA reactors to remediate ageing issues and to replace outdated components which might have disappeared from the market. The procedures were usually based on detailed inspection plans covering all safety related reactor components and systems to be re-examined regularly [43–45]. As the majority of operational TRIGA research reactors are more than 45 years old (see Fig. 30), ageing management of these facilities becomes crucial in order to continue their operation (see Fig. 31). This section describes the different challenges in ageing management and important tasks to address ageing related safety concerns [46–49].

A very important component in ageing management for TRIGA reactors is the periodic visual inspection of the reactor fuel, reactor tank internals and the regular cleaning of the primary water system. Inspection of fuel is performed using handling and dimensional tools provided by the reactor manufacturer. Visual inspection of the fuel as well as reactor tank internals can also be carried out either with a watertight endoscope or with commercially available underwater cameras. The endoscope is a modular optical device which allows optical inspection in any place of the reactor tank, including the fuel elements in the core. With integrated lights and various objectives, practically all areas in the
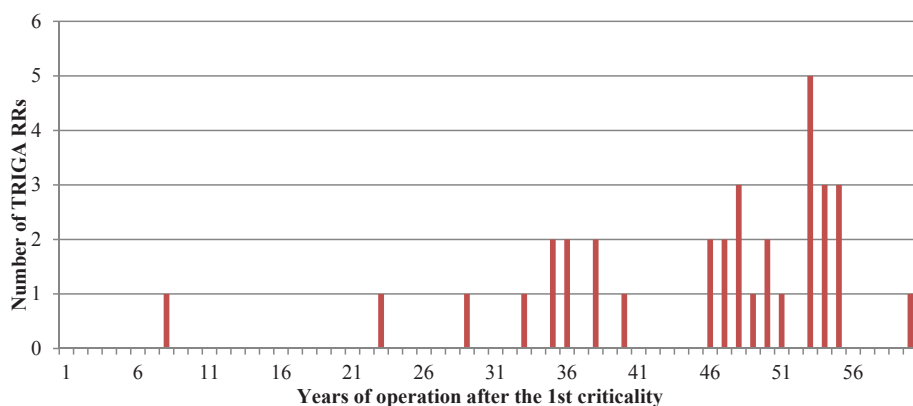
Source: IAEA Research Reactor Data Base.

FIG. 30. Age distribution of 34 operational TRIGA research reactors.
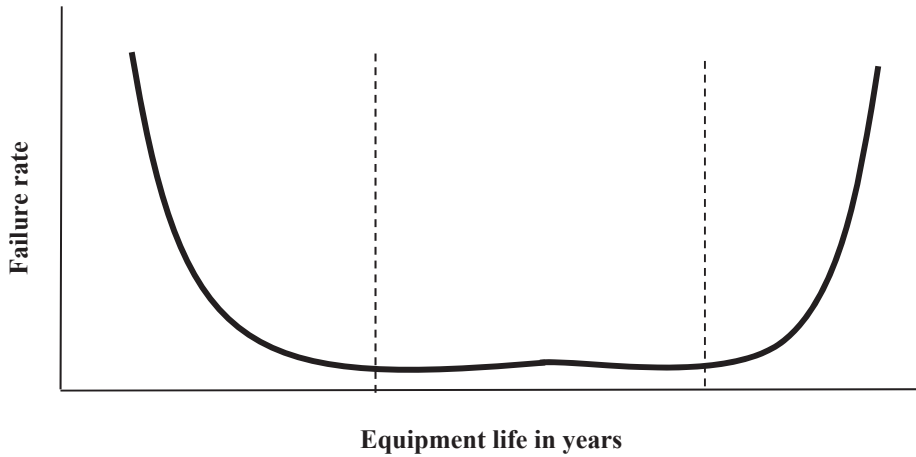
FIG. 31. Classic 'Bathtub' Reliability Curve, which can also be applied to the research reactor technology.

reactor tank can be inspected. Another possibility is to use underwater cameras; these are, however, more radiation sensitive than endoscopes.

Regular cleaning of tank internals is equally important, using a high pressure water jet to stir up all deposits from tank surfaces, in addition to a special pump with integrated filters to collect the deposits if necessary (see Fig. 32).

FIG. 32. Collected pieces in the coarse filter (courtesy of Atominstitut, Vienna).

The endoscope, together with the water jet and the tank cleaning pump, has been successfully applied in the past in several research reactors through bilateral cooperation and assistance among the TRIGA community (see Section 6.2). In fact, in one case the visual inspection and maintenance saved the operator tedious repair work of several months or a possible permanent shutdown [50].

6.1. IN-SERVICE INSPECTION OF TRIGA FUEL

In TRIGA reactors, periodic inspection of fuel elements is a key component of ageing management to ensure nuclear fuel integrity during the life of the fuel. Individual fuel elements have to be examined periodically with an underwater measuring device provided by the manufacturer for clad elongation and bowing to ensure they meet performance requirements, and which are typically incorporated into the reactor licence through technical specification requirements. Equally important is the use of underwater cameras to inspect for clad defects and to ensure its integrity. The periodicity of such inspections can vary; for pulsing reactors, these inspections are usually carried out annually or every two years, depending on the pulsing frequency. For fuel operated in steady state mode only, these requirements can vary from not being required at all to less frequently than pulsing reactors. The failure rate of fuel from elongation, bowing or other clad defects is varied, depending on the frequency and nature of pulsing operations, as well as cycling between steady state and pulsing operations. Fuel removals due to bowing, elongation or other clad defects are rare in TRIGA reactors, especially when the duty cycle is low and there is proper maintenance of the reactor pool water in accordance with manufacturer's specifications, which are designed to inhibit corrosion of the cladding as well as reactor internals. For example, in the case of the TRIGA reactor in Vienna, where the duty cycle is relatively high, only 8 out of 104 fuel elements had to be removed during more than 50 years of operation: only one due to a cladding defect; the others due to excess elongation.

In addition, based on many years of experience of operating TRIGA research reactors, the TRIGA managers recommend storing spent fuel in the pool itself or dry fuel storage under a controlled atmosphere (nitrogen) to prevent further corrosion while in storage [51–53].

6.2. TYPICAL IN-SERVICE INSPECTION AND CLEANING EQUIPMENT

At TRIGA reactors, in-service inspection (ISI) is usually carried out, in addition to the irradiated fuel elements, on components — including those that are not directly accessible owing to a high radiation level, such as the reactor

tank and the core structure. For these ISI inspections, tools and methods have been developed based on experience in non-nuclear applications and modified or adapted to the nuclear environment. Other non-radioactive components may be inspected with methods used in conventional industries. The following methods and tools are typically used in a TRIGA Mark II reactor but may easily be adapted for any other low or even high power TRIGA research reactor:

(a)   Visual inspections using:
        — Nuclear underwater telescope;
        — Underwater endoscope;
        — Underwater camera using radiation hardened systems.
(b)   Replica material.
(c)   Tank cleaning pump.
(d)   Underwater jet.
(e)   Underwater lights.
(f)   Underwater brush.
(g)   Ultrasonic equipment.

6.2.1.   Visual inspection

6.2.1.1. Nuclear underwater telescope

Nuclear underwater telescopes are high resolution devices (resolution 0.1 mm) with continuously variable magnification. They allow remote underwater viewing of the reactor tank and core components, such as fuel elements and core support structures, both vertically and horizontally. Such a telescope penetrates the water level while the water fills up the periscope tube, providing complete radiation shielding for the viewer. Since no radiation sensitive optical element is built in at the lower end of the unit, the optical image quality is not influenced due to radiation induced defects. In order to facilitate acquisition of the object and detail observation, the magnification can be continuously controlled. Photo and video recording is also possible for some types of equipment [54].

