CURRENT STATUS AND FUTURE CHALLENGE OF TRR-1/M1 THAI RESEARCH REACTOR

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Abstract.

Since 1962, Thailand has been utilizing a nuclear research reactor for a wide variety of applications including radioisotope production, research and development (R&D) and educational training. The Thai Research Reactor-1/Modification 1 (TRR-1/M1), is located in Bangkhen Area next to Kasetsart University. TRR-1/M1 is an open pool type TRIGA-Mark III using light water as a coolant, moderator, reflector and shield with concrete biological shield and four neutron beam tubes. The maximum power of TRR-1/M1 is licensed at 1.3 MW. TRR-1/M1 uses two types of low enriched uranium TRIGA fuel elements; 8.5 wt.% uranium and 20 wt.% uranium. The major achievement for the TRR-1/M1 includes pool repainting and the seismic analysis. For pool repainting, the plan A project team was organized specially for this project and a detailed execution plan was developed. The project activities include removing foreign objects and highly activated materials from the pool section, cleaning, inspecting, re-painting the pool surface and testing for water leaks. Preparation of the repainting activities had begun 2 years in advance. Periodic maintenance of the reactor pool has been conducted by cleaning the pool surface and re-painting with epoxy coating. The maintenance of the TRR-1/M1 small-section pool was conducted in early 2012. All unexpected incidents could be well controlled and the project could be conducted within the scheduled time period. In addition, the radiation exposure to all personnel was much lower than target limit and there was no accident causing any injury during the project. Currently, the small-section pool is in use again and there is no leak found after the maintenance. After the Fukushima accident, International Atomic Energy Agency (IAEA) recommended to have seismic analysis for TRR-1/M1. The TRR-1/M1 seismic analytical results, under different critical combinations of dead load, live load and seismic load, indicate that the maximum stress that will develop in the beam and column is significantly lower than the member strength. This can be explained by the interaction of the reactor pool and its building that effectively shorten the overall structure period and reduces the member forces. It can be concluded that both the reactor pool and its building structure are safe from earthquake loading and consequently no strengthening measure is required for the structures under consideration. The future plan for the new research reactor is considered and resumed but the infrastructure, detailed plan and schedule is being revised and rescheduled.

1. INTRODUCTION

TRR-1/M1 is an open pool type TRIGA – Mark III using light water as a coolant, moderator, reflector and shield with concrete biological shield and four neutron beam tubes. TRR-1/M1 was converted from an MTR-type reactor (TRR-1) in 1975. The operation had been started in November 1977. The nominal power is 1.3 MW. The TRR-1/M1 core is located in the reactor pool and heat generated by fission process is cooled by natural circulation of the pool water which in turn is dissipated by means of heat exchanger and cooling tower arrangement. The TRR-1/M1 experimental facilities include four neutron beam tubes, a graphite thermal column, a rotary specimen rack, a pneumatic transfer system, and several in-core and out-core irradiation positions. The safety of the reactor have been analysed and evaluated to demonstrate that it can be operated safely under all conditions, and that the health and safety of the public have been assured even under the most severe accident conditions postulated for the reactor. The overall safety objective for the TRR-1/M1 is to protect individuals, society and the environment by establishing and maintaining an effective protection against radiological hazards under all conditions of operation. In order to ensure that this objective is met, the TRR-1/M1 has been evaluated, operated and maintained in accordance with the IAEA Safety Requirement No. NS-R-4 “Safety of Research Reactors”. This includes the requirement that the total radiological releases and doses from all reactor components be kept as low as reasonably achievable (ALARA), and well within the limits prescribed by the competent Thai regulatory body referring as Office of Atoms for Peace

(OAP). Specifically, the radiological target values for doses and releases enforced by the Thai regulatory body are equivalent with those recommended in the ICRP-60 guidelines.

1.1. Brief history

The first criticality in Thailand took place in October 1962. At the time, the Thailand Research Reactor-1 (or TRR-1) was initially put into operation. TRR-1 was a swimming pool type reactor with thermal power up to 1 MW. The reactor employed highly enriched uranium fuel and utilized light water as moderator and coolant. The reactor core was composed of MTR-type (plate) fuel elements and graphite reflector elements arranged in rows on a grid plate approximately 6 meter below the water surface. The fuel elements and the other major components and systems were designed and provided by the Curtiss-Wright Corporation of Princeton, New Jersey, USA. The cooling water was circulated through the core by natural convection of the power levels up to 1 MW.

Within the first few years after TRR-1's initial operation, Curtiss-Wright was no longer active in the field of nuclear reactors, so the company was no longer a supplier of components and fuel. In 1975, TRR-1 was shut-down for core conversion. The highly enriched uranium plate core was replaced by a low enriched rod-type fuel designed and marketed by General Atomics in their TRIGA reactors. At that time, the control systems and safety systems were also replaced so that TRR-1 became essentially a TRIGA reactor in most important respects. The newly installed reactor has been operating since November 1977. To remark this conversion, the reactor was renamed from TRR-1 to Thai Research Reactor-1/Modification 1 (TRR-1/M1).

Initially, the TRR-1/M1 core contained 8.5/20 fuel (wt.%/enrichment). Due to relatively heavy reactor use, the higher loaded TRIGA 20/20 fuel, which also contains erbium as a burnable poison was introduced into the core in 1980 (from Core Loading No. 2). Since the presently installed and operating reactor contains typical features of TRIGA research reactors, the design and operating characteristics and the safety considerations of this reactor are very similar to those of a typical TRIGA reactor.

1.2. Objective of the TRR-1/M1 reactor

The TRR-1/M1 is a research reactor and it is used for the following purposes:

- To produce radioisotopes for medical, industrial, and agricultural uses,
- To conduct beam experiments, neutron radiography, and prompt-gamma neutron activation analysis,
- To conduct applied research and technology development in the nuclear or other fields,
- To provide general training and study on the fundamental principles of the reactor and its operation, and
- To provide a neutron source for neutron activation analyses.

1.3. Description and site location of the TRR-1/M1 reactor

The TRR-1/M1 is located in Bangkok, the capital of Thailand. The reactor site is next to Kasetsart University and the total area of the site is about 13,000 m^2. The TRR-1/M1 core utilizes approximately 20% enriched uranium zirconium hydride (UZrH) fuel which is loaded into two types of fuel elements: 8.5 wt.% and 20 wt.% uranium. The TRR-1/M1 fuel element is clad with 304 stainless steel. The 20 wt.% fuel element also contains about 0.5 wt.%

erbium as burnable poison which is intended to extend the operation lifetime of TRIGA fuel and provides significant fraction of the prompt negative temperature coefficient for reactivity feedback.

The fuel elements are positioned in a grid plate forming hexagonal configuration. The TRR-1/M1 uses five control rods, i.e., a safety rod, a regulating rod, two shim rods and a transient rod. The regulating, shim and safety rods are sealed in 304 stainless steel tubes while the transient rod has aluminium clad. The TRR-1/M1 can be operated in manual or automatic modes. The reactor power levels can be varied up to 1.3 MW (thermal).

The core cooling is maintained by natural convection. A circulation coolant system provides a sufficient heat removal capacity for 1.3 MW thermal through a primary coolant system and transferred heat through heat exchangers, i.e., a shell and tube type and a plate type. The extracting pipe of the primary cooling system is installed at the reactor pool about 1 meter below the pool water surface. The extracted water coolant is passed through the heat exchangers prior to be fed back to the bottom of the reactor pool. The pool water in the primary cooling system is absolutely isolated from the coolant water in the secondary system.

The TRR-1/M1 experimental facilities include beam ports, horizontal thermal column with graphite blocks, vertical tube in thermal column, rotary rack specimen, high-speed pneumatic transfer system, and several in-core and out-core irradiation facilities. The beam port is the facility which transports the neutron from the reactor core to the irradiation zone outside the reactor pool. Most uses of the beam ports are for neutron radiography, neutron computed tomography and neutron scattering experiments. The other irradiation facilities are in-core and out-core irradiation facilities which are being used for radioisotope production, neutron activation analysis (NAA), gemstone irradiation for gemstone colorization and other experiments upon requests.

2. CURRENT TECHNICAL STATUS

2.1. Building and support infrastructure

Reactor structures are a set of components designed including the mechanical, neutronic and thermal-hydraulics functions associated with the reactor core. It should be noted that some of the reactor structures are also designed to facilitate the utilization of neutrons produced in the core. Figure 1 shows the perspective view of the TRR-1/M1 structures.

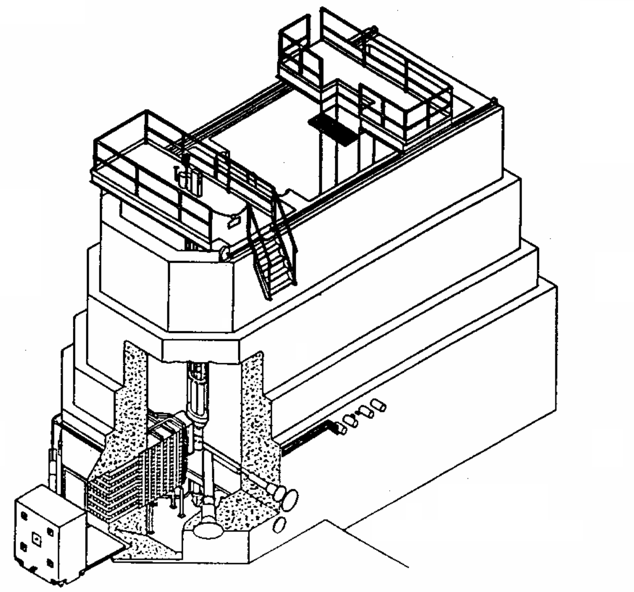

FIG. 1. Perspective view of TRR-1/M1 structures.

The thermal column provides a readily accessible field of thermalized neutron (approximately 0.025 eV) for experimental purposes. This field is provided by graphite blocks (approximately 150 × 150 × 240 cm^3 in size) extending from the reactor core into the biological shield. The reactor end of the thermal column is covered with a 7.6 cm lead shield to reduce the gamma irradiation in the thermal column. The shield end is lined with boral plates to keep the wall materials from becoming activated.

The neutron beam ports are connected to the void tank which conveys the neutrons from the reactor core. There are four beam ports installed in TRR-1/M1 extending from the core periphery through the water and concrete to the outer face of the shield structure to provide high-intensity beams of neutron and gamma radiation for a variety of experiments. There are two kinds, categorized by a diameter, of neutron beam ports, i.e., two of 20.0 cm (8 in.), and two of 15.0 cm (6 in.) in diameter. All beam ports, fabricated from commercially 6061 aluminium alloy, are of stepped design to reduce radiation streaming through the gap between the beam tube and shielding plugs. The embedded portions of all beam ports are stainless steel. Special shielding plugs reduce the radiation outside the concrete shield to safe level when the beam port is not in use. All beam ports terminate in blank aluminum plates which are bolted to stainless steel flange of the beam tube when the tube is not in use. A gasket between the plate and flange seals the tube to prevent loss of shielding water should the beam port develop a leak. Two beam ports (one 20 cm and one 15 cm) are located on the north side of the reactor pool while the other two beam ports are located on the south side of the reactor pool.

Several in-core irradiation facilities are installed in the reactor core to conduct experiments or to irradiate small samples in the core at the points of respectively high flux. Essentially, the in-core irradiation facility is a 3.81 cm OD aluminium tubes inserted straight down through the hole of the upper grid plate to bottom grid plate. A specimen in a capsule of a maximum diameter of 3.50 cm can be inserted in through these in-core irradiation facilities. The location of the in-core irradiation facility can be varied for each core loading.

The out-core irradiation facilities are attached to the reactor bridge. They consist of 3.81 cm (1-1/2 in) OD aluminium tubes that extend from the reactor bridge straight down to the flux region out-core. The end of the tubes are located in the relatively high flux region out-core. There are three categories of out-core facilities, one is the bare tube, another is cadmium covered tube (in order to obtain the neutron in epithermal and fast energy range) and the other is the graphite covered tube (in order to obtain high thermal-to-fast flux ratio). The location of each out-core irradiation facility can be moved depending on the utilization. Moreover, when required, additional out-core irradiation facilities can be easily installed. Figure 2 shows the drawings of the out-core irradiation facilities (namely A1, A4, CA2, CA3 and TA).

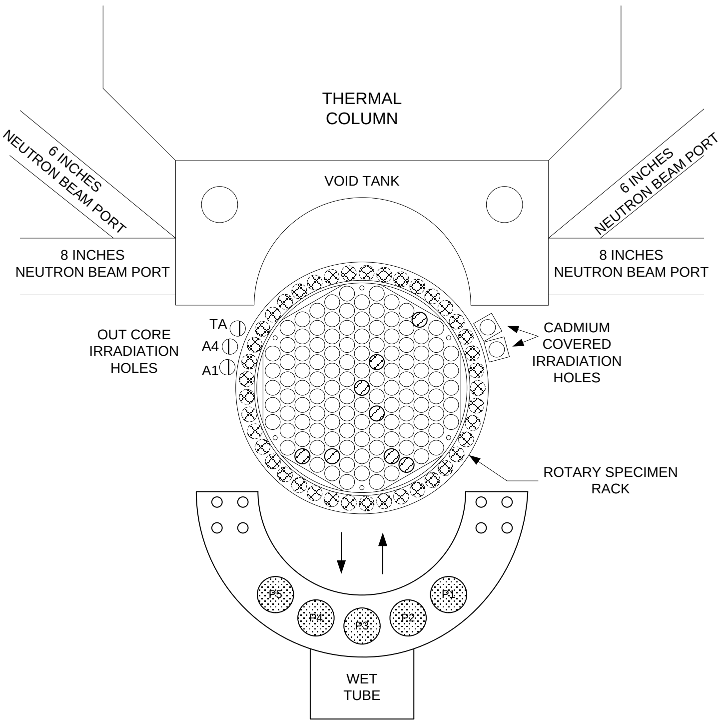

FIG. 2. Out-core irradiation facilities.