6.2.1.2. Underwater endoscope

For the inspection of the inner surface of neutron beam tubes or internal core structures, a modular endoscope is found to give excellent results (see Fig. 33). A typical system consists of a set of 1 m long (18 mm diameter) ocular and rigid optical extension pieces.
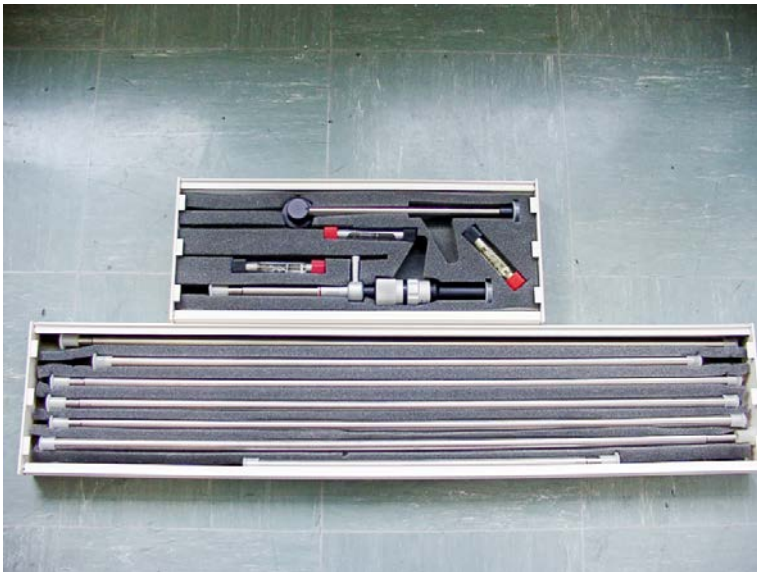
FIG. 33. Underwater endoscope (courtesy of Atominstitut, Vienna).

These modules can be coupled to the desired length, up to several metres. The front end of the endoscope houses the objective together with an integrated 100 W/12 V lamp powered by a transformer. Various objectives with forward, 45° forward, 90° and 45° backward viewing angles are available. Photos or videotapes can also be taken through the endoscope for a permanent record. In case of gamma radiation streaming out of the beam tube, the ocular can also be mounted at an angle of 90° and viewing can be performed from outside the radiation field. Some systems have flexible sections that may turn as needed to reach remote areas [55]. During the last ten years, endoscopes have, however, been replaced by underwater TV cameras.

6.2.1.3. Underwater camera using radiation hardened systems

Some facilities may use specially designed underwater video cameras or place a video camera inside a watertight housing to perform routine or non-routine ISI. Often, a set of underwater lamps is necessary to illuminate the object deep inside the reactor pool. The output from the camera may be sent to a video monitor or recorder for on-line or off-line inspection (see Figs 34 and 35) [56].
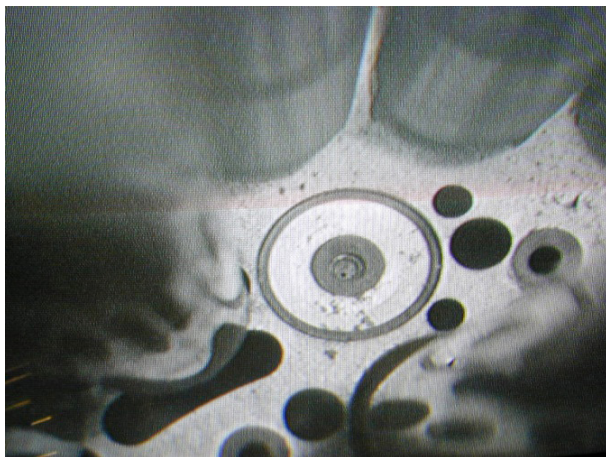
FIG. 34. Lower grid plate (courtesy of Atominstitut, Vienna).
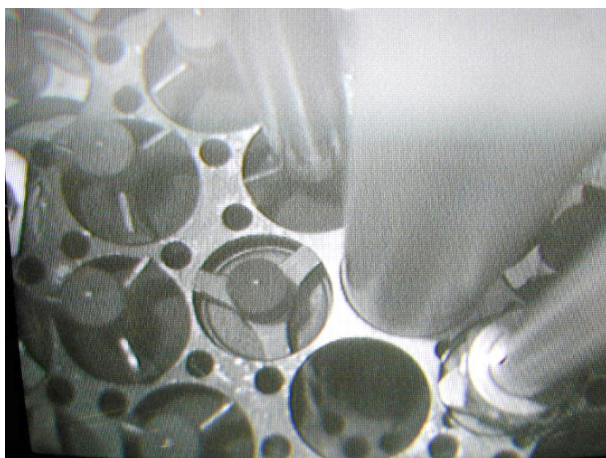
FIG. 35. Upper grid plate (courtesy of Atominstitut, Vienna).

6.2.2. Replica material

To determine the dimension of a corrosion spot (i.e. the surface structure of small activated items in the core region), a two component silicon based material (similar to that used by dentists) has been found to be very useful. Typically, a plastic cap of a powder bottle is mounted at the end of an aluminium rod and filled with the mixed silicon paste (see Fig. 36). This material remains soft or

FIG. 36. Replica material to determine the dimension of a corrosion spot (courtesy of Atominstitut, Vienna).

pliable for about 3 minutes in ambient air and water. The rod is then lowered into the reactor tank (water temperature around 30°C) and immediately pressed on the corrosion crater for 4–5 minutes. Within this period, the silicon paste hardens completely and the system can be removed from the reactor tank. The hardened material gives an exact replica of the corrosion crater for further investigation [44].

6.2.3. Tank cleaning vacuum with integrated filters

Dirt or debris in the reactor tank may cause cloudiness or potentially cause thermal and hydraulic problems within the reactor fuel. The most effective manner of keeping the reactor tank clean is to eliminate the source by covering the top of the pool with acrylic glass and by being careful not to drop materials into the water when working above the open pool. Most research reactors have some system of purifying the primary coolant. These systems are generally not designed to remove relatively large debris that sinks quickly to the pool bottom.

A conventional plastic pump used for cleaning swimming pools has been found to be useful to clean the tank bottom from small debris. A typical system is equipped with a coarse filter to collect larger objects (such as screws) and 12 units of candle type fine filters for collecting small particles. One advantage is that these fine filters are reusable, and they may be washed and reinstalled into the pump (see Fig. 37). Some reactor facilities repeat pool cleaning periodically.
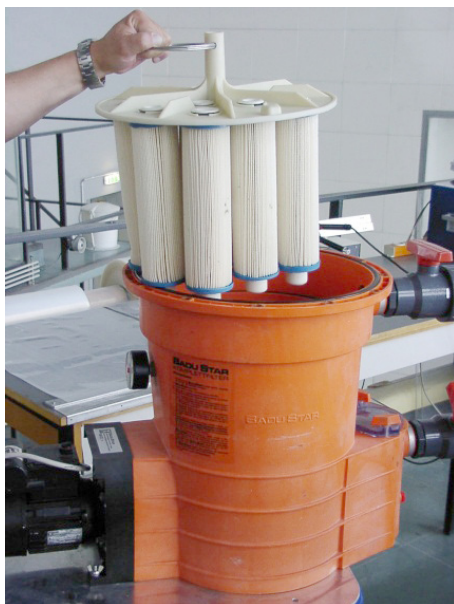
FIG. 37. Tank cleaning pump with integrated filters (courtesy of Atominstitut, Vienna).

6.2.4. Underwater jet to remove deposits

A tool that has been found to be very useful in cleaning remote areas in reactor tanks from debris is a strong water jet (16×10^6 Pa) produced by a portable compressor together with different types of jet nozzle. The material stirred up from the tank bottom or any deposits removed from the tank wall will ultimately be collected in the filters of the water purification system. It is important to remove the material quickly with a local vacuuming system as described in Section 6.2.3. Some of these jet nozzles are small enough to be inserted through a hole in the top grid plate right into the core volume and can be used to clean the core of debris or corrosion deposits. Operators need to be cautioned that

high pressure water jets can cause damage to sensitive reactor components and therefore the jet should not be aimed directly at fuel elements.

6.2.5. High intensity underwater lights

Miniature, high intensity underwater lamps are necessary to inspect remote areas in reactor tanks. Generally, this is done in conjunction with the use of an underwater camera or a pair of binoculars used at the pool surface. This 24 V DC lamp (13 cm length, 6 cm diameter) has a power of 250 W and can only be operated under water. The lamp, mounted on modular 1 m aluminium tubes that are coupled to the desired length, can be directed to any desired position in the reactor tank for optimal viewing. Another useful system for illuminating objects underwater has been the high intensity directional lamp used from the pool surface. These 12 V DC lamps are usually extremely bright (1 000 000 candela) and focused in a very tight beam of about 6–10 cm in diameter.

6.2.6. Rotating underwater brush

In many areas of a reactor tank, small surface spots of corrosion may be seen during inspections. If desired, these spots can be brushed away using an underwater rotating brush connected to a standard drilling machine by an extension shaft. Practically all areas inside the reactor tank can be cleaned using various types of brush (radial and pot type). As with other cleaning equipment used around the reactor, operators need to be extremely cautious to prevent damaging the object they are attempting to clean.

6.2.7. Ultrasonic equipment

Ultrasonic inspections can be performed under water. The underwater technique requires some different preconditions from inspections carried out from outside the water. The procedure is basically prepared as an immersion ultrasonic technique. First, the ultrasonic probe as well as cable connections need to be watertight. The ultrasonic probe needs to be at a given distance from the metal sheet, and this distance should be maintained precisely during the entire measurement. The ultrasonic probe should be fixed to a guide tube with a tilting connection to allow inspection in different directions. A typical ultrasonic emitter may apply a vibrator 5 mm in diameter and with a nominal frequency of 5 MHz. It is obvious that the measuring system has to be calibrated in a laboratory room and later again on-site. These procedures are also used for training the inspection personnel. Inspection on-site is usually performed from a platform above the reactor tank. Usually, the scanning follows the same vertical lines as for visual

inspection. Due to the geometry in the transition between the wall and bottom of the tank, it is normally difficult to maintain a constant distance from the metal; therefore, some areas are not accessible for this type of inspection [53].

6.3. EVOLUTION OF INSTRUMENTATION AND CONTROL SYSTEMS FOR TRIGA RESEARCH REACTORS

From the mid 1950s, I&C systems for research reactors evolved parallel to the development of I&C systems for nuclear power plants, but given that research reactor I&C did not need to meet all the rigorous standards applied to nuclear power plant systems, the designs could be simplified. And even though the nuclear power plant market was much more demanding, it was more than the research reactor market, so the number of suppliers for research reactors was limited. When orders for nuclear power plants and research reactors decreased in the late 1970s, it became increasingly difficult and costly for smaller research reactors to acquire a new I&C system. Only a handful of companies worldwide were able to supply acceptable design research reactors I&C systems and at an acceptable price. In addition, requirements from regulatory authorities became more demanding by applying, in some countries, requirements for I&C systems following nuclear power plant standards such as redundancy, diversity and seismic resistance, which also had a major impact on the I&C system costs. Additional factors to be considered during the evolution of research reactor I&C systems are the requirements for recalibration, maintenance, documentation and the human–machine interface. The following sections describe the evolution of TRIGA research reactor I&C systems from the early days to the present.

6.3.1. Generation 1

When TRIGA reactors were developed in the mid 1950s, the typical state of the art I&C systems were based on vacuum tube based electronics, and used to monitor basic reactor parameters such as thermal reactor power, reactor period, fuel and water temperature, dose rates, radioactivity in air and water, and several other classical parameters. If safety related parameters deviated from a given range, safety driven counter actions, or even a reactor shutdown, had to be initiated. All these criteria were already taken into account with such simple I&C systems based on electronic tubes (see Fig. 38).

FIG. 38. Generation 1 TRIGA operating console (courtesy of General Atomics).

6.3.2. Generation 2

Towards the mid 1960s, solid state transistors replaced vacuum tube electronic in I&C systems, and a new type of TRIGA instrumentation based entirely on transistors was marketed. During the 1970s, therefore, many TRIGA reactors converted to such a type of I&C systems; and some of these are still in use (see Fig. 39). This type of reactor instrumentation was easy to maintain and spare parts could easily be replaced for about 20 years, until they slowly disappeared from the market.

6.3.3. Generation 3

With the advent of personal computer technology in the 1980s, a new type of digital, then state of the art, software based I&C system was pioneered by General Atomics and became available from General Atomics as well as other vendors from the late 1980s onwards. As software based I&C systems were usually not accepted by the regulatory authorities, especially for safety related system performance, parallel hard wired shutdown systems for the licence required safety related parameters to continue to be used, and therefore an
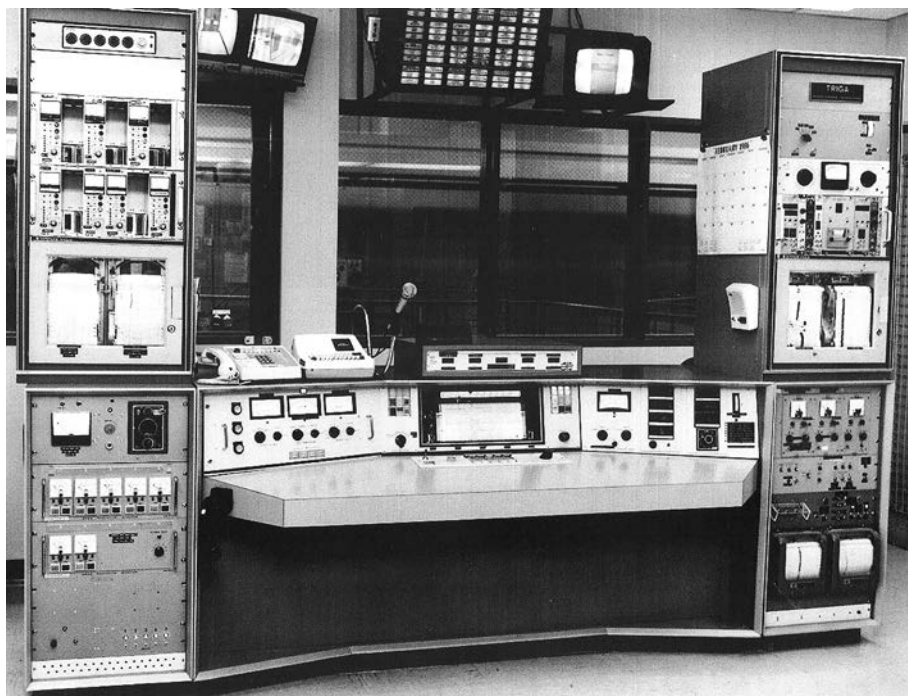
FIG. 39. Typical installation of a Generation 2 TRIGA I&C system based on solid state components supplied by General Atomics (courtesy of General Atomics and Oregon State University, United States of America).

analogue–digital hybrid remained the state of the art I&C systems in most of TRIGA research reactors worldwide and continues today. An early Generation 3 design is shown in Fig. 40. And even though analogue based instrumentation continued to be utilized, particularly for subsystems that were required to perform a safety function, their designs evolved into compact, modular units with modern, state of the art circuit designs and components. These systems also allowed digital archiving of real time operations data which could be played back at a later time either as a teaching tool or to analyse any anomalies in operation.

Many TRIGA facilities have performed a complete replacement of their analogue I&C system with digitally based systems. While most TRIGA reactors use General Atomics supplied systems, including the nuclear safety instrumentation, some have chosen to use other suppliers to upgrade their older systems. Recently, for example, the TRIGA facility in Malaysia has successfully replaced the Generation 2 I&C system with technology provided by a company from the Republic of Korea. Similarly, the IAN-R1 TRIGA conversion reactor

FIG. 40. With the evolution of desktop computers, PC based I&C systems were introduced and various designs continue to be in use today (courtesy of General Atomics and CNESTEN, Morocco).

in Colombia has upgraded the original Generation 3 I&C system by integrating nuclear safety instrumentation originally supplied by General Atomics with state of the art, Ethernet based monitoring and communications equipment designed by the National Institute for Nuclear Research (Instituto Nacional de Investigaciones Nucleares, ININ) of Mexico.