2.2. Fuel elements

The fuel of the TRR-1/M1 fuel element is solid, homogeneous mixture of uranium-zirconium hydride alloy with the uranium-to-zirconium atom ratio of 1.6 to 1.7. To facilitate hydriding, a small hole is drilled through the center of the active fuel section and a zirconium rod is inserted in this hole after hydriding is complete. The uranium in the uranium-zirconium hydride mixture is enriched to approximately 20% U-235. There are two types of fuel elements loaded in TRR-1/M1 core, i.e., 8.5% uranium by weight type and 20% uranium by weight type. It is noted that the fuel of the 20 wt% fuel element is a mixture of uranium-erbium-zirconium-hydride (UErZrH) alloy containing approximately 0.5 wt% erbium in order to extend the fuel lifetime. The number of fuel elements in the core varies for each core loading. The first core loading only loads 8.5 wt% fuel elements. The 20 wt% fuel elements have been used in the core starting from the core loading no. 2.

The physical dimensions of both 8.5 wt% type and 20 wt% type are identical. The fuel element is approximately 3.73 cm in diameter and 73.15 cm in overall length and the active part of the fuel element is 38.1 cm long [1]. The fuel element is placed in the reactor core through the upper grid plate and sits on the bottom grid plate in the reactor core. Figure 3 shows a perspective view of reactor core configuration.

The fuel used in TRIGA reactors was developed around the concept of inherent safety. A basic requirement for the fuel was that it must have a large prompt negative temperature coefficient of reactivity such that the fuel temperatures resulting from inadvertent large positive reactivity insertions would automatically cause the power excursion to terminate before any fuel damage resulted. Uranium zirconium hydride (UZrH), with or without erbium, was found to have this characteristics, as well as other excellent properties. These properties are summarized below. A detailed discussion of the properties of TRIGA UZrH fuel can be found in GA's literature [2].

The specific characteristics of the TRIGA fuels are described as followed. The characteristics include $ZrH_{1.6}$ being a single phase up to 750°C, low hydrogen equilibrium disassociation pressure at normal fuel temperature, high hydrogen retention, high heat capacity, low thermal expansion coefficient, relatively low reactivity in water, no significant damage or swelling due to irradiation effects, high fission product retention, very large negative temperature coefficient reactivity, high burnup possible by addition of burnable poison, high loading of uranium possible with insignificant change in fuel material properties.

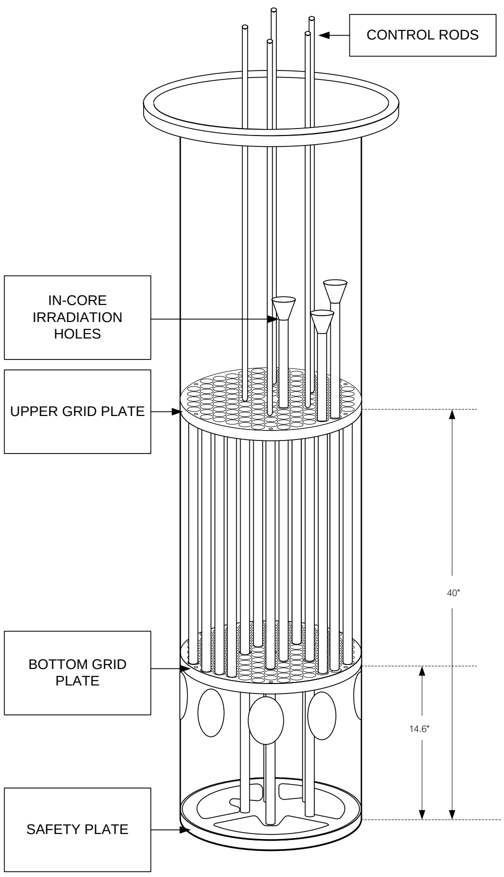

FIG. 3. Perspective view of reactor core configuration.

2.3. Core configuration

The calculation model of TRR-1/M1 was constructed using MCNPX [3]. The core geometry is described in three dimensional details to achieve the purpose of the specific calculation. Each grid position is modeled in the exact location in the grid plate. All components which fit into the core grid plate have the exact dimensional of their constituent materials. The geometry specification for each fuel element includes the central zirconium rod, the fuel meat and the cladding as well as the top and bottom graphite reflectors and end pieces (see Fig. 3). Control rods and fuel followers are located in the exact locations. The core configuration for the TRR-1/M1 using MCNPX model is shown in Fig. 4.

After a core loading has been operated for a certain period, the core excess reactivity becomes low and it needs reloading. New fuel elements are to be added into the core to essentially become a new core loading. The objective to design a new core loading is to keep the safety factors of the new core loading within the envelope of safety limits analyzed in this Safety Analysis Report (SAR). The strategy for designing a new core loading is to maximize the fuel utilization while minimizing the power peaking factors.

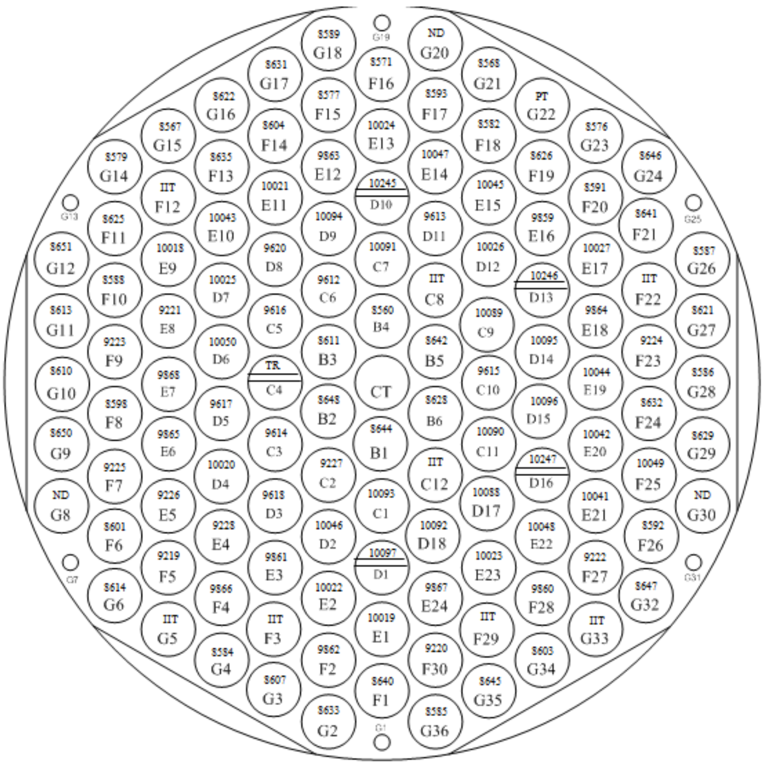

FIG. 4. Core configuration using MCNPX model.

2.4. Safety limit and condition

The maximum fuel temperature in a standard TRIGA fuel element shall not exceed 930ºC during steady-state operation. The maximum temperature in a standard TRIGA fuel element shall not exceed 1100ºC during pulse operation.

For steady-state operation of the reactor, the limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. A setting of 600ºC provides a safety margin at the point of the measurement of at least 330ºC for standard TRIGA fuel elements in any condition of operation. A part of the safety margin is used to account for the difference between the true and measured temperatures resulting from the actual location of the thermocouple. When the instrumented fuel element having the power density of 80% of the peak power density is at 600ºC, the temperature of the hottest fuel element shall not exceed 715ºC. Therefore, the hottest fuel element still does not exceed the safety limit of 930ºC with the safety margin of at least 200ºC.

For pulse operation of the reactor, the same limiting safety system setting applies. However, the temperature channel will have no effect on limiting the peak power generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, the temperature trip will act to limit the energy release after the pulse if the transient rod should not reinsert and the fuel temperature continues to increase.

2.5. Utilization and experimental program

The reactor provides a broad range of irradiation services for TRR-1/M1 researchers and other governmental and private sectors. The experimental programs conducted within the reactor facility include:

- Production of radioisotopes for nuclear medicine, agricultural, industrial and research applications;
- Neutron beam experiments such as Neutron Scattering experiments;
- Neutron Radiography;
- Neutron Activation Analysis;
- Experiments for training in nuclear technologies;
- Other experiments for research and development in the nuclear field;
- Training of reactor operation and education for nuclear technology.

2.6. Staff and training program

In each shift operation, a minimum of three reactor operation crew are assigned on duty, i.e., one shift supervisor and two console operators. Normally, the shift supervisor shall be a licensed reactor supervisor. On special cases, the reactor manager may assign a licensed senior reactor operator as a shift supervisor. Additional reactor operation crew may include reactor operator trainees who are allowed to manipulate the controls of the reactor under the direct supervision of a licensed reactor operator. In addition to the reactor operation crew, a health physicist is on duty during shift operation as well.

The reactor management ensures that adequate number of licensed reactor operation personnel is reserved in order to staff the reactor shift operation as planned without overloading the reactor operators. This means there shall be no console operator to operate TRR-1/M1 continuously more than eight hours.

During reactor startup, the shift supervisor and health physicist shall be present at the reactor facility. Also, whenever reactor maintenance is being performed or irradiations are being conducted, a health physicist on duty shall be readily available within the reactor site. For whole shift operation period, the shift supervisor and health physicist shall be readily available within the reactor site.

The training program for the operation and maintenance of TRR-1/M1 is established to develop and maintain qualified personnel for the operation and maintenance of the reactor. The content of the training is consistent with the OAP's Operator Training Standard [4] and the IAEA NS-G-4.5 Safety Guide "The Operating Organization and the Recruitment, Training and Qualification of Personnel for Research Reactors" [5] which includes the study of as-built and existing facilities, current procedures, and administrative rules and regulation.

The training program carries the trainee through documented stages of studies and on-the-job training. Certification is achieved after successful completion of the training and examinations. In addition to reactor operation and maintenance personnel, the reactor technical support personnel also undergo same training as described above. Additional training particular to each role in Reactor Management System (RMS) is provided based on his/her job functions and deficiency in critical areas to perform the job.

The training in radiation protection and emergency preparedness and response is mandatory to all positions working within the reactor facility. Facility users shall receive training in related facility utilization and radiation protection. Visitors and temporary workers are briefed with radiation protection measures before entering the controlled areas and are under close supervision of the responsible reactor staff.

The Reactor Safety Advisory Group (RSAG) is appointed by the Thailand Institute of Nuclear Technology (TINT) Executive Director and has the responsibility to review issues of operational significance and to advise the reactor manager regarding nuclear and radiation safety. The RSAG shall review nuclear safety associated with the operation and use of TRR-1/M1.

3. APPLICAION, UTILIZATION AND COLLABORATION

3.1. Application and utilization

The reactor provides a broad range of irradiation services for reactor users and researchers. The experimental program to be conducted with the reactor includes: Radioisotopes produced by TRR-1/M1 are I-131, Sm-153 and P-32. Some types of gemstones (such as topaz) can be irradiated by neutrons in order to enhance their value. Irradiations can be performed in the in-core and out-core facilities. For out-core irradiation facilities (normally using wet tubes and irradiation tubes in void tank), the typical irradiation time is approximately 6 months. After irradiation, the irradiated gemstones are stored in a storage area under water until the activity decays significantly.

Neutron beam ports are installed with utilization facilities including Neutron Scattering, Prompt Gamma Neutron Activation Analysis (PGNAA) and Neutron Radiography. Several kinds of samples are routinely irradiated for Neutron Activation Analysis (NAA). These include agricultural samples, mining samples, environmental samples etc. The amount of materials to be irradiated for NAA is usually small (milli-gram or gram range). Moreover, the NAA is performed in an out-core irradiation facility. Routine operations indicate that the change of reactivity caused by NAA samples is negligible.

3.2. Collaboration

TINT has collaboration with the local universities and research institutes abroad (EU, IAEA, Japan, Korea, USA) to conduct the study and the research. For educational utilization, training experiments are conducted for training purposes to reactor operator trainees, researchers and students. For nuclear engineering study, the training experiments are usually performed as routine calibrations or during normal operations such as reactor power calibration, control rod worth calibration, reactor power changes and etc. The experiments in this category are in compliance with Operating limits and Conditions (OC). For collaboration with other institutes for the research including reactivity insertion experiment, fuel experiment, thermal hydraulic, fuel burnup, radiotracer, NAA, PGNAA and other special experiments are conducted using the TRR-1/M1 reactor as well. The experimental facilities are under the administrative control of the reactor manager. All new experimental facilities and special experiments must be authorized by the reactor manager.

4. SUCCESS STORY AND MAJOR ACHIEVEMENTS FOR TRR-1/M1

4.1. Pool repainting

4.1.1. Brief Description of the Reactor Pool

The TRR-1/M1 pool is located in the center of the reactor building. Essentially, the pool is rectangular in shape and is divided into two sections by means of a removable watertight aluminum gate. The pool wall consists of 2 concrete layers of different densities; the inner layer made of conventional concrete and the outer layer made of high density concrete for biological shielding. The reactor core is hung by the reactor bridge which can be moved to either “small section” or “large-section”. When the watertight gate is put in place, the section without the reactor core can be drained for in-pool maintenance. The reactor core is usually positioned in the small-section which is the main experimental area. There are a number of fixed irradiation facilities in this area. These include thermal column, void tank and neutron beam ports as shown in Fig 5. When the reactor core is positioned next these facilities for utilization of the facilities, the materials of these facilities are constantly irradiated which produces highly radioactive materials as a consequence. However, the material of these facilities is mainly high purity aluminum alloy which is not activated into long half-life radio-nuclides.

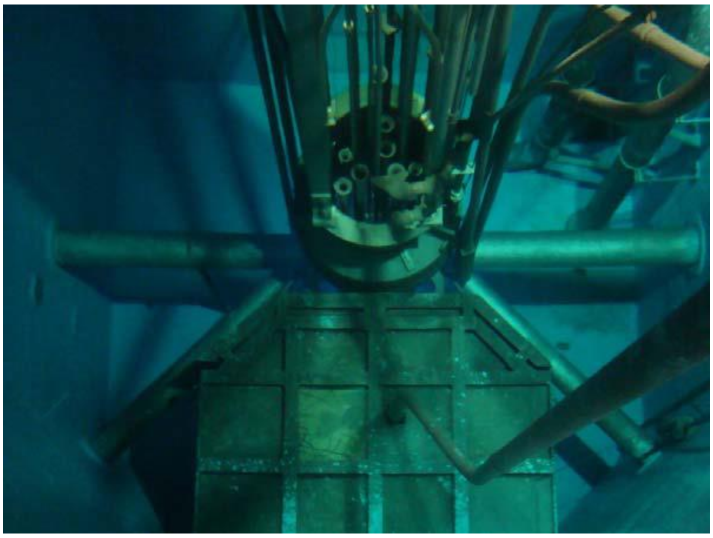

FIG. 5. Irradiation facilities in the small-section area.