6.3.4. Generation 4

After more than 20 years use of operating with software based but hardwired digital I&C systems, various components of those systems have again reached their end of life. The situation is also hindered by the non-availability of spare parts for systems installed in the 1990s, software updates, or replacement computer components and old generation computers themselves, such as the situation encountered in Colombia and described above. New Generation 4 digital I&C systems are now being introduced, which can monitor and control variables and parameters of physical and other processes, component and system statuses, as well as react on predefined process limits and safety conditions. These new I&C systems are capable of performing all the functions in both standard and abnormal conditions, including emergency situations. The technical equipment

monitors and records all main parameters that have an impact on facility safety and also collects and stores all the information required for reliable and safe reactor operation. These new I&C systems are equipped with appropriate control and safety devices to keep critical variables within technical specification limits. The individual digital components, which guarantee the communication between the different sensors and the instrumentation and the control systems, communicate with each other using serial bus via fibre metallic and optic cables. With this kind of communication, self-diagnostic testing of the different signals is possible. For all other communications, Ethernet is used. Some schematics and flow chart of a Generation 4 I&C system is provided in Fig. 41.

Via touch screen monitors, modern digital I&C systems offer the possibility to view historical trends of the system relevant parameters during the startup phase and also during long term operation. Relevant parameters such as the reactor period, the fuel temperature or the position of the control rods can be studied directly on parallel screens in the reactor control room. The historical trend of the control rod position can be used to determine poisoning effects and is helpful for burnup calculations. All the data are stored on two parallel hard disks and can be transferred to different storage media for backup and additional visualization. Via Ethernet, the most important parameters of the reactor, such as reactor power, fuel and water temperature, can be transferred to additional screens in the reactor hall and shown to the users, trainees or visitors of the facility.

The system software is developed and tested by standards, methodology and quality assurance procedures required by the regulatory body for similar research reactors. The safety system software development includes a graded approach for research reactors. In addition, new Generation 4 digital I&C systems are designed, manufactured and tested on the basis of general qualification procedures for seismic resistance.

6.4. COOPERATION AND SHARING OF EXPERIENCE

Within the TRIGA community, a continual information exchange and mutual support exist to share both operation experience and development of new methods for maintenance and inspection. For example, the methods described in Section 6.1 have been successfully applied in several TRIGA reactors, such as TRIGA Rome, TRIGA Pavia and TRIGA Kinshasa in cooperation with the TRIGA Vienna.
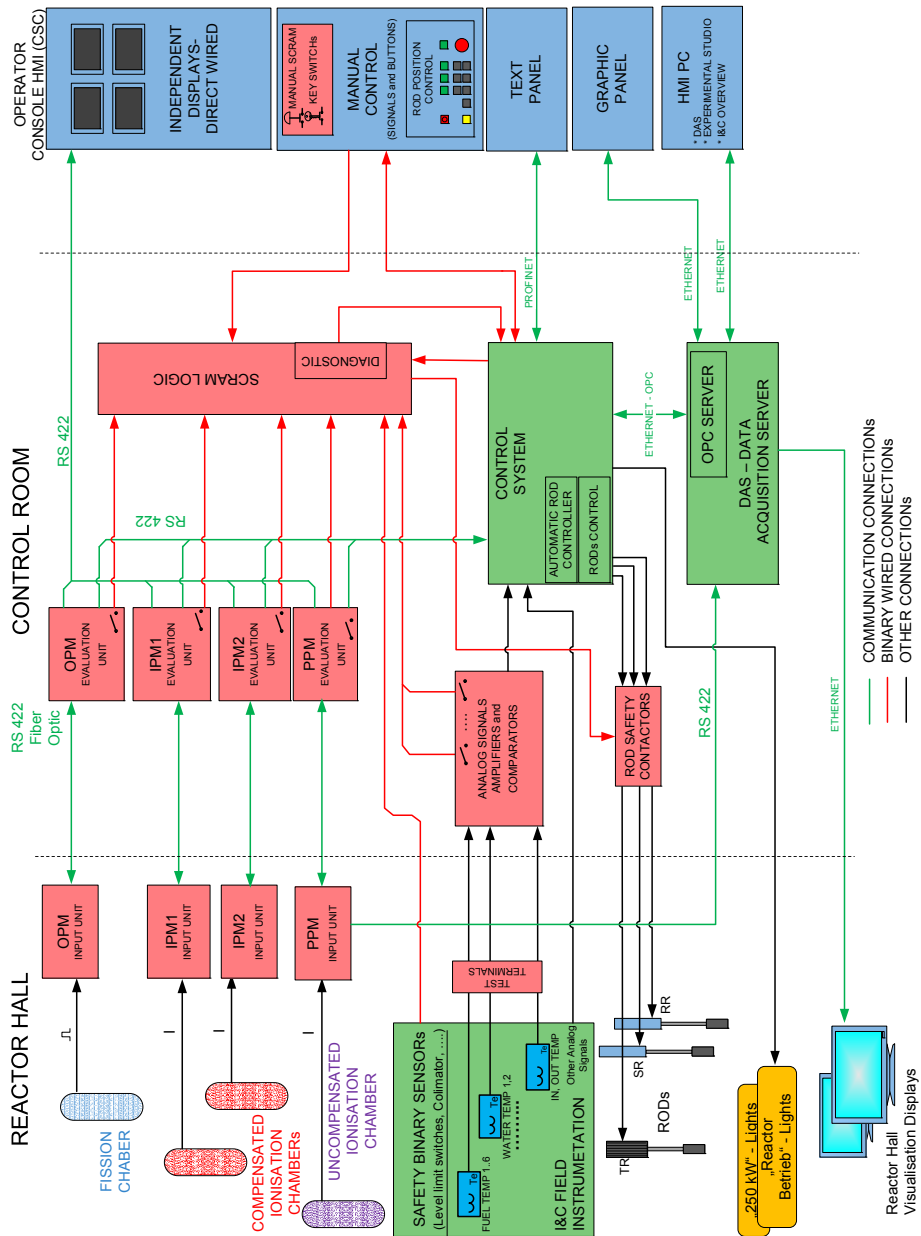
FIG. 41. Flow chart of input parameters and signals as well as their connections and linkage within the modern I&C system.

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6.5. CONCLUSIONS

In spite of the excellent performance of all TRIGA reactors for nearly 60 years, several incidents occurred owing to ageing (and also inadequate preventative maintenance) and which resulted in unplanned financial and human resource investments, as well as loss of utilization time:

(a)   Reactor tank corrosion from outside to inside as a result of overfilling (Mexico) [57];
(b)   Water soaking the graphite reflector due to leaking of the graphite reflector envelope (University of Texas Austin);
(c)   Leakage of beam tubes (Bangladesh);
(d)   Leaking reflector (Oregon State University) [58];
(e)   Corrosion of a TRIGA tank that caused leakage (United States Geological Survey reactor in Denver) [59].

While none of these incidents resulted in any release of radioactivity or overexposure of the public, repairs were costly and resulted in extended reactor outage time.

It is obvious that an effective ageing management programme that integrates maintenance, inspection and periodic testing of the research reactor structures, systems and components has a positive influence on the technical state of the reactor and may extend its lifetime considerably. Reactor facility life extension is best accomplished by establishing and implementing an ageing management programme at an early stage in the facility's operation. High quality, routine maintenance of all reactor safety systems and operation within the established technical specifications are also essential to ensure the safety of the reactor and the public. For nearly 60 years, which is the average age of most TRIGA reactors, maintenance and inspection methods have constantly improved and new methods have been developed.

Together with a strong ISI programme TRIGA reactors around the world can be kept in an excellent technical state without any major ageing effects. It is hoped that many of the existing TRIGA reactors will continue to safely and efficiently operate for many more years to perform all types of application, as described in Section 4.

7. ISSUES AND CHALLENGES OF TRIGA REACTORS

The challenges facing research reactors, including TRIGA research reactors, may change over time, but some generic issues will always be present and are discussed in more detail in this section. The history and characteristics of TRIGA type reactors have been extensively described in Sections 2 and 3. Besides individual challenges facing facilities (e.g. ageing management, refurbishment projects and enhancement of utilization), the main issues for many TRIGA research reactors include the continuing supply of TRIGA fuel in the long term due to a lack of sufficient demand to justify the costs of operating a fabrication facility, back end options related to the spent nuclear fuel return to the country of origin programme (2016/2019 deadline for non–US reactors), underutilization of some reactors, and concerns about continuing technical support at a high level from the original reactor designer and manufacturer as demand for new research reactors continues to weaken.

7.1. DEMAND OF FRESH FUEL

The costs of new fuel elements have risen dramatically since 1996, and there will likely be further increases as regulators impose post-Fukushima safety upgrades on nuclear material handling facilities. Some TRIGA reactors need fuel soon, others have reserves to serve them for a period of time, some others may be shut down, or plan to shut down soon, possibly releasing their fresh fuel for use in other facilities. To evaluate the present situation of operational TRIGA research reactors and to estimate future needs of fuel in the next five years, the IAEA has prepared and sent out a specific questionnaire to operational TRIGA research reactors. Figure 42 shows an overview of global fuel in storage related to the fuel needed in the next five years. According to these preliminary data of 2014, stored and unused fuel can fully meet the demand for the coming five years.

Unfortunately, there are research reactors where the fuel is needed now, and they are obliged to operate their facilities at reduced power/cycles (see Fig. 43). In support of those countries, a future challenge might be to transfer fuel from one country to another.

In a recent core conversion project at the TRIGA reactor in Vienna, Austria, it was proven that slightly irradiated fuel released from a shutdown or decommissioned facility, could be utilized in an operating reactor [59]. This opportunity was made available from 'used' fuel in storage at INL, and facilitated by the DOE through its GTRI and the IAEA to repatriate, to the United States of America, the high enriched fuel formerly at the TRIGA reactor in Vienna.
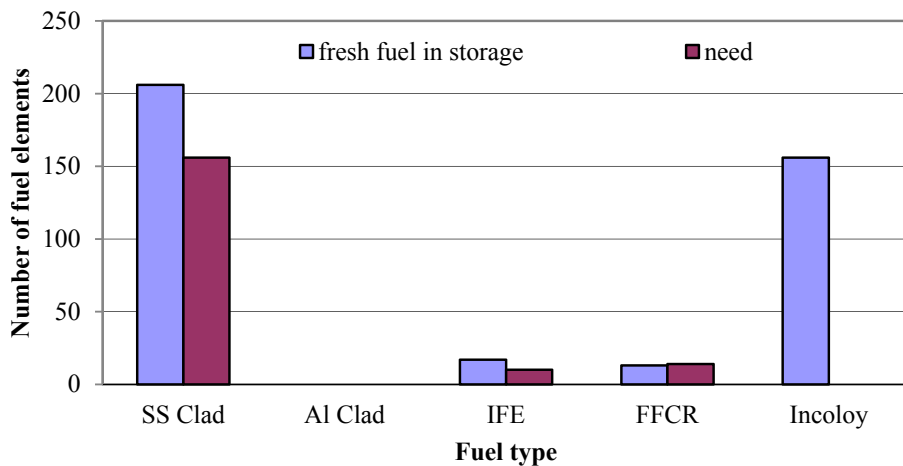
Note: Al — aluminium; FFCR — fuel follower control rod; IFE — instrumented fuel element;
SS — stainless steel; Incoloy — cladded in Incoloy-800 alloys.

FIG. 42. Global situation of TRIGA fuel, representing 20 TRIGA facilities worldwide.
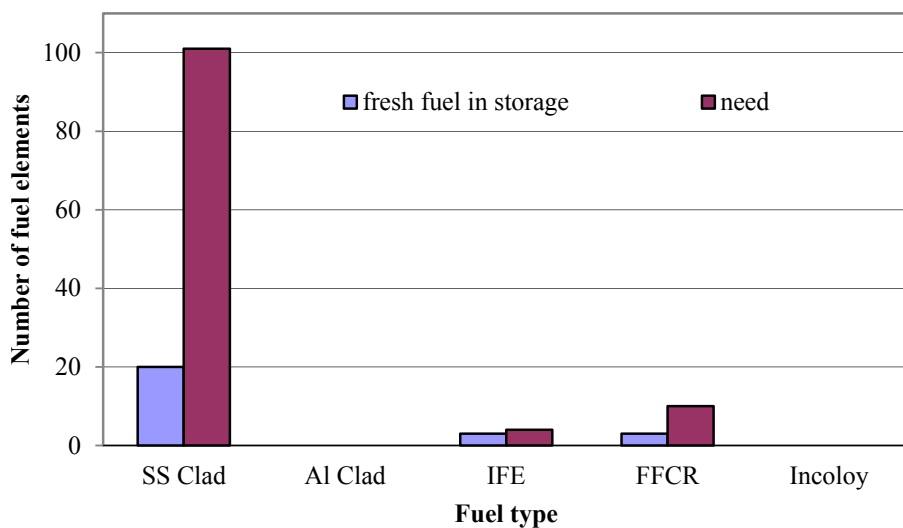
Note: Al — aluminium; FFCR — fuel follower control rod; IFE — instrumented fuel element;
SS — stainless steel; Incoloy — cladded in Incoloy-800 alloys.

FIG. 43. Fuel situation in Indonesia and Thailand.

A further complication is that not all reactors use the same type of fuel. For many years, TRIGA fuel has been manufactured with several different weight percentages of uranium (namely 8.5%, 12%, 20%, 30% and 45%) and also with different external dimensions for standard versus conversion core configurations. While reactors can have the flexibility to mix these various fuel loadings, appropriate analyses and licence amendments have to be done in order to do so and to achieve an acceptable core configuration. Hence, the availability of unused fuel may not meet the needs of all facilities. Central 'exchange' opportunities proposed in regional areas need to be aware of such cases, and an additional survey needs to identify all the available versus required fuel elements, including their type.

Finally, the concern about the lack of a fresh fuel supply can be alleviated as soon as there is sufficiently quantified demand, with a clear commitment from the TRIGA community, to make a clear business case for continued operation of a fuel fabrication facility.

7.2. SPENT FUEL RETURN

Originally, all the facilities and their fuel elements were provided by General Atomics, located in the United States of America. Spent fuel has always been returned to the United States of America for safe and secure storage. By 2013, all HEU elements that were fuelling TRIGA reactors were returned to the United States of America and replaced by high density LEU fuel elements manufactured by General Atomics and provided by the US Government as part of the GTRI programme. All TRIGA research reactors worldwide are now operating with less than 20% enriched fuel.

The DOE has announced that its spent nuclear fuel programme to repatriate US origin fuel will cease in 2016/2019. All fuel to be returned must cease irradiation in May 2016 and all return shipments must be completed by May 2019. The non–US TRIGA community is thus facing the challenge to find final storage for several thousand spent fuel elements (see Fig. 44). Non–US TRIGA research reactors in the near future will need to find alternative spent fuel disposition, or cease operations by 2016. Some initiatives were taken by the TRIGA hosting countries to see whether the repatriation programme could be extended. Several non–US TRIGA facilities will probably continue operating, even if the DOE's intention to cease the fuel return programme is affirmed, and consequently fuel will be used beyond 2019, which requires an alternative solution for their ultimate spent fuel disposition.