4.1.2. Radiological Characteristics of the Small Pool

In order to prepare for the maintenance, the reactor core was repositioned to operate in the large-section pool (away from the maintenance area) approximately 1.5 years prior to the maintenance activities. This period allowed for short half-life radio-nuclides to substantially decay. The radiological characteristic of the area was assessed by measuring the dose rate in-pool under water. The measurement used high-dose GM detector (measurement range up to 99 R/hr) connected to a long cable. Contact dose rates at various points in the area were measured as report in Fig. 6.

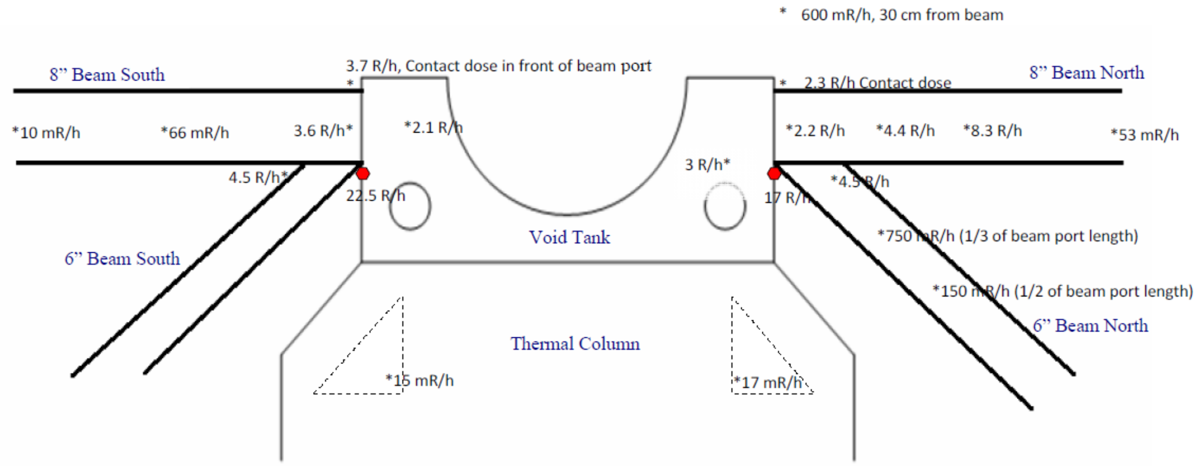

FIG. 6. Radiation dose rate at various positions in the small-section area.

From the radiation survey, three hot spots were identified. The first two areas were on the left and right side of the void tank. It was found then that the high dose rate was caused by 4 sets of bolts and nuts supporting the void tank. These bolts and nuts were made of stainless steel as supposed to aluminum alloy. However, it was also found that one of the washers was corroded suggesting that this particular washer was made of galvanized steel rather than stainless steel. The last hot spot was near the middle of the 8" north neutron beam. It was believed that there was also stainless steel acting as a structure inside the beam. Spectrum analysis of the radiation detected Co-60 as the main nuclide in these hot spots.

4.1.3. Area Classification

In order to prevent spread of contamination, the working area was classified into three zones (i.e., Zone I, Zone II and Zone III). Zone I was the zone where contamination was highly possible. The area in this zone included in-pool and loading/unloading area. A high level of contamination control was enforced for this zone. Protective cloth and digital dosimeters were required to enter this zone. Contamination monitoring was conducted for all personnel and equipment in and out of this zone. Zone II was the buffered zone where contamination was possible but limited. This zone included the area surrounding the Zone I. Contamination control was also enforced in this area including periodic survey of the area, protective shoes and gloves and contamination check of personnel by a hands and shoes monitor. Lastly, Zone III was the clean zone where there was no expected contamination. Some degree of contamination control was conducted to ensure contamination-free area such as periodic survey of the area.

4.1.4. Draining Pool Water

The next step was to drain the pool water until the water level remained 1 meter above the void tank. This step was done in preparation for removing the activated bolts and nuts. The pool water was released in a step-wise manner to the retention pond of the radioactive waste treatment facility. Before each release step, water was sampled and measured for radio-nuclide concentration in the gamma spectroscopy laboratory. In addition, radiation dose survey was performed including periodic air sampling.

4.1.5. Reducing Radiation

When the water level was approximately 1 meter above the void tank, the next step was to remove the activated bolts and nuts from the in-pool area. The water thickness of 1 meter was selected as the optimal point for compromising between the shielding efficiency and the difficulty in removing the items from far distance. A special tool set to disassemble the bolts and nuts from the void tank were prepared. The tool set was essentially pneumatic drill with extension to loosen the bolt while holding the nut, a dip net to catch falling objects and an extended tongs to remove the bolt. The task of removing each bolt and nut set took two workers. A new pin made of high purity aluminum alloy was prepared and used to replace the original bolt. All four sets of the bolts and nuts were successfully removed with minor difficulties. The other hot spot which was on the middle of the north beam was handled by covering with leaded sheet. A supporting structure covering the beam was prepared and the leaded sheet was placed on the structure in order to reduce radiation from the beam.

4.1.6. Coating

The recoating work was conducted after the major radiation sources were removed. The coating of the upper portion of the pool (i.e., above 3-meter area) was conducted first with water still covers the irradiation facility. By doing this, the exposure to the workers was minimized while the workers were coating the majority of the area. Three layers of epoxy paint including primer were coated in the project according to the paint company recommendation. When the coating of the upper portion of the pool was completed, the pool water was drained completely. A radiation survey into area was performed. Figure 7 shows the radiation dose rate map of the reactor pool when completely drained. The radiation dose rate map showed that the area in the line of sight from the north beam had relatively higher dose rate than other area. Workers to perform the coating were advised to avoid or spend minimum time in the area. Strict radiation control such as time limit, equipping all workers with digital dosimeters for real-time monitoring, daily exposure record, contamination monitoring was enforced. The coating activities were finally completed according to the planned schedule. The exposure to all personnel in the project was found to much less than the target limit – the maximum exposure was 1.1 mSv and the collective dose for the whole project was 8.4 man-mSv.

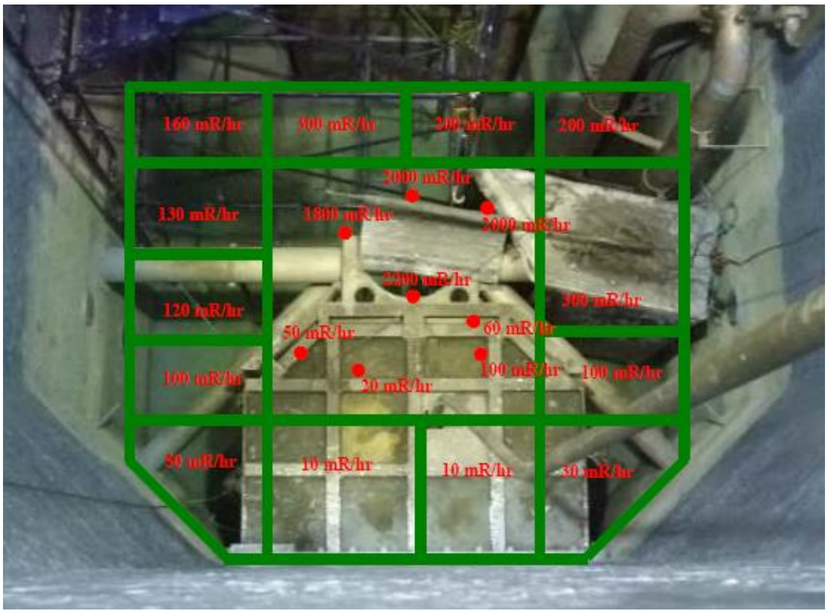

FIG. 7. Radiation dose rate mapping at the reactor pool when completely drained.

4.1.7. Waste Management

During the course of the project, radioactive wastes were generated. Attempts were made to reduce the radioactive waste by segregating waste types. Essentially, the project produced approximately 11 bags of low level wastes mainly from protective cloth. The activated bolts and nuts were transferred to the waste management center for final storage.

4.1.8. Lessons learned

The major lessons learned after completion of the repaint project as the following:

- Thorough planning and practice in advance is critical to the success of the project.
- Engineering drawing of the facility is a critical document for project planning and execution and shall be well maintained.
- Practices and simulations shall be conducted as closely to the actual environment as possible.
- Using photographic images is an excellent way to communicate among members and outside workers.
- The ability to have visual image in real time is important to understand the problem on the field. This ability shall be considered in the planning.
- Reusing some of materials may reduce the amount of unnecessary radioactive wastes.

4.2. Seismic analysis

4.2.1. Background for Seismic Analysis

The reactor pool and the building of the TINT have been in use for more than 50 years. During those years, the seismic response of structures was not as widely investigated as compared to today where useful information on earthquake engineering was widely published and easily accessed. The critical active fault that can have significant effect on building in Bangkok is in the Karnchanaburi province which has the potential to create an earthquake of

7.0 to 7.5 Richter magnitude. Moreover, the subsurface condition of Bangkok area is of soft soil type which can result in the amplification of the seismic wave. Thus, it is necessary to study the seismic response of the reactor pool and its building. Furthermore, after the Fukushima accident, the International Atomic Energy Agency (IAEA) has suggested that seismic analysis of reactor pool and its building can provide useful preliminary data for automatic shutdown system of the reactor upon the detection of an earthquake. Therefore, it is crucial to study the effect of earthquake loading on the reactor pool and its building.

4.2.2. Seismic Hazard Map and Response Spectrum

To develop the design response spectrum, the dynamic characteristics of soil layers (i.e. shear wave velocity) were investigated. 28 seismic time-history analyses were used to construct the response spectrum [6]. It was found that soil profile at the studied location can significantly amplify the structural response, in particular for structures with period at or lower than 1 second. The maximum value of the response spectrum, occurring at the structural period of 0.7-0.8 second, was estimated to be 0.27 g and 0.37 g for median response and 84th percentile response, respectively (Fig. 8).

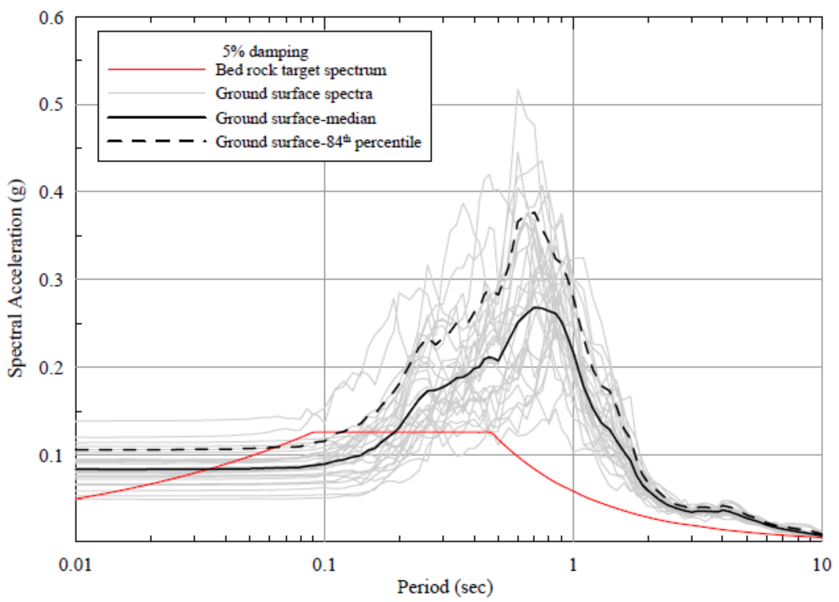

FIG. 8. Response Spectra.

4.2.3. Structural Analysis

Non-destructive tests, using rebound hammer and ultrasonic pulse velocity measurement, were performed on the reactor pool and the building to obtain an upper bound estimate of the concrete compressive strength. Even though the tests indicated an in-situ compressive strength ranging from 30-50 MPa, the value used in the analysis was conservatively taken as 20.5 MPa, the same value specified in the original design drawing. The structures were analyzed using a combination of equivalent static and dynamic procedures. The dynamic procedure used in this study involved time-history analysis of spring-mass models capable of simulating the behavior of contained water in the reactor pool, as shown in Fig. 9. It was found that the maximum tensile stress of the concrete reactor pool obtained from the analyses was lower than the allowable tensile strength. Therefore, the reactor pool can withstand the seismic-induced loading that might occur [7].

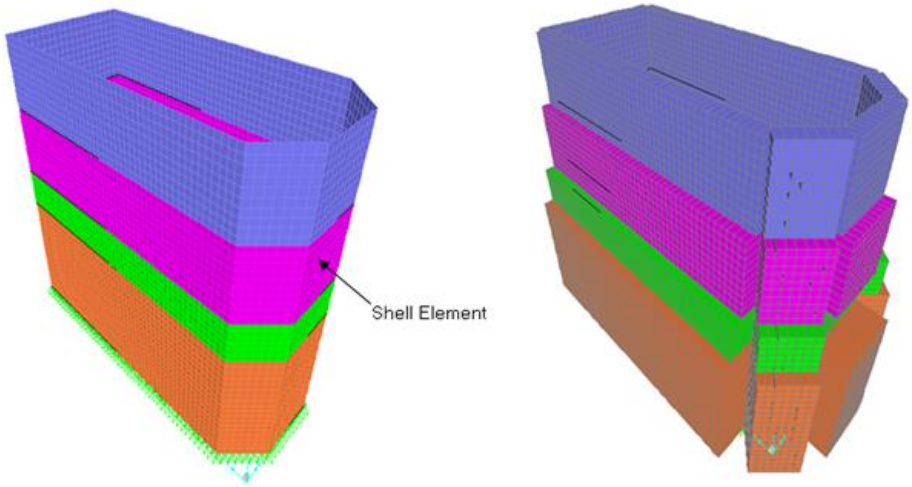

FIG. 9. TRR-1/M1 Reactor pool model.

To capture the interaction of the reactor pool and its building, a model comprising of the reactor pool (with contained water) attached to the building with flexible foundation was developed [7]. The finite element model is shown in Fig. 10.