Note: Al — aluminium; FFCR — fuel follower control rod; IFE — instrumented fuel element;
SS — stainless steel; Incoloy — cladded in Incoloy-800 alloys.
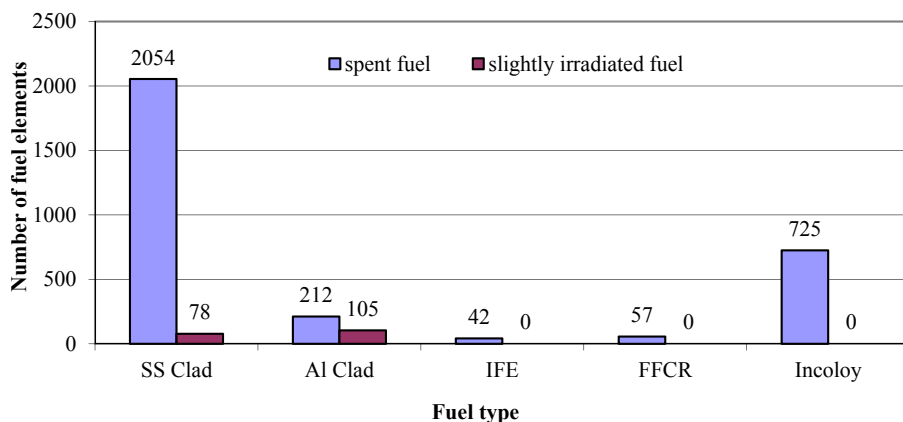
FIG. 44. Global situation of spent fuel as per data collected from 20 TRIGA research reactors worldwide.

Nevertheless, for continuing operation of TRIGA research reactors beyond the 2016/2019 fuel return deadline, Member States should have clear justification and commitment for the fuel back end options. Since the IAEA is fully aware of this situation, regional solutions, by mutual agreement, may be able to be accomplished. However, the available time is running short, and therefore requires coordination followed by immediate steps to be taken by all parties involved.

7.3. ENHANCING UTILIZATION OF TRIGA RESEARCH REACTORS

Other than fuel front end and back end issues, TRIGA research reactors are facing an underutilization challenge (see Fig. 45). The efforts to address the problem include developing and implementing strategic utilization plans, installing additional experimental capabilities, attracting more users and industrial partners, and establishing better relationships with universities and research institutions. In addition, providing facilitated access to the reactors for the countries without such facilities as well as creating regional research reactor networks and coalitions are other means the IAEA uses to assist the research reactor community worldwide.
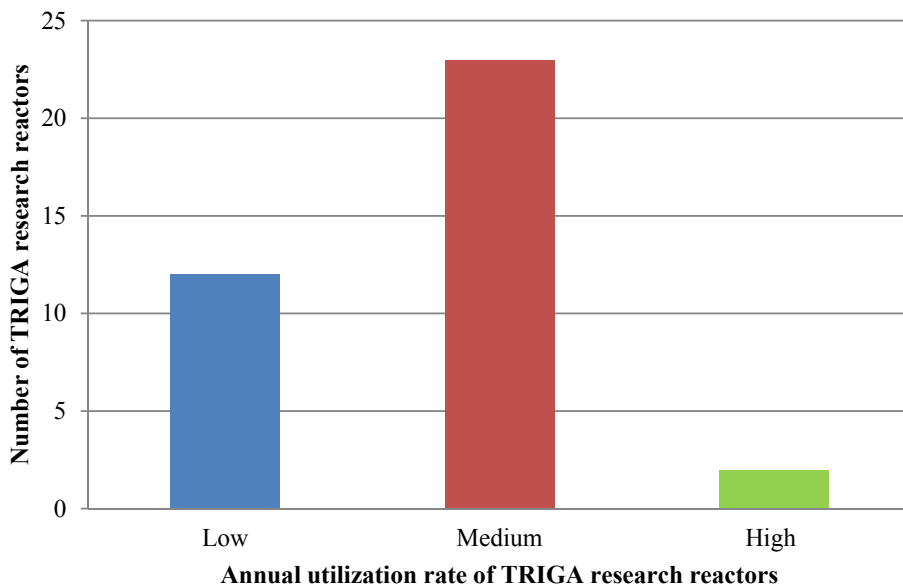
FIG. 45. Distribution of the annual utilization rate of TRIGA research reactors.

Of course, TRIGA reactors in the low to medium power range cannot compete with high power research reactors in many of the industrial applications such as silicon doping, material testing, routine isotope production or providing enough neutrons for state of the art research applying neutron scattering techniques. However, owing to their inherent safety for reactivity insertion events, simple design and numerous experimental facilities, TRIGA reactors play an important role in academic research, education and training as well as in some areas where low operational costs, suitability of low neutron flux and low background, and easy access are advantageous and cannot be provided elsewhere. In this context, international networking and cooperation of TRIGA facilities become increasingly important and usually can be more flexible and more efficient when compared with large scale production facilities.

7.4. TECHNICAL SUPPORT AND AVAILABILITY OF REPLACEMENT PARTS

Like any other type of technical facility, research reactors require constant maintenance, periodical refurbishment, and modernization in order to keep the technical state of the art status and to meet the latest safety requirements. This, of course, is also applicable to all TRIGA facilities still in operation worldwide. TRIGA research reactors have been successfully operated for nearly 60 years, with relatively little major maintenance or modernization. This has been possible as a result of the high degree of safety provided by the robust $UZrH_x$ fuel as well as the simplicity of control and safety systems. Modernization, however, can be expensive as well as time consuming, sometimes resulting in long shutdown periods, interfering with research and education schedules. However, ageing facilities could be kept in operation even at moderate costs and time commitments if sources of originally designed spare parts can be made available.

Some facilities have maintained sets of spare parts; some decommissioned facilities have provided usable parts to help to keep other reactors in operation. Regional inventories or even actual collections of such materials, donated by existing facilities or scavenged during decommissioning should help continued operation. The IAEA will be able to support regional centres for such a purpose. In the United States of America, General Atomics and the TRTR network, especially the TRIGA subnetwork, can provide such information, including the provision of spare parts exchange services.

7.5. QUALIFIED STAFF

As TRIGA reactors age, so do the staff involved. Many of them have participated in the construction or in the early years of operation, and have collected an immense amount of open and tacit knowledge. Nowadays, most of them have retired and a large amount of this knowledge disappeared with them unless there is a considerable parallel transition time between the retiring staff and the successors. It is of the utmost importance that a TRIGA facility deals with this problem at an early stage and before it is too late in order to be prepared for a smooth transition from one TRIGA staff generation to the next [60–64].

8. GLOBAL TRIGA RESEARCH REACTOR NETWORK

In recent years, the IAEA has been promoting networking, coalitions and regional collaboration to improve the efficient and sustainable utilization of research reactors. A number of research reactor coalitions and networks have been developed with the IAEA's support to encourage better utilization of research reactors and to facilitate access for the Member States without such facilities. The coalition/network concept involves putting in place cooperative arrangements among research reactor operators, user entities, customers and other stakeholders. Ideally, a strong partnership is formed, leading to increased utilization of individual research reactors through collective efforts, including improved self-sustainability and self-reliability. The IAEA acts as a catalyst and facilitator towards self-reliance, assists in preparation of strategic and business plans, and also provides initial support via its technical assistance programmes.

With the IAEA support, seven such networks, with shared research reactor facilities and competencies, have collectively offered services to regional and international users, secured entrepreneurial interest, supported upgrading existing or developing new facilities and improved access for countries without research reactors:

— BRRN: Baltic Research Reactor Network;
— CARRN: Central Africa Research Reactor Network;
— CISRRC: Commonwealth of Independent States Research Reactor Coalition;
— CRRC: Caribbean Research Reactor Coalition;
— EARRC: Eurasian Research Reactor Coalition;
— EERRI: Eastern European Research Reactor Initiative;
— MRRN: Mediterranean Research Reactor Network.

The efforts of research reactor coalitions and networks currently involve more than 50 Member States (over 30 have research reactors and over 20 are without such facilities).

However, in spite of this success for creating formal partnerships, more effort is required to achieve the objective of increased and self-sustainable research reactor utilization worldwide. In addition to their individual utilization strategies, coalition partners need to put in place coalition-based common strategic and management plans. They also need to pursue more detailed market analysis and business development to identify specific pay back opportunities through sustainable commercial activities such as complementary marketing and

delivery of irradiation products and services as well as education and training among other revenue generating applications of research reactors.


8.1. FOUNDATION OF A NEW NETWORK

The Global TRIGA Research Reactor Network (GTRRN) was started during the IAEA Technical Meeting on Research Reactor Coalitions: Concerted Actions in the Mediterranean Region, held in July 2012 in Istanbul, Turkey. At this meeting, there was a joint agreement among 11 TRIGA facilities representing 9 countries worldwide that it is important to establish a global network for a number of important and urgent issues to be addressed by the TRIGA community. These include: back end options related to the spent nuclear fuel return programme of the DOE (with key dates of 2016/2019); a potential cessation of operations of the TRIGA fuel fabrication facility (announced by TRIGA International in March 2010); perceived reduction in available technical support from the original designer/manufacturer; and a need for enhanced utilization.

The founding members of this network were representatives from 11 TRIGA research reactors in 9 countries: Austria (represented by Germany), Germany, Finland (represented by Germany), Indonesia (2 TRIGA research reactors), Italy (Pavia and Rome), Malaysia, Morocco, Slovenia and Turkey. On this basis, a global network could include and coordinate regional groups such as operational TRIGA facilities in Africa (2), the Asia Pacific (7), Europe (8) and Latin America (3) in a manner comparable to the existing US located TRIGA research reactors (18).

The formation of GTRRN was further elaborated at the September 2012 National Organization of Test, Research and Training Reactors (TRTR) meeting in San Diego, United States of America, and more recently at the April 2013 European Research Reactor Conference (RRFM) in St. Petersburg, Russian Federation. In addition to TRIGA operators, representatives of General Atomics and TRIGA International were also present.

The formal creation of GTRRN took place during the dedicated IAEA Technical Meeting in Vienna on Research Reactor Coalitions: Global Issues of TRIGA Research Reactors, in November 2013, attended by 17 TRIGA research reactor facilities, distributed over 15 Member States, together with several representatives from the IAEA, DOE and TRIGA International (see Fig. 46).
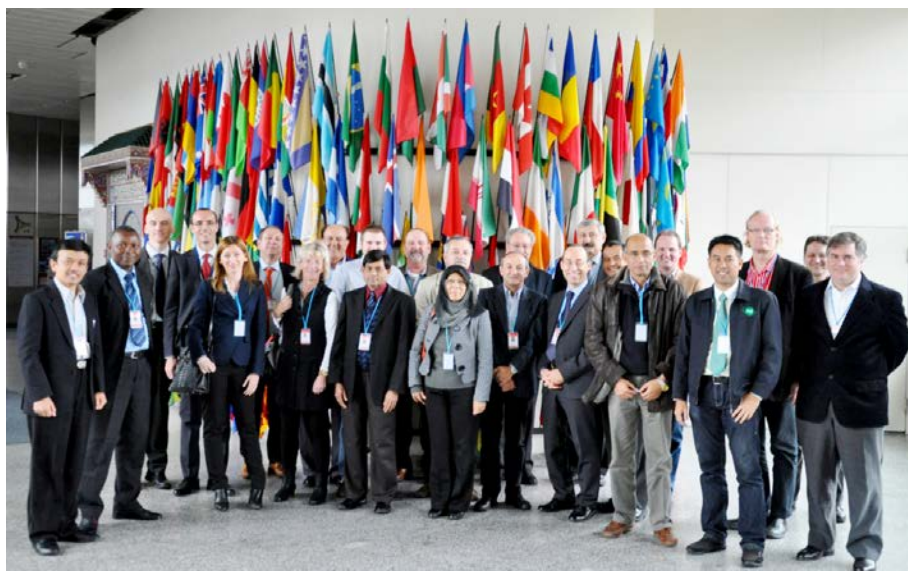
FIG. 46. Participants of IAEA Technical Meeting on Research Reactor Coalitions: Global Issues of TRIGA Research Reactors, 4–8 November 2013, Vienna, Austria.

The meeting defined and established the organizational structure and goals of GTRRN in order:

(a) To strengthen regional and global cooperation among TRIGA facilities towards the development of solutions for common issues and challenges;
(b) To enhance TRIGA reactor utilization through common and complementary products and services as well as exchange of experiences and practices;
(c) To increase viability and visibility of TRIGA reactor operation in the future through contacts and effective relationships with national and international stakeholders.

The network officially formed a steering committee from five members (one per each region) and created a formal agreement of organization through a Memorandum of Understanding (MoU) to be signed by all involved parties. Fourteen members of GTRRN, all TRIGA reactor operators, had signed the MoU by December 2014.

8.2. FUTURE AIMS OF THE GLOBAL TRIGA RESEARCH REACTOR NETWORK

The Executive Summary of the IAEA Technical Meeting in November 2013 describes both the accomplishments and the future aims of GTRRN. In addition to preparing this publication, the following main topics were extensively discussed and constitute the basis for the future work of the network:

— MoU: Agreement to work together and to collect the signatures from the remaining TRIGA research reactors;
— Promote TRIGA utilization and applications, and seek strengthened support from various stakeholders;
— Enhance utilization of TRIGA research reactors;
— Continue to work together on possible TRIGA fuel options (front end and back end);
— Improve the exchange of information and technical experience, including the exchange of experts and available spare parts;
— Create and maintain a dedicated web portal for GTRRN.

9. FUTURE OPPORTUNITIES FOR TRIGA REACTORS

TRIGA reactors continue their significant contribution to new scientific achievements in nuclear science and technology, in the training of a nuclear workforce as additional States vie to achieve energy independence using nuclear energy, and in the preservation of knowledge in peaceful uses of nuclear energy through numerous applications such as radioisotopes for nuclear medicine and non-destructive examinations for industrial applications. Even though many TRIGA research reactors have been permanently shut down, many of the remaining 38 facilities have developed new utilization plans, performed, or are undergoing, major upgrades and refurbishment projects, and scheduled development of future new applications and services. Thanks to their intrinsic safety features, TRIGA reactors will continue to be used in many different areas of education and training as well as for academic research and technology development.

9.1. FUEL RELATED PLANS

Sustainability and long term operation of TRIGA research reactors will strongly depend on the availability of fresh TRIGA fuel elements. Equally, underutilization, a lack of a sufficient budget and unclear spent fuel back end options might lead to some of the facilities to shut down and decommission.

Until fuel production is restarted, the use of slightly irradiated TRIGA fuel from shutdown reactors is one possible option, as chosen in Austria, where INL, using such fuel temporarily stored at its facilities, offered 76 very low burnup stainless steel clad LEU elements for further reactor operation of the TRIGA reactor in Vienna, while 91 spent fuel elements were returned to INL, including HEU fuel. New fuel element availability depends on the restart of the TRIGA International fuel fabrication plant in Romans-sur-Isère, France, which has been shutdown for safety upgrades but is in danger of not being restarted owing to a lack of future fuel orders. The fuel fabrication facility operated by TRIGA International at CERCA's facilities (TRIGA International is a joint venture of General Atomics and CERCA-AREVA, formed in 1995) was relocated from General Atomics in San Diego to Romans-sur-Isère and remained fully operational until February 2011. There is a current call for installation and testing at the upgraded plant in 2016, followed by process qualification and production of reload fuel in 2017.

When it comes to the spent fuel return to the country of origin (in this case the United States of America), the majority of non–US based TRIGA research reactors confirmed that their operation will continue beyond the 2016/2019 deadline as defined by the Take Back Programme of the US DOE in connection with the research reactor spent fuel of US origin — that is, they will not comply with the deadline for the spent fuel return to the country of origin. In addition, all of them indicated that they will look for opportunities whether this deadline could be extended further on exceptional basis for TRIGA type fuel.