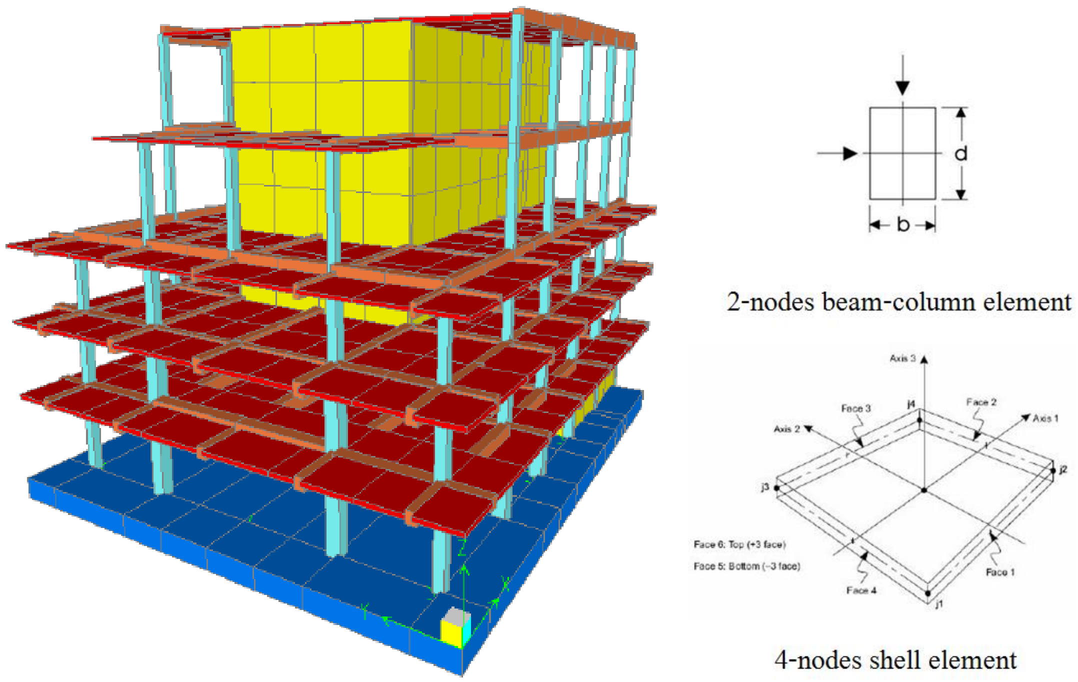

FIG. 10. Building and reactor pool interaction model for TRR-1/M1.

4.2.4. Seismic Analysis Results

The analytical results, under different critical combinations of dead load, live load and seismic load, indicate that the maximum stress that will develop in the beam and column is significantly lower than the member strength. This can be explained by the interaction of the reactor pool and its building, which effectively shorten the overall structure period and reduces the member forces [7]. It can be concluded that both the reactor pool and its building structure are safe from earthquake loading and consequently no strengthening measure is required for the structures under consideration.

5. FUTURE PROSPECTS

Since Thailand plan to have the nuclear power plant according to the Thailand Power Development Plan, PDP 2007 prepared by EGAT and approved by the National Energy Policy Council (NEPC) on June 4, 2007 and by the Cabinet on June 19, 2007 included recommendations to have new nuclear plants available for commercial operation. The PDP recognizes that nuclear generation is a “promising option” to respond to the future power demand growth in Thailand. TINT will support the nuclear power program in aspect of the educational and training which provides for the required education and training of staffing for those facilities and to provide an information program for public understanding and acceptance of a Thailand nuclear electrical generation capability.

TINT will collaborate with the local educational institutions (Chulalongkorn, Chiangmai, Kasetsart, Mahidol and Prince of Songkhla Universities, etc.) for both professional and vocational education and training to provide the support in nuclear engineering and nuclear science, radiology, radiation dosimetry, nuclear physics, etc. TRR-1/M1 can be utilized as an important tool for teaching future staff of the nuclear power plants and can provide the basic reactor experiments, radiation safety laboratory and research facilities for them. TINT avails its laboratory equipment and research facilities to complement the nuclear energy education of national universities. With the existing resources at the universities in focus and at TINT, expansion of the national nuclear education to encompass power generation application is in great potential.

The expansion of study and training programs to encompass power generation can be realized with respect to the potential resources at TINT and the universities. TINT will establish the educational and training program using research reactor to serve the country needs of manpower as following;

(1) Education and training on the applications of reactor theory, reactor engineer, nuclear radiation technology will be available, adequate and are of acceptable international standard.

(2) Research reactor (TRR-1/M1) can be utilized in areas of reactor technology. Remedy can be taken by expanding study and training program coverage. TINT can play a major role in establishing power reactor operation training programs.

(3) Establishing the initial training of the safety culture.

(4) Partnering with the universities, vocational schools, unions and other educational groups to assist in the program expansion and in the recruitment of staff and students.

(5) Enhancing public awareness programs so that the public is knowledgeable of the needs and benefits of the nuclear power program.

6. CONCLUSION

Currently, there are two major challenges for TRR-1/M1 management, namely, fuel supply situation and ageing components. The fuel supply situation is caused by the shutdown of the only TRIGA fuel fabrication plant for safety upgrade. The situation is still unclear when the plant would be operational again. Due to this fuel supply situation, the operation of

TRR-1/M1 has been decreased in order to reserve fuel consumption. The operation of TRR-1/M1 is reduced from 46 hrs per week to 26 hrs per 2 weeks and the I-131 production using TRR-1/M1 is currently stopped. The second major challenge is ageing components due to the old systems, structures and components which are approaching their lifetimes. A number of systems, structures and components are planned for upgraded in the next coming few years including electrical system, instrumentation and control system and secondary cooling system. Finally, the new research reactor construction plan is resumed but the detailed plan and schedule is being established.

REFERENCES

[1] GENERAL ATOMICS COMPANY, "Standard TRIGA Mark III: Safety Analysis Report", San Diego, USA (1975).

[2] SIMNAD, M.T., "The U-ZRHx Alloy: Its Properties and Use in TRIGA Fuel", Nuclear Engineering and Design, **64** (1981) 403-422.

[3] PELOWITZ, D.B., MCNPX User Manual Version 2.6.0, Los Alamos National Laboratory (April 2008).

[4] OFFICE OF ATOMS FOR PEACE (OAP's), Operator Training Standard, 2009

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, The Operating Organization and the Recruitment, Training and Qualification of Personnel for Research Reactors, IAEA Safety Guide No. NS-G-4.5, IAEA, Vienna (2008).

[6] ABRAHAMSON, N.A., "Non-Stationary Spectral Matching" Presented at 87th Annual Meeting of the Seismological Society of America (1992).

[7] REONGRATSAMEE, A, "Evaluation of Seismic Resistance for TRR-1/M1 Structure and Pool" (June 2010).

PUSPATI TRIGA REACTOR

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Abstract.
The PUSPATI TRIGA Reactor is the only research reactor in Malaysia. This 1 MW TRIGA Mk II reactor first reached criticality on 28 June 1982 and is located at the Malaysian Nuclear Agency premise in Bangi, Malaysia. This reactor has been mainly utilised for research, training and education and isotope production. Over the years several systems have been refurbished or modernised to overcome ageing and obsolescence problems. Major achievements and milestones will also be elaborated in this paper.

1. BRIEF HISTORY

June 28, 1982 marks a significant milestone for Malaysia. It was the day that the first and only nuclear reactor in the country, PUSPATI TRIGA Reactor or RTP, went critical at 5pm, with 66 fuel elements. The 1 MW TRIGA Mark II reactor from General Atomics, was purchased through a tripartite agreement signed on 22 September 1980 between the Governments of Malaysia and United States of America and the International Atomic Energy Agency (IAEA).

RTP is located at Malaysian Nuclear Agency (Nuclear Malaysia)'s complex in Bangi, Selangor, surrounded by secondary forest (Fig. 1). Other facilities within the complex are the Radioactive Waste Treatment Centre, Secondary Standards Dosimetry Laboratory, Isotope Production Facilities, Non-Destructive Testing Facilities, Instrumentation and Control Laboratory, Advanced Materials Laboratory and Engineering Workshops.

FIG. 1. Malaysian Nuclear Agency Bangi Complex.

The reactor has been operating continuously since 1982 and Fig. 2 shows the operating history. During1987-1990, the reactor was operated for medical isotope production trials, hence there was an increase of operating hours reaching a maximum of about 2700 hours in 1990. Unfortunately the trials showed that the specific activity of molybdenum-99 through the (n,γ) route was low and unsuitable for medical use and the trials ceased in 1991. The reactor was then mainly operated for neutron activation analysis, environmental tracer production as well as education and training. Currently, the operating hours is between 400-600 hours per year depending on user requests.

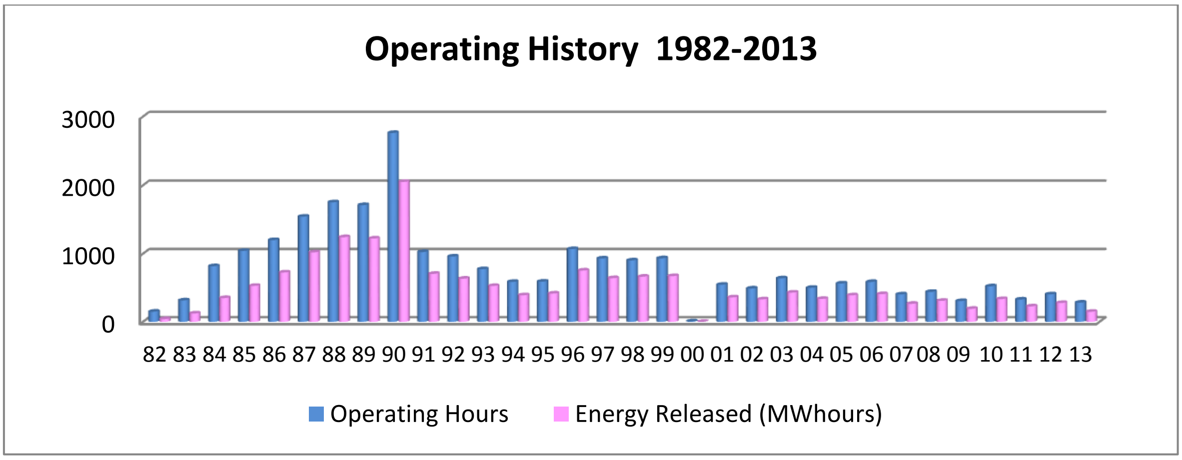

FIG. 2. Operating History of PUSPATI TRIGA Reactor from 1982-2013.

Between 1984 and 1996, efforts went towards developing new experimental facilities at the beamports to strengthen neutron science research. The neutron radiography facility (NuR) and small angle neutron scattering facility (SANS) was designed, installed and commissioned by local researchers. From the year 2000 onwards, the reactor systems and components showed signs of ageing and had to be replaced, refurbished or modernized [1]. Figure 3 shows a brief history of activities from 1982 until 2013. The reactor was in extended shutdown as follows:

2000 Active Ventilation System (Replacement)
2010 Primary Cooling System (Upgrade)
2013 Digital Reactor Control Console (Modernisation).

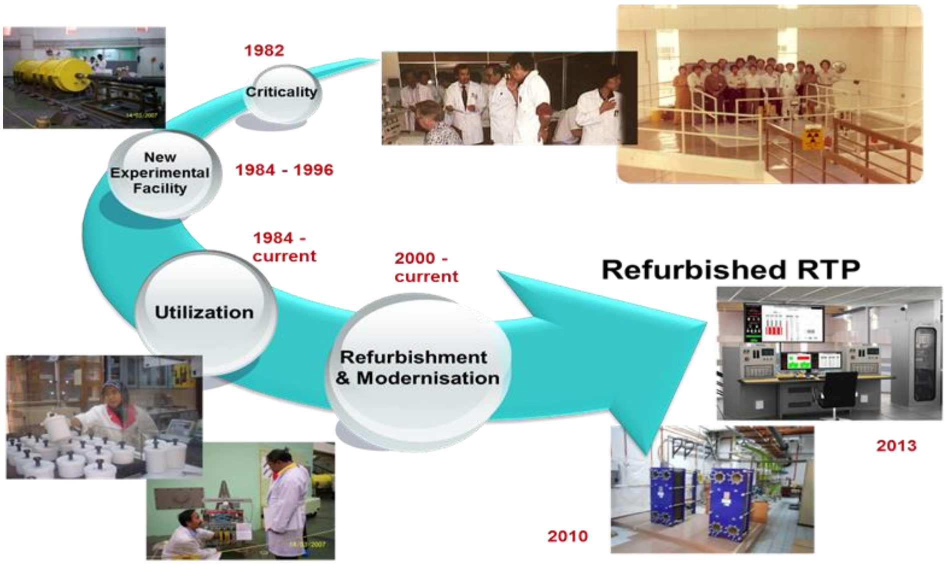

FIG. 3. Brief History of PUSPATI TRIGA Reactor.

2. CURRENT TECHNICAL STATUS

The RTP is typical pool-type reactor, where the reactor core, located 5 meters below the water surface in an aluminium tank, is visible from the reactor top [2]. Standard TRIGA fuel elements composed of uranium homogeneous mixed with zirconium hydride and clad in stainless steel are arranged in six circular rings in the core. Surrounding the aluminium reactor tank is a 2.5 m thick high density concrete. Vertical and horizontal cut-out views of RTP are shown in Fig. 4 while a view of the reactor and core are shown in Fig. 5.

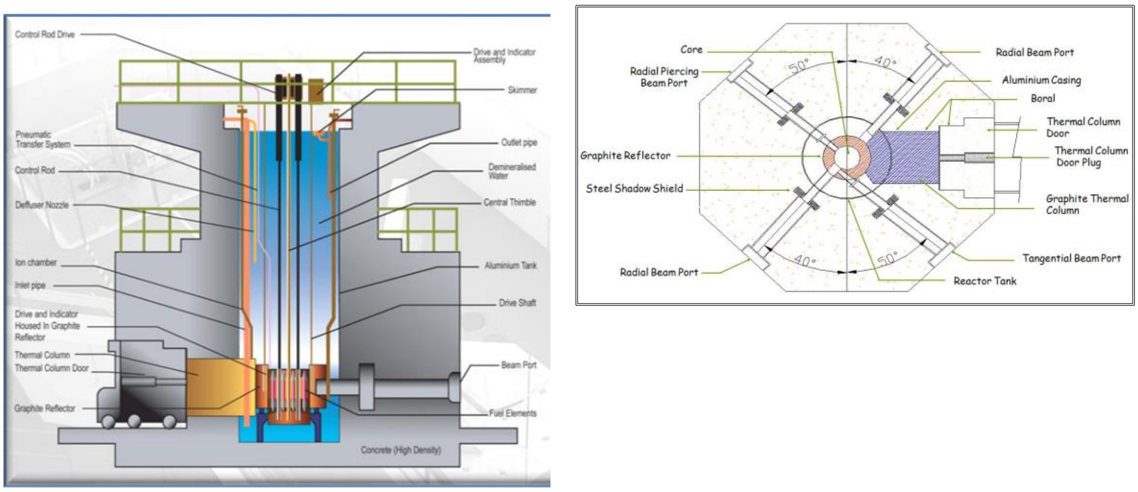

FIG. 4. Vertical and Horizontal Cross-section of PUSPATI TRIGA Reactor.