9.2. STRATEGIC FUTURE UPGRADE AND UTILIZATION PLANS

Matching the new needs in the field of research reactor applications is a challenge that potentially involves the upgrading of facilities, the installation of new systems and the approval of new activities. The ageing management of reactor structure, services, beam facilities and laboratories also means additional investment and preparation of new strategic plans for research reactor utilization. These plans have to fulfil not only the needs of existing users and consumers or potential needs for enlarging the scope of services and applications, but they also have to comply with current safety standards, regulation requirements and

recommendations. Many ageing facilities should envisage dedicated efforts to ensure the provision of necessary spare parts for uninterrupted reactor operation and smooth maintenance. Furthermore, a number of strategic plans foresee major reactor upgrading and new installation projects such as:

(a) Digital I&C system to fulfil modern safety requirements for I&C systems of TRIGA type reactors.
(b) Modern experimental facilities installed in beam ports:
    — Digital neutron radiography/imaging facility;
    — Other specialized neutron beam tube experiments (e.g. neutron diffraction or PGNAA).
(c) Automation of NAA laboratories to increase analysis capacity and throughput.
(d) Provision of specialized education and training remotely or via the Internet.

When it comes to the requirements for upgrades with a priority on safety related systems for example, among others the following list of system upgrades can be highlighted:

(i) Ventilation system, including emergency ventilation and required filtration system;
(ii) Emergency power supply (batteries or diesel generator) to ensure the operation of the items important to reactor safety in case of loss of electrical power supply (e.g. core monitoring, radiation protection and monitoring systems, ventilation system, emergency lights and fire detectors);
(iii) Integrity and efficiency of the heat exchanger;
(iv) Assessment of the physical status of the beam tubes and control rod drive mechanisms to be performed as soon as possible;
(v) Components of the primary and secondary cooling system (e.g. pipelines and valves) in order to confirm the absence of significant corrosion;
(vi) Reactor internals, including fuel, reflectors and the core support structure, visually inspected at appropriate intervals in order to properly define the ageing management programme;
(vii) Existing radiation monitoring system, including fixed area monitors and portable detectors;
(viii) Reactor safety documents (e.g. safety analysis, SARs, operating controls and limits, and the operational radiation protection and emergency plan).

9.3. FUTURE APPLICATIONS

The present major applications of TRIGA reactors have been grouped and discussed in detail in Section 4. In brief, TRIGA reactors are multipurpose facilities, and the variety of their applications depends on power level, flexibility of irradiation facilities (i.e. rotary irradiation rack, pneumatic transfer system), readiness of beam port facilities (i.e. cold neutron source, diffractometer and neutron radiography), and availability of adequate radiochemistry laboratories — without forgetting qualified engineers and scientists. The potential future fields of TRIGA reactor applications may be expected in the following main areas:

(a)   Both academic and specialized education and training;
(b)   Material science and technology;
(c)   Geochemistry and geology;
(d)   Various branches of industry (e.g. building, car, food, mining and electronics);
(e)   Environment;
(f)   Forensics;
(g)   Radiation science and technology;
(h)   Radiobiology and health physics;
(i)   Nuclear reactor science and technology;
(j)   Nuclear and radiological safety:
    — Waste management;
    — Nuclear safeguards.

The academic staff employed in TRIGA facilities usually have a high knowledge and expertise in selected nuclear areas. This expertise should be made available at a national level as expert services to public institutions (e.g. ministry, regulatory authorities, universities and hospitals), industrial partners as well as for general public including media. On an international level expert services could also support international organizations (e.g. IAEA, Euratom and Preparatory Commission for the Comprehensive Nuclear-Test-Ban Treaty Organization) and other related facilities in emerging nuclear States.

Last but not the least, as many of TRIGA research reactors are located on university campuses, these facilities will continue their contribution to the advances in nuclear science and technology by developing the new generation of nuclear scientists and engineers. In each country, independent of its nuclear orientation, nuclear technology related topics are a matter of public discussion and concerns, and TRIGA reactor facilities contribute to the public acceptance of nuclear science and technologies for peaceful uses and socioeconomic development.

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CONTENTS OF CD-ROM

AFRICA

Moroccan TRIGA Mark II research reactor utilization
B. Nacir (Morocco)

ASIA AND PACIFIC

Current status and future challenge of TRR-1/M1 Thai research reactor
K. Tiyapun (Thailand), N. Klaysuban, S. Boonmark, C. Tippayakul

Puspati TRIGA reactor
Z. Masood (Malaysia)

BAEC TRIGA research reactor
M.A. Salam (Bangladesh), M.A. Sarder, A. Haque, M.A. Malek Soner, M.M. Uddin, A. Kalam, M.M. Rahman, S. Uddin, A. Rahman, M.B. Shohag, I. Kamal, M.M. Haque, M.A. Zulquarnain

Thermalhydraulics analysis and current status of Bandung TRIGA research reactor
E. Umar (Indonesia), K. Kamajaya

EUROPE

TRIGA 14 MW research reactor status and utilization
D. Barbos (Romania), M. Ciocanescu, C. Paunoiu

The past and the future of the TRIGA reactor Vienna
H. Böck (Austria), Y. Hasegawa, E. Jericha, G. Steinhauser, M. Villa

The research reactor TRIGA Mainz
K. Eberhardt (Germany), C. Geppert

ITU TRIGA Mark II training and research reactor
M. Mercimek (Turkey)

TRIGA RC-1 at ENEA C.R. Cassaccia
M. Palomba (Italy), M. Iorio, M. Carta

TRIGA utilization experiences at the University of Pavia
A. Salvini (Italy), D. Alloni, G. Magrotti, S. Manera, M. Oddone, M. Prata

ABBREVIATIONS

ACPR	Annular Core Pulsing Reactor
B_4C	boron carbide
BNCT	boron neutron capture therapy
CERCA	Company for the Study and Atomic Fuel Creation (Compagnie pour l'Etude et la Réalisation de Combustibles Atomiques)
CNESTEN	National Centre for Nuclear Energy, Sciences and Technology
DOE	United States Department of Energy
Euratom	European Atomic Energy Community
FFCR	fuel follower control rod
FLIP	Fuel Life Improvement Programme
GTRI	Global Threat Reduction Initiative
GTRRN	Global TRIGA Research Reactor Network
HEU	high enriched uranium
I&C	instrumentation and control
IFE	instrumented fuel element
INEA	Institute of Nuclear Science and Alternative Energies
INL	Idaho National Laboratory
ISI	in-service inspection
MoU	Memorandum of Understanding
LEU	low enriched uranium
MCNP	Monte Carlo N-Particle
MPR	multipurpose reactor
MTR	materials testing reactor
NAA	neutron activation analysis
NRC	United States Nuclear Regulatory Commission
PGNAA	prompt gamma neutron activation analysis
RAI	requests for additional information
RERTR	Reduced Enrichment for Research and Test Reactors
RRDB	IAEA Research Reactor Database
RRFM	International Topical Meeting on Research Reactor Fuel Management (now European Research Reactor Conference)
SAR	safety analysis report
scram	safety control rod axe man

SER	safety evaluation report
TRIGA	training, research, isotopes, General Atomics
TRTR	National Organization of Test, Research, and Training Reactors
UZrH	uranium–zirconium hydride
WSU	Washington State University
WSUR	Washington State University research reactor
ZrH	zirconium hydride

CONTRIBUTORS TO DRAFTING AND REVIEWING

Bartholomew, K.M.	International Atomic Energy Agency
Böck, H.	Atominstitut, Vienna University of Technology, Austria
Hampel, G.	Consultant
Iunikova, A.	International Atomic Energy Agency
Marshall, F.	International Atomic Energy Agency
Miller, G.E.	University of California Irvine, United States of America
Nacir, B.	National Centre for Nuclear Energy, Sciences and Technology, Morocco
Rao, D.V.H.	International Atomic Energy Agency
Razvi, J.	General Atomics, United States of America
Ridikas, D.	International Atomic Energy Agency
Salvini, A.	University of Pavia, Italy
Villa, M.	Atominstitut, Vienna University of Technology, Austria
Wall, D.	Nuclear Radiation Center, Washington State University, United States of America

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