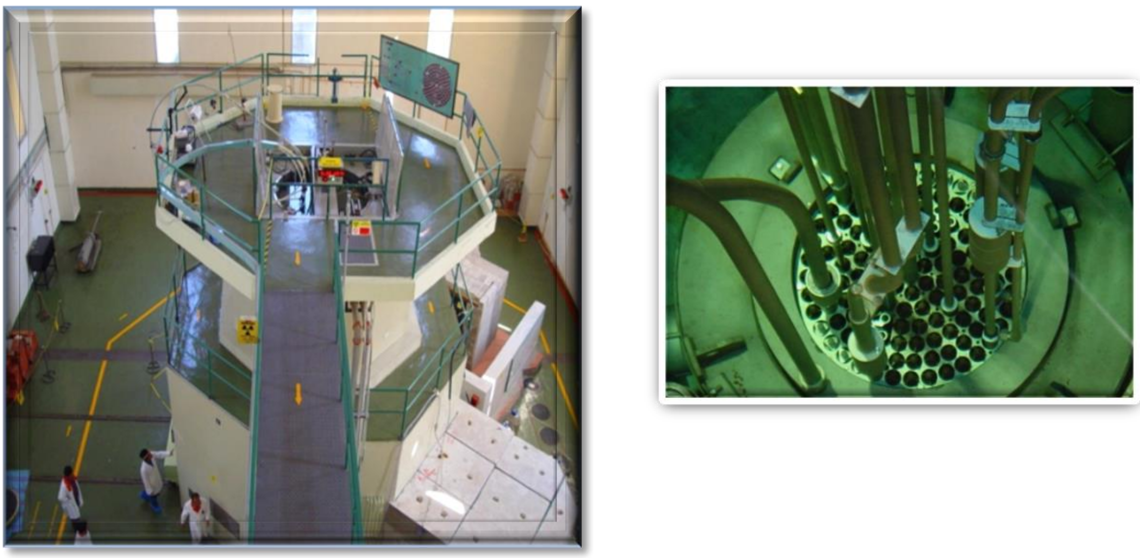

FIG. 5. View of PUSPATI TRIGA Reactor and Core.

Currently, the reactor is fuelled with LEU TRIGA fuel elements with three different weight percentages i.e. 8.5%, 12%, and 20% uranium. The maximum steady state power remains at 1 MW but the pulsing capability for reactor has been disabled after the installation of the new digital control console. Important reactor parameters are listed in Table 1.

TABLE 1. REACTOR PARAMETERS

Reactor Model	**TRIGA Mark II Pool Type Reactor**
Maximum Power	1 MW Thermal (Steady State)
Reactor Core	Cylindrical with 6 circular rings and 127 holes/positions
Nuclear Fuel	$U\text{-}ZrH_{1.6}$ Standard TRIGA Fuel
Fuel Cladding	Stainless Steel 304
Enrichment	19.9% U-235
Uranium Content	8.5%, 12%, 20%
Control Rods	Boron Carbide (3 Fuel Follower, 1 Air follower)
Moderator / Coolant	Demineralized Light Water
Reflector	Graphite
Typical Neutron Flux	1×10^{12} $n\cdot cm^{-2}\cdot s^{-1}$ @ Rotary Rack
Maximum Neutron Flux	1×10^{13} $n\cdot cm^{-2}\cdot s^{-1}$ @ Central Thimble

Irradiation facilities that were initially built together with the reactor are the central thimble, pneumatic transfer system, rotary rack, thermal column and four beam-ports (three radial and one tangential). Other experimental facilities installed later are: neutron radiography (NuR) and small angle neutron system (SANS) at the beam-ports and delayed neutron activation (DNA) terminus and dry tube terminus at in-core locations. The location of the irradiation and experimental facilities are listed in Table 2 while Fig. 6 shows the SANS facility.

TABLE 2. IRRADIATION / EXPERIMENTAL FACILITIES

Irradiation/Experimental Facility	Location
Central Thimble	A-1 (Centre of Core)
Dry Tube	F-11
Pneumatic Transfer System (PTS)	G-20
Delayed Neutron Activation System (DNA)	F-29 Bare Terminus G-1 Cadmium-lined Terminus
Neutron Radiography (NuR)	Beamport #3 (radial)
Small Angle Neutron Scattering (SANS)	Beamport #4 (radial piercing) (Beam size at specimen12-15 mm)
Thermal Column	In between Beamport #1 and #2
Unused Beamports	Beamport #1 (radial) Beamport #2 (tangential)

FIG. 6. Small Angle Neutron Facility.

3. APPLICATION AND UTILIZATION

At the PUSPATI TRIGA Reactor, neutron activation analysis (NAA) accounts for more than 90% of reactor utilization with the remaining for material science research, education and training, radioisotope production for environmental studies and public awareness programme. More than 3000 samples are irradiated for NAA annually as there is a high demand for this service. Neutron radiography and imaging is used for studying archeological artifacts/objects and condensed matter while the small angle neutron scattering (SANS) facility is used to study various material.

In recent years, several universities have introduced nuclear technology and engineering into their undergraduate programmes. Students are taken on a facility familiarization tour and participate in reactor experiments. Students can also conduct short term projects based on experiments at the reactor as part of their undergraduate or postgraduate requirements.

PUSPATI TRIGA Reactor operators have to undergo a certification programme based on the national standard. This programme consists of lectures, facility walkthrough and console operation and training is conducted by Nuclear Malaysia while the examination is conducted by the regulator.

The reactor is also open to visitors and around 2000 visitors are received each year. This is part of our efforts in communicating and dispelling the myths and misconceptions on nuclear reactor and nuclear energy.

4. SUCCESS STORIES AND MAJOR ACHIEVEMENTS

One of the great successes is the safe operation of the reactor without any untoward incident since 1982. This can be attributed to the competence and dedication of the operators, engineers and scientist in all aspects of reactor operation. The manufacturer's recommendation for maintenance and inspection are adhered to and new safety features and practices are added to enhance safety and security. Any modification or experiments which have an impact on reactor safety goes through a thorough evaluation by the in-house safety committee and the regulator.

The development of experimental facilities for neutron radiography and imaging, small angle neutron scattering and delayed neutron activation by local researchers is another achievement. These facilities are currently being upgraded to enhance performance.

Ageing and obsolescence often plaque aged research reactor but fortunately funding was secured to combat these problems. The upgrading of the primary cooling system in 2010 saw the change of the old tube and shell type heat exchanger to a more efficient plate-type heat exchanger and the installation of a SCADA for remote control. The most recent project was the modernisation of the reactor control console from an analogue to a digital system (Fig. 7). This project was implemented in 2013 in collaboration with the Korean Atomic Energy Research Institute (KAERI) and local engineers had the opportunity to be involved in the design, installation and commissioning phases. The technology transfer during this project was effective in developing the capability of Malaysian engineers in reactor instrumentation and control system.

FIG. 7. New Primary Cooling System and Digital Control Console.

In the academic field, over 60 students received their postgraduate degree based on their research using the reactor. Hundreds more conducted short term projects and industrial training at the RTP.

Amid the growing interest in future deployment of nuclear energy in Malaysia, expertise on reactor technology and science can play a pivotal role in assisting the policy makers in their decision making. Reactor personnel have the relevant expertise and have already been called upon to assist the government in several studies.

Malaysia is one of the countries that give equal opportunity to qualified women candidates to lead various areas of nuclear and reactor science and technology. Currently, the reactor operations and maintenance manager is a woman. In addition, woman scientists and engineers also lead in reactor technology assessment, neutronics modelling and calculation, probabilistic and deterministic safety assessment (PSA and DSA) as well as reactor decontamination and decommissioning (D&D).

5. FUTURE PROSPECTS

The development and upgrading of reactor systems, structures and components will continue as part of the safety enhancement and ageing management programme. In 2014-2015 projects implemented or planned are irradiated fuel transfer cask, spent fuel pool, upgrading of secondary cooling system, active ventilation system and electrical system. These will greatly enhance the reactor performance and safety.

New experimental facilities are also being developed or planned. A prompt gamma neutron activation (PGNAA) system and neutron diffractometer will be installed in the next few years and will diversify the research opportunities. A small angle X ray scattering system (SAXS) is also being developed as a complimentary technique to SANS at the reactor.

Education and training in nuclear and reactor science and engineering at the reactor will continue to be strengthened as more universities offer academic programmes in these areas. Engagement with prospective universities will also be conducted to promote research and development in neutron science since the reactor will be equipped with more experimental facilities in the near future.

Currently, source material for medical radioisotopes are imported. The production of medical radioisotopes in the future will depend on the reactor power upgrading. A study on the neutronics and themalhydraulics requirements for power upgrading is being carried out. Since the reactor power upgrading will require a large investment, the economic benefits of such a project will have to be ascertained before funding can be secured.

6. CONCLUSIONS

The PUSPATI TRIGA Reactor has been safely operated and utilized since 1982. Various projects are currently being implemented or planned to enhance the reactor safety and performance as well as developing new experimental facilities. Education and training in nuclear and reactor engineering will increase in the future due to new academic programmes introduced while promotion of neutron science research will be pursued. However, NAA will continue to dominate since there is a high demand for this service. The reactor will continue to be part of the public awareness programme and will play a vital role if nuclear power is deployed in Malaysia.

REFERENCES

[1] MASOOD, Z., ZAKARIA, M.F., HUSSIN, I.A., Refurbishment and Modernisation of PUSPATI TRIGA Reactor: Lessons Learnt, Proc. IGORR 2013 and IAEA International Workshop on Research Reactor Aeging, Modernisation and Refurbishment, Daejeon, Korea (2013).

[2] MALAYSIA NUCLEAR AGENCY, PUSPATI TRIGA Reactor Safety Analysis Report (2014).

BAEC TRIGA RESEARCH REACTOR

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1. INTRODUCTION

The 3 MW TRIGA Mark-II research reactor of the Bangladesh Atomic Energy Commission (BAEC) achieved its first criticality on 14 September 1986. The reactor has been used for manpower training, radioisotope production (Iodine-131), and various R&D activities in the field of Neutron Activation Analysis (NAA), Neutron Radiography (NR), and Neutron Scattering. The reactor has been operated successfully since its commissioning with the exception of a few incidents. Full power reactor operations remained suspended from 1997–2001 when a corrosion leakage problem in the N-16 decay tank threatened the integrity of the primary cooling loop. The new tank was installed in 2001 and some modification and upgrades were carried out in the reactor cooling system such as that the operational safety of the reactor could be strengthened. The cooling system upgrade mainly included replacement of the fouled shell and tube-type heat exchanger by a new plate-type one, modification of the cooling system piping layout, installation of isolation valves, installation of a chemical injection system for the secondary cooling system, modification of the Emergency Core Cooling System (ECCS), etc. After successful completion of all these modifications, the reactor was made operational again at full power of 3 MW in August 2001 [1].

In 2009, i.e., almost 23 years after the commissioning of the BAEC TRIGA Research Reactor (BTRR), it became necessary to remove the beam port (BP) plugs from inside the radial beam port -1 (RBP-1) so as to facilitate installation of the collimator of the newly procured high resolution powder diffractometer (HRPD). While doing this, the graphite plug which was the innermost part of the beam port shielding arrangement was found to get stuck very tightly to the aluminum part of the BP-1. The stuck graphite plug was removed by cutting it with the help of locally designed and fabricated special hand tools. The broken graphite plug was removed on Sept. 2009. After removal of the broken graphite plug, the RBP-1 was found leaking water at a rate of about 500 ml/day on Sept. 2009. It was then decided to install the HRPD at RBP-2. Leakage of water through RBP-1 was stopped temporarily by installing a silicon rubber band at the outer surface of the damaged part of the BP. For the semi-permanent solution of the leakage problem a split type encirclement clamp was designed and fabricated locally for installing at the damaged part of the BP [2].

BAEC decided to replace the previously used Analog Control Console (ACC) system with Digital Control Console (DCC) system by an ADP project. The installation at 3 MW TRIGA MK-II research reactor started on July 2011 and commissioning of the DCC system was completed on June 2, 2012. This DCC system was supplied and installed by General Atomics (GA), USA. After the installation of the digital console system, some nuclear safety parameters were measured to ensure the safe operation of the reactor [3]. The measured safety parameters were compared with the previous measured (analog system) values for the validation of the digital control system of the reactor [4, 5].

2. TECHNICAL STATUS

2.1. Reactor

The BAEC TRIGA Mark-II reactor is a light water cooled, cylindrical shaped pool type research reactor, which uses uranium-zirconium hydride fuel elements in a circular grid array. The array also contains graphite dummy elements, which serves to reflect neutrons back into the core. The core is situated near the bottom of water filled tank and the tank is surrounded by a concrete bio-shield, which acts as a radiation shield and structural support. The reactor is housed in a hall of 23.5×20.12 m^2 having a height of 17.4 m. The reactor is licensed by the BAEC to operate at a maximum steady state power of 3 MW (thermal) and can also be pulsed up to a peak power of about 852 MW with a maximum reactivity insertion of up to $ 2.00 having a half–maximum pulse width of nearly 18.6 milliseconds [6]. Some of the design parameters of the reactor are given in Table 1.

TABLE 1. DESIGN PARAMETERS OF THE REACTOR [7]

Items	Values
Power level (thermal)	Steady state = 3000 kW; Pulsing = 852 MW
Fuel-moderator material	uranium = 20.00 wt.%; $ZrH_{1.6}$ = 79.53 wt.%; ^{167}Er = 0.47 wt.%
Uranium enrichment	19.70 wt.%
Prompt negative temperature coefficient of reactivity	1.07×10^{-4} $\Delta k/k/^{o}$C
Fuel element dimensions (overall)	diameter = 3.73 cm; height = 75.18 cm
Cladding	material = Type 304 SS; thickness = 0.5 mm
Active core volume	diameter = 55.25 cm; height = 38.1 cm
Core loading	fuel elements = 93; IFE* = 02; FFCR** = 05
Control rod	material = boron carbide (B_4C) number = 6 (Transient-1, Shim-4 and Regulating-1)
Reflector	material = graphite with Al cladding; radial thickness = 19 cm height = 52.7 cm
Maximum neutron flux	7.46×10^{13} n·cm^{-2}·s^{-1}

*instrumented fuel elements, **fuel follower control rods

2.2. Reactor tank

The reactor tank (which is also called the pool liner) accommodates the reactor core. The reactor core is located near the bottom of the reactor tank. The tank is made of special aluminum alloy and has a length of 8.23 m and a diameter of 1.98 m. It is filled up with 24,865 l (6578 gallons) of demineralized water.

2.3. Reactor core

The reactor core consists of 100 fuel elements (93 standard fuel elements, 5 fuel follower control rods (FFCR) and 2 instrumented fuel elements), 6 control rods (5 FFCR and 1 air follower control rod), 18 graphite dummy elements, 1 Dry Central Thimble, 1 pneumatic transfer system irradiation terminus and 1 neutron source. Figure 1 shows the core configuration of the reactor. All these elements are placed and supported in-between two 55.25 cm diameter grid plates and arranged in a hexagonal lattice.

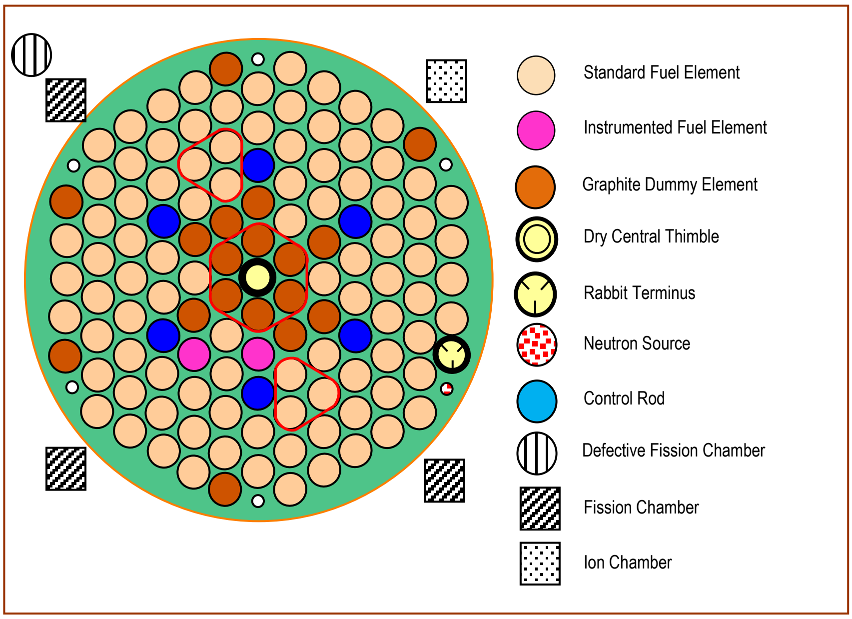

FIG. 1. Core Configuration of the Reactor.

2.4. Reactor fuel element

Fuel of the BAEC TRIGA research reactor is composed of 20 wt.% uranium enriched to 19.7% (the amount of ^{235}U isotope is 19.7%), zirconium hydride (ZrH1.6) and burnable poison erbium (167Er). Figure 2 shows standard TRIGA fuel element. The inherent safety feature of the TRIGA fuel design has been achieved through the use of erbium uranium zirconium hydride (Er-UZrH) material for the fuel-moderator elements. This gives the TRIGA core a large prompt negative temperature coefficient of reactivity and thus makes the core to withstand pulsing operation. The nominal value of prompt negative temperature coefficient of reactivity for the reactor is about 1.07×10^{-4} $\Delta k/k$/°C. The burnable poison erbium in the UZrH matrix contributes to the long core lifetime for the TRIGA reactors.

2.5. Reactor control system

The reactor is controlled by six control rods, which contain boron carbide (B_4C) as the neutron absorber material. When these rods are fully inserted into the reactor core, the neutrons emitted continuously from the start-up source ($^{241}Am/^{9}Be$-source) are absorbed by the rods and the reactor remains subcritical. If the control rods withdrawn from the core, the number of fissions in the core and the power level increase. The start-up process, which is accomplished by withdrawal of all control rods in steps, takes roughly about 10 minutes for the reactor to reach a power level of 3 MW from the subcritical state. The reactor can be shutdown (scram) either manually or automatically by the safety system through the insertion of control rods into the core. Figure 3 shows the fuel follower control rod.

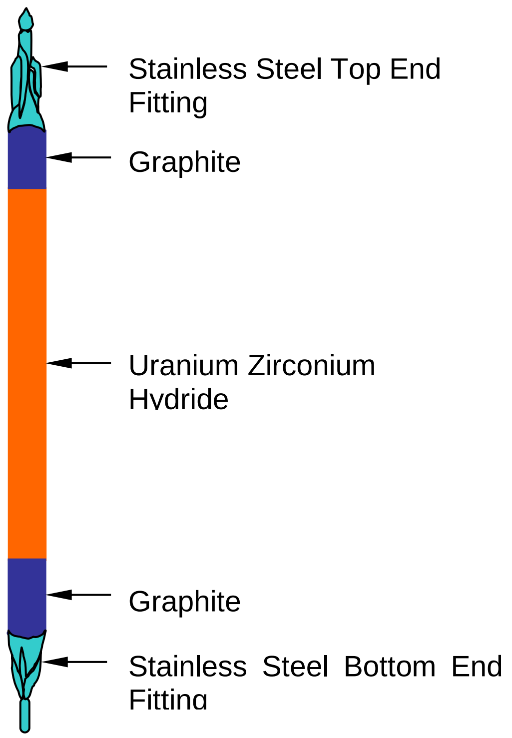

FIG. 2. Standard TRIGA Fuel Element.

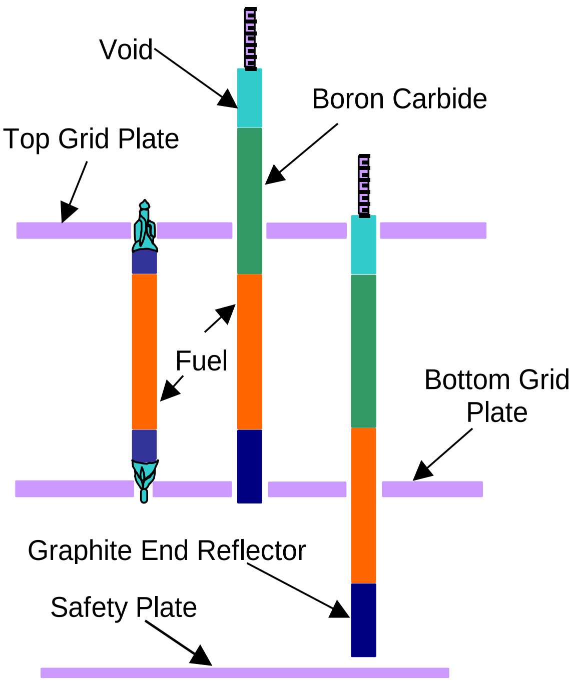

FIG. 3. Fuel Follower Control Rod (Withdrawn and Inserted).

2.6. Instrumentation and control system

Instrumentation and Control (I&C) system plays the key role in ensuring safe operation of a nuclear reactor. The reactor I&C system is a computer based system which include instrumentation for monitoring reactor parameters during all operational states and for recording all variables important to reactor operation. It also manages all control rod movements taking into account the choice of operating mode and interlocks. There are three major system components, the Control System Console (CSC), Data Acquisition and Control (DAC) and Reactor Protection System (RPS). The CSC provides the necessary controls to safely operate the reactor in its various modes of operation. It contains the indicators, annunciators and monitors to present the data in meaningful engineering units and graphic displays to the operator. The DAC is a computer-based system that provides interface functions between the CSC and the reactor. It acquires data in the form of electronic signals from instrumentation in the reactor and auxiliary systems, processes it, and transmits it to the CSC for display. The primary function of the RPS is to scram the reactor by causing the control rods to insert into the core in response to certain abnormal reactor operating conditions. The power level of the reactor is monitored by four neutron channels during normal operation. For neutron detecting elements four Fission/Ion chambers are in use, each connected to separate modules, which allow efficiently system control. The startup function and precise monitoring of reactor power level is achieved by the Nuclear Log Wide Range Channel (NLW-1000)/Nuclear Linear Power channel (NMP-1000) of the (I&C) system of the reactor. The channel indicates the current level which is proportional to the neutron flux from source level (approximately 2 nV) to full power level (7.46×10^{13} nV). Figure 4 shows control console system of the reactor [8].

FIG. 4. Reactor Control Console System.

2.7. Experimental facilities

The experimental facilities available in the reactor are as follows:

2.7.1. Beam Tube

The reactor is equipped with four beam tubes. The beam tubes penetrate the concrete shield and aluminum tank and pass through the reactor tank water to the reflector. These tubes provide beams of neutron and gamma radiation for a variety of experiments. They also provide irradiation facilities for large specimens (up to 15.24 cm diameter) in a region close to the core. Three of the beam tubes are radially oriented with respect to the center of the core; the fourth tube is tangential to the outer edge of the core. Two of the radial tubes, one of which is aligned with a void in the graphite reflector, terminate at the outer edge of the reflector assembly. The third radial tube also known as piercing beam tube penetrates the graphite reflector and terminates at the inner surface of the reflector. The tangential beam tube terminates at the outer surface of the reflector, but it is also aligned with a cylindrical void in the reflector graphite. Currently, the piercing beam tube is being used for the triple axis spectrometer, while the tangential beam tube for neutron radiography. At present, radial beam tube # 2 is being used for neutron scattering experiment.

2.7.2. Dry Central Thimble

The Dry Central Thimble (DCT) is located at the center of the reactor core and provides space for irradiation of samples at the point of maximum flux, which is about 7.46×10^{13} $n \cdot cm^{-2} \cdot s^{-1}$. It provides the extraction of a highly collimated beam of neutron and gamma radiation. It is equipped with a dogleg bend located at a depth of about 4.7 m from the pool water surface. This bend helps to avoid direct streaming of radiation from the core. This experimental facility is mainly used for the irradiation of tellurium dioxide to produce iodine-131. It is also suitable to irradiate the samples of water, sand, human hair, vegetables, soil, etc. for the determination of elements content like arsenic, chromium, uranium, thorium in it by NAA at low power.

2.7.3. Pneumatic Transfer System (PTS)

The Pneumatic Transfer System also called the Rabbit System terminates directly in the reactor core. A cylindrical shaped specimen capsule of 3.17 cm diameter (outside) and 13.97 cm length is used here. The system is applied for the production of very short-lived radioisotopes, according to rapidly convey a specimen to and from the reactor core. It allows the transfer of a sample to be irradiated into the reactor core or out from there in about 4.6 s, thus handling of short-lived radioisotopes, particularly for the purpose of NAA, is possible.

2.7.4. Rotary Specimen Rack (Lazy Susan)

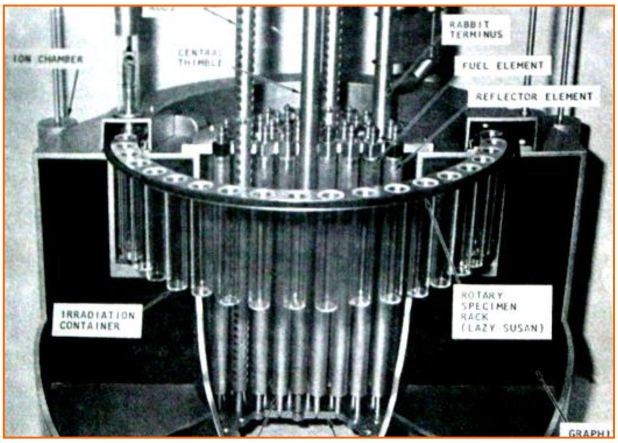

FIG. 5. Rotary Specimen Rack.

The rotary specimen rack also called the Lazy Susan is a “Donut” shaped watertight device placed in the upper part of the graphite reflector assembly around the reactor core. The Lazy Susan assembly consists of a stainless steel rack that holds specimens during irradiation and ring-shaped, seal-welded aluminum housing (see Fig. 5). The rack can be rotated inside the housing and supports 41 evenly spaced aluminum tubes that are open at the top and, except for the tube in the number one (1) position, closed at the bottom. These tubes serve as receptacles for the specimen containers. Each of these tubes (except the tube in the number one position) can accommodate two 13.97 cm long and 3.17 cm diameter (outside) standard specimen containers in it. The top 13.97 cm of the first aluminum tube (position one on the dial indicator) is identical to the other 40 tubes, but the bottom half necks down to an outer diameter of 2.23 cm and is open at the bottom. Only one standard specimen container fits into this tube. We can use this system for the production of radioisotopes.

2.7.5. Cutouts in the Core

The reactor is also equipped with three in-core experimental positions. Two of them are known as triangular cutouts, the other as hexagonal cutout. At present triangular cutout locations are occupied by fuel elements. However, if the fuels and cutout plates are removed from these locations, these facilities can be used for irradiation of sample having a container diameter of up to 6.1 cm. Hexagonal cutout allows in-core irradiation of sample having container diameter 11.18 cm. This large diameter of the sample container is not suitable for its insertions into the DCT or Lazy Susan.

2.7.6. Thermal Column

This facility is not in usable condition now. This is because, instead of being filled up with graphite blocks the thermal column liner is now filled up with heavy concrete blocks. However, if it is needed in future, concrete blocks would be replaced by graphite blocks and the facility would be made operational for the users.

2.8. Reactor cooling system

The BAEC TRIGA research reactor has a maximum continuous thermal power output of 3 MW. The reactor has been designed for operation under three operation modes namely, steady state, square wave and pulse mode. The steady state mode of operation could be performed under two cooling modes – (i) Natural Convection Cooling Mode (NCCM) and (ii) Forced Convection Cooling Mode (FCCM). The NCCM can be used for a power level of up to 500 kW. During NCCM of reactor operation, heat generated in the reactor core is removed by the tank water through natural convection cooling mechanism. For operation at higher level up to the full power of 3 MW, FCCM is used. Heat produced during this mode of operation is dissipated into the atmosphere through a cooling system consisting of primary and secondary cooling circuits.

When the primary water passes through the reactor core, the oxygen present in the water interacts with fast neutron and gets converted into highly radioactive ^{16}N according to the following (n, p) reaction:

$$^{16}_{8}O + ^{1}_{0}n \rightarrow ^{16}_{7}N + ^{1}_{1}p$$

The primary water containing this highly radioactive ^{16}N is passed through the ^{16}N decay tank (capacity: 32,000 l), which holds the water for about 140 seconds before it enters into the primary pumps. The decay tank and the piping between it and the tank is covered with about 102 cm thick heavy concrete shielding in order to attenuate the high energy gamma (~6.13 MeV) being emitted by the N-16 nuclei. During this period activity of the short lived N-16 ($T_{1/2}$= 7.14 s) decays down to low level. Figure 6 shows the research reactor water systems.

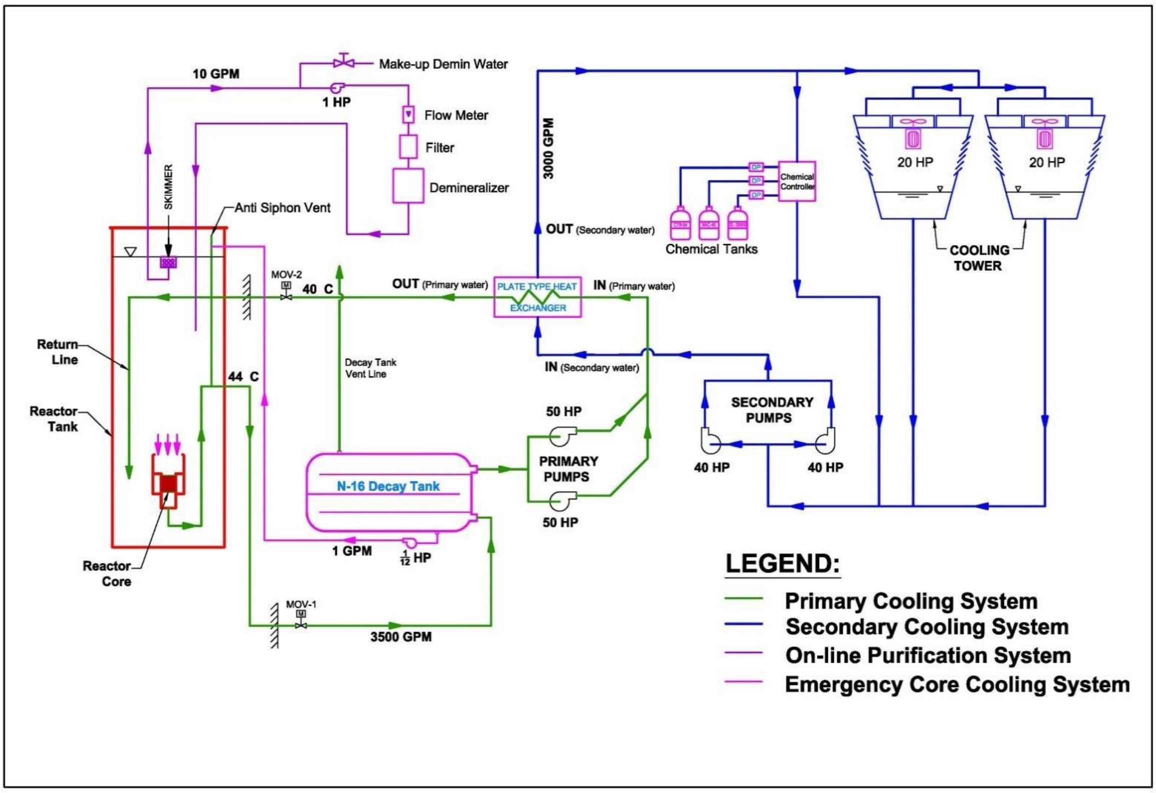

FIG. 6. Research Reactor Water Systems [3].

The reactor core must be covered with water for a period of time following the loss of the primary coolant. The emergency core cooling system is designed to meet this requirement. The ECCS consists of the reactor core assembly; two float level switches, the N-16 decay tank, a battery-operated pump, and a backup water supply (2000 gallon water tank located at the top of the reactor building).

2.9. Emergency core cooling system

The reactor core assembly includes an upper shroud, which is designed to retain 76.2 cm (2.5 ft) of water over the reactor core in the event of a tank or beam tube rupture or a break in the primary coolant line. The water retained within the core assembly will remove the fission product decay heat from the fuel element primarily by evaporation. As the water is boiled out of the reactor core assembly, it must be replaced by the emergency core cooling system. The system acts in the following manner:

The second set of redundant float level switches turns the emergency core cooling system on when the pool level drops to 4.42 m (14.5 ft) below the normal pool level. The pump and the float level switches receive their electrical power from a 12 V Ni-Cd battery set. The battery set is connected to the building emergency electrical power supply system so as to keep the battery set fully charged all the time. The capacity of the battery set is about 168 Ah. The battery set is capable of operating the ECCS pump for about 28 hours even if the building power supply is totally disrupted [2].

A backup water supply system is provided by connecting the 7,560 l (2000 gallon) capacity reactor building rooftop water tank to the emergency piping system through a normally closed and pad-locked Worcester valve. This backup tank receives water from two sources. One of the sources is the central water system of AERE and the other is the 75,600 l (20,000 gallon) underground tank located near the cooling towers of the reactor cooling system. A 10 HP centrifugal pump is used to transfer water from the underground tank to the overhead backup tank. This backup water supply system can be used in the event of failure of the emergency pump or loss of water from the N-16 decay tank. The operation of the backup system is manually controlled. It is to be noted that the loss of coolant analysis indicates that it will require approximately 0.75 hours for the initial charge of water in and above the core to begin boiling and an additional 8.6 hours to be boiled off, thus allowing time to start the backup system if required.

2.10. Special safety features

2.10.1. Intrinsic Safety Features

The objective of intrinsic safety requires that a high degree of operational safety be achieved independent of mechanical, electrical devices, or human actions. The means by which this can be achieved is a design, which has a large prompt negative temperature coefficient that is sufficient to control the effects of a sudden large insertion of positive reactivity. This safety criterion has been achieved by the development of a solid fuel-moderator element consisting of a hydride U-Zr alloy. The combined fuel-moderator material of these elements provides a large prompt negative temperature coefficient, which has been found to be sufficient to allow sudden large insertions of positive reactivity with no damaging effects to the fuel-moderator material. Furthermore, tests have shown that if the inserted positive reactivity remains as excess reactivity above that required for cold criticality, the reactor power following the power transient would decrease to an equilibrium value that is within normal steady-state operating limits.

In addition to achieving the desired negative temperature coefficient, characteristics are necessary for safety as following:

(i) Metallurgical properties of the fuel-moderator alloy that ensure integrity of the material during either sudden increases in temperature or prolonged periods of high temperature

(ii) A suitable cladding that contains the fission products. This requires properties that withstand the thermal and mechanical stresses and strains resulting from high temperatures and gas pressures in the fuel region.

(iii) A core that is under-moderated, as a safety measures against the loss of water moderator or the formation of voids in the core region.

Using an H/Zr ratio of 1.6 to 1.7 the capabilities of the fuel-moderator material extend beyond 1000°C. The stainless steel cladding offers high-temperature capability, the integrity of it depends on its tensile strength relative to the internal pressure at high temperature.

As a safety measure against the loss of water moderator or the introduction of voids into the core, the volume fraction of water makes the system under-moderated to obtain a negative void coefficient of reactivity. The core matrix is arranged in six evenly spaced, concentric hexagons in order to obtain uniform cell structure and uniform cooling.

2.10.2. Passive Safety Features

In designing the reactor the following passive safety features have been considered:

i. High fission product retention in the fuel matrix
ii. No metal-water reaction
iii. High fuel clad strength
iv. Ability to withstand high fuel temperature (up to 1150°C)
v. Anti-siphon cooling line

2.11. Radiation protection

In order to protect operational personnel and environment from the radiation during reactor operation, the following systems are included in the reactor facility:

i. Reactor hall ventilation system
ii. Area radiation monitoring system
iii. Continuous air monitoring system
iv. Hand and foot monitoring system
v. Shielding
vi. Air tight doors
vii. N-16 decay tank
viii. Stack monitoring system

2.12. Emergency power supply

ROMU has a 1000 kVA, 11 kV/0.4 kV electrical substations fed by 11 kV feeder of the Dhaka Palli Bidyut Samiti located at a distance of about 5 kilometers from the reactor site. The total loads of the ROMU are about 450 kVA. In case of failure of normal power, the reactor facility has uninterruptible power supplies (UPSs), one portable petrol generator and two diesel generators. UPSs are directly connected to the digital control system and thermal power calculator of the reactor. The UPS serves uninterruptible power (120 VAC) to the reactor control system. The ECCS battery supplies 12 VDC power to the ECCS pump-motor unit. The capacity of the single phase petrol generator is 5 kVA and the capacities of the other two three phase diesel generators are 250 kVA and 650 kVA respectively. In absence of normal power supply, the battery pack (which has a capacity of 168 Ah) can supply power to the ECCS pump for about 28 hours. The 650 kVA diesel generator is capable of catering the total loads of the ROMU and the radioisotope production laboratory.

2.13. Manpower

Reactor Operation & Maintenance Unit (ROMU) is responsible for operation and maintenance of the reactor and all associated equipment and systems ensuring appropriate level of safety. At present nineteen (19) technical staff are working in ROMU including two senior reactor operators and seven reactor operators. The same personnel are responsible for operation as well as the maintenance activities of the reactor facility. Figure 7 shows the organization structure of ROMU.

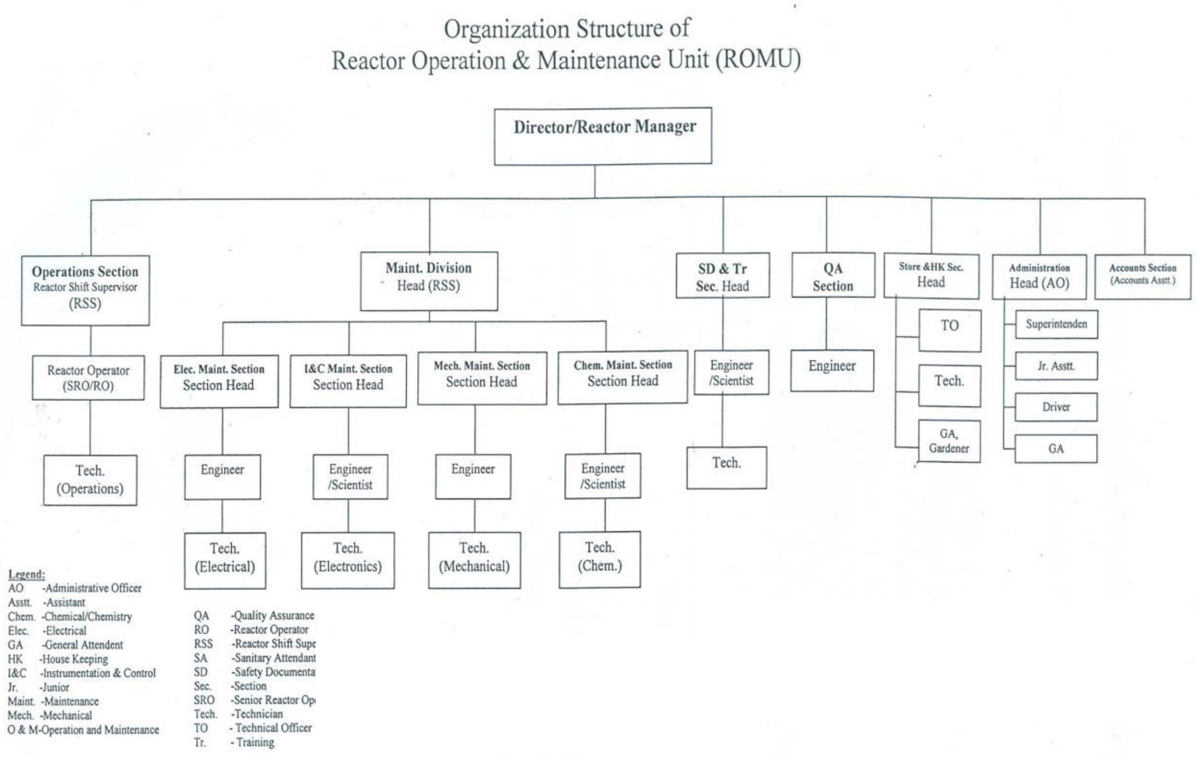

FIG. 7. ROMU Organization Structure.

User personnel of Institute of Nuclear Science and Technology carry out R&D activities using different experimental facilities of the research reactor. There also exist several different committee including expert scientists of BAEC who are responsible for ensuring safe and reliable operation, maintenance and utilization of the research reactor.

3. APPLICATIONS AND UTILIZATION EXAMPLES, INCLUDING COLLABORATIONS

The reactor has so far been used for operator training, sample analysis and material study through neutron activation and neutron scattering, production of radioisotopes and neutron radiography. Up to 31st December 2013, about 3000 samples have been irradiated under 1400 Irradiation Requests (IRs) received from different reactor users. The reactor is now being operated for irradiation of different samples for various R&D groups. A brief description of reactor utilization is given in the following subsections.

3.1. Manpower training

The reactor has been used for the following training programs:

SRO/RO training: The facility has been used for training and retraining programs of the reactor operating personnel (including foreigners) to the level of Senior Reactor Operator (SRO) and Reactor Operator (RO). The RO and SRO licensing examination and tests conducted by the regulator as per the Nuclear Safety and Radiation Control Rules-97 (NSRC Rules-97) of Bangladesh.

Academic training: The reactor facility along with the associated laboratories of the INST, AERE has been used successfully for carrying out routinely the thesis works of PhD students and MSc/MPhil students from different public universities of the country. Undergraduate engineering students from Engineering Universities of Bangladesh received their industrial training at the reactor facility.

3.2. Radioisotope production

In 1987, MoO_3 powder was irradiated in the reactor to produce Tc-99m, which is the first radioisotope ever produced in the country. The Tc-99m production program was continued until 1993. Afterwards BAEC started the manufacture of Tc-99m generators based on imported fission molybdenum. The other medical radioisotopes produced using the reactor include I-131 and Ho-166 (Ho-166 was produced for research purposes only). Additionally, Sc-46 was synthesized in the reactor for isotope hydrology research work.

3.3. Research and development (R&D)

3.3.1. Neutron Activation Analysis

The Neutron Activation Analysis (NAA) lab is engaged in the determination of major, minor and trace elements in geological, biological, industrial, nutritional and health-related environmental samples. Particularly the following can be mentioned: trace and toxic elements analysis in geological samples, e.g., soil, sediment and rock; a study of environmental pollution due to discharge of solid waste and liquid effluents from tannery, textile and dyeing industries, a fertilizer factory, a paper mill; determination of trace, toxic and essential elements in local and imported food items in order to establish baseline data of the element profiles in relation to human health and nutrition; and determination of impurities in industrial products, semiconductor materials, etc. At present arsenic toxicity in groundwater is a serious problem for the nation. The NAA laboratory has therefore given special emphasis on the determination of arsenic toxicity in water, soil, foodstuff and biometrics of arsenic-affected patients. The NAA laboratory gives analytical services to arsenocosis patients coming from different hospitals, government departments and non-governmental organizations, universities, etc. The acceptable results in proficiency tests on specimens like IAEA-Algae, IAEA-Lichen and FNCA-Sediments conducted by International Atomic Energy Agency (IAEA) and Forum for Nuclear Cooperation in Asia (FNCA) have established high standard of the quality of chemical analysis of this NAA laboratory. The NAA technique is also applied to nuclear data measurement at different neutron energies using the TRIGA reactor and neutron generator. Since the installation of the research reactor, the radial piercing beam port has been utilized for NS experiments. Recently, this beam port has been used for thermal neutron capture cross-section measurements at neutron energy of 0.0536 eV.

3.3.2. Neutron radiography

Neutron Radiography (NR) Facility was established at the tangential beam port of the reactor for non-destructive testing of materials with an objective to utilize the reactor more potentially. The film neutron imaging method is being used from the beginning of the facility. Recently, digital neutron radiography set-up has been added to the facility along with a change in biological shielding arrangement. Introduction of digital neutron set-up has significantly reduced the experimental time and digitized images of the objects are obtained very fast and processed by software to improve the quality. However, film technique is still used in parallel for some specific purposes. Since installation of the facility many neutron radiography experiments have been done on samples relating to many areas of applications. Rubber and leather products, wood and jute plastic composites, radiation shielding materials and a variety of industrial materials/products have been studied for detection of internal defect, strength and integrity. Internal structure and water absorption behavior in ceramic tiles and other building materials have also been investigated.

3.3.3. Neutron scattering

The Triple Axis Neutron Spectrometer (TAS) has been in use in the reactor facility since 1992. The facility is now being used mainly for neutron diffraction studies. The major research activities carried out so far using the TAS includes different types of ferrite materials, namely, $BaFe_{12}O_{19}$ $Zn_{0.85}Ni_{0.15}Fe_2O_4$, $Zn_xMg_{0.75-x}Cu_{0.25}Fe_2O_4$, etc., at room temperature, and also different types of superconductors. With some necessary adaptations, a test Small Angle Neutron Scattering (SANS) experiment was also performed on alumina samples using the TAS in a double crystal method using the Bonse and Harts technique. A high performance Neutron Powder Diffractometer (NPD) SAND (Savar Neutron Diffractometer) has been installed on radial beam port 2 in an air conditioned room fabricated on site in order to maintain the temperature and humidity with sample environment controlling devices. SAND is additionally equipped with ^3He cry refrigerator for low temperature (10°C) and furnace (600°C) for high temperature experiments. The TAS, a LCR bridge, a sample preparation laboratory equipped with a ball-mill, a hydraulic press, a micro-balance and several furnaces for fabrication and heat treatment of the samples. NS group mainly deals with the characterization of functional materials such as:

◘ Metals ◘Metallic Alloys ◘Metallic oxides ◘ Ceramics◘ Ferrites◘ Super conductor ◘Amorphous materials ◘Perovskites

3.4. National and international collaboration

Reactor and Neutron Physics Division (RNPD) have been running collaborative research programs with national and international Research Institutes and Universities. The division has research collaboration with Chalmers University, Gothenburg, Sweden on Neutron Scattering research. RNPD has also close cooperation with the national universities relating to academic programs and research activities.

4. SUCCESS STORIES, MAJOR ACHIEVEMENTS

Reactor building started through the “Ground Breaking Ceremony” in May 1981. And after several years of hard labor the reactor achieved its first criticality on 14 September 1986. This also starts a new era in technology and research fields. In October 1986 the reactor operated at full power of 3 MW (Thermal). Since its commissioning, the facility faced several problems but with the help of IAEA the sole manpower of ROMU successfully overcome those issues. Even now the reactor is operated in safe and stable and maintains all safety conditions. The major achievements are as follows.

Transfer of technology in the area of operation and maintenance of research reactor.

Successful completed installation of high Performance Neutron Powder Diffractometer in the Radial Beam Port-2 of the Reactor in February 2010.

Leakage incident of the Radial Beam Port-1 took place after removal of the beam port plugs in September 2009.Restoration of Reactor Operation after solving the Beam Port leakage problem and installation of high performance Neutron Powder Diffractometer in March 2010.

- Preparation of various operational and safety documents as required by the Nuclear Safety and Radiation Control Rule (Rule-97).
- Upgrading of reactor cooling system.
- Restoration of full power operation of the reactor after the decay tank leakage incident of 1997.
- Repair of RBP-1 using local technology and expertise.
- Entry into the Integrated Safeguards (IS) of IAEA since January 2007.
- Modernization of reactor control system by installing digital control console system.
- Upgrading of safety documentations, etc.

5. FUTURE PROSPECTS (ISSUES AND CHALLENGES)

The reactor is in operation for 28 years. ROMU’s engineers and scientists took entire responsibility of operation and maintenance work after installing of the reactor. Due to long operation and maintenance experience, good maintenance capabilities for instrumentation, electrical and mechanical systems have been achieved. Significant domestic participation during initial installation and subsequent phases of major repair/replacement of decay tank, heat exchanger, cooling system modification, digital console system installation etc. enhanced knowledge in the field of nuclear technology. This facility has sincere and dedicated work force. The government has great interest to use this facility as nuclear hub of the country to develop human resources in the field of nuclear sciences. As a consequence few annual development projects funded by government of Bangladesh have been finished successfully which ultimately confirm the long time operation of the facility. It is to be mentioned that an ADP project funded by Bangladesh Government has been submitted to BAEC in order to enhance the life time of the reactor. Under this project spent fuel management facility, stress test analysis of reactor, physical protection systems and radiation protection program will be developed. Bangladesh Govt. has the highest priority to establish nuclear power program in the country. BAEC management is very much interested to enhance R&D activities in frontier areas of reactor engineering, reactor physics, etc. and manpower training in reactor technology using this facility. The facility was greatly involved in production of I-131 to meet national need of diagnoses different diseases which will be strengthened in future. Presently the facility is involved reactor physics calculations and measurements which will ultimately ensure the safety of the reactor. The major challenges of the facility are uncertainty in the supply of spare parts, replacement components, collection of new fuel elements, inadequate facilities for handling and storage of spent fuel and drainage of trained/qualified manpower. It is basic need of the facility to develop manpower in the field of reactor physics, thermal hydraulics and safety. The reactor facility needs to upgrade safety, O&M and training documents considering different major up gradation/changes.

6. CONCLUSION

The BAEC TRIGA research reactor has so far been operated safely for various peaceful applications in the field of nuclear science and technology in the country. Since its commissioning, a few incidents took place at the facility, which were managed satisfactorily. Most of the modification, rectification and upgrading works of the facility were carried out locally. The reactor is being utilized for producing Radio–Isotope (RI) for its medical uses, conducting various R&D activities and manpower-training program of the country. As a part of the modernization program a new digital control console, digital neutron radiography facility and digital neutron powder defractometer has been installed. There is a plan to develop the unused experimental facilities such as the radial beam ports and the thermal column for strengthening the R&D activities around the reactor. The reactor will be utilized for manpower development of the future nuclear power program in the country.

REFERENCES

[1] M. A., ZULQUARNAIN, M. M., HAQUE, M. A., SALAM, M. S., ISLAM, P. K., SAHA, M. A., SARDER, A., HAQUE, M. A. M., SONER, M. M., UDDIN, M. M., RAHMAN, I., KAMAL, M. N., ISLAM, S. M., HOSSAIN, Experience with the operation, maintenance and utilization of the 3 MW TRIGA Mark-II research reactor of Bangladesh" International Journal of Nuclear Energy Science and Technology, **4**, 4 (2009) 299 - 312.

[2] M. A., SALAM, A., SARDER, A., KALAM, A., HAQUE. M.A.M., SONER, M.M., UDDIN, M., SHABUDDIN, M. M., RAHMAN, A., RAHMAN, B., SHOHAG, M. G., S., ISLAM, I., KAMAL, M.M., HAQUE, AND M. A., ZULQUARNAIN, Refurbishment, modernization and ageing management program of the 3MWTRIGAMark-IIResearch Reactor of Bangladesh, Conference on International group of Research reactor IGORR 2013 and IAEA Technical Meeting 13-14 Oct, 2013 Proceeding of IGORR 2013, Index Number 02-1038

[3] M. A., SALAM, M.A.M., SONER, M. A., SARDER, A., HAQUE, M.M., UDDIN, A., RAHMAN, M. M., RAHMAN, M. M., SARKER, S. M. A., ISLAM, Measurement of neutronic safety parameters of the 3 MW TRIGA Mark-II research reactor, Progress in Nuclear Energy, **74**, (2014) p.160-165.

[4] M. A., SALAM, M. A., SARDER, A., HAQUE. A., RAHMAN, M. B., SHOHAG, M., M., RAHMAN, M.M., SARKER AND S. M. A,. ISLAM, Control Rod Drop Time Measurement of 3 MW TRIGA Mark-II Research Reactor, Jahangirnagar University Physics Studies, **19** (2013).
[5] M. A., SALAM, M.A.M., SONER, M. A., SARDER, A., HAQUE, M. M., UDDIN, M. M., SARKER, S. M. A., ISLAM, Measurement of control rod reactivity and shut down margin of 3 MW TRIGA Mark-II research reactor using analog and digital I&C system, Annals of Nuclear Energy **68** (2014) p. 257-261.
[6] Safety Analysis Report of the BAEC 3 MW TRIGA MARK-II Research Reactor, BAEC, Dhaka (2006).
[7] GENERAL ATOMICS (San Diego, CA), System Operation and Maintenance Manual, Document No. E 161 1114, January 1983 Revised (1987).
[8] Reactor Instrumentation & Control system console upgrade, BAEC TRIGA Reactor Facility at Savar, Bangladesh, GA document.

THERMALHYDRAULICS ANALYSIS AND CURRENT STATUS OF BANDUNG TRIGA RESEARCH REACTOR

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Abstract.

Bandung TRIGA reactor is the first nuclear reactor in Indonesia and reached first criticality in year 1964. In the first time, the reactor was operated at a power of 250 kW. The reactor was upgraded to 1000 kW power level in 1971, and to 2000 kW in 2000. The Bandung TRIGA research reactor completed a commissioning program for 2000 kW power and receives an operating license at 2000 kW in the year 2000. During the commissioning tests, bubbly flow phenomena at the reactor core were observed. Due to some safety problems relating to heat transfer in the core, both innovative numerical and experimental methods have been applied as the approach to solve the thermal-hydraulic problems occur in the reactor core. Numerical approach for three-dimensional analysis using CFD software was applied in the finite volume method. The results of this research showed that the coolant flow in the reactor core is not purely induced by the force due to the density difference of the coolant, but also affected by the flow coming in from the primary cooling system. However, since September 2011, the Bandung TRIGA research reactor operating license has been suspended by the Indonesian Regulatory Body because of technical issues related to fuel burn-up. The government of Indonesia (BATAN) expects the Bandung TRIGA research reactor can be operated more safely and strongly supports the continued operation of the Bandung TRIGA research reactor. Recently, we are planning several options for operating reactors again such as procurement of fuel element standard TRIGA, utilize the existing TRIGA fuel and create a new control rod by BATAN and conversion of the standard TRIGA fuel into the plate type fuel.

1. INTRODUCTION

The Bandung TRIGA research reactor is a pool type reactor with a concrete shield, four beam ports, a thermalizing column, and a thermal column (Fig. 1). The reactor reached its first criticality in 1964 and was operated at 250 kW. During operation condition, the reactor is cooled by natural circulation cooling. The heat generated by fission in fuel elements is transported to the coolant (pool water) by natural convection and the pool water is drawn through a coolant pump and forced through a heat exchanger. The pool water is cooled by primary and secondary cooling systems [1, 2]. The reactor was upgraded to 1000 kW in year 1971 and to 2000 kW in year 1996. The Bandung TRIGA research reactor completed a commissioning program and receives an operating license at 2000 kW in the year 2000. During the commissioning tests, bubbly flow phenomena at the reactor core was observed and considered acceptable by GA experts. The sub-cooled boiling observed during commissioning tests confirmed the predictions of the thermal-hydraulics calculations performed by GA and presented in the safety analysis report [1]. In the previous works are also reported that the exit clad temperature for maximum and average powered fuel rod exceeded the boiling point of reactor coolant [2-5]. To accomplish safety requirements, a set of actions has to be performed following the recommendations of the IAEA Safety Series 35 applied to research reactor [6]. Such actions are considered in modernization of the old systems, improve the cooling system and safety evaluations (determine the temperature field in the reactor core). In the present work, a numerical study of velocity fields around the reactor core and temperature distribution in the coolant channel of the Bandung TRIGA research reactor has been carried out using CFD package. The purpose of the analysis is to predict the influence of the flow coming in from the primary cooling system to the velocity fields in the coolant channel and to estimate temperatures of coolant water in the reactor core in order to assure that the reactor operation is safe and under control or the safety limits are not exceeded.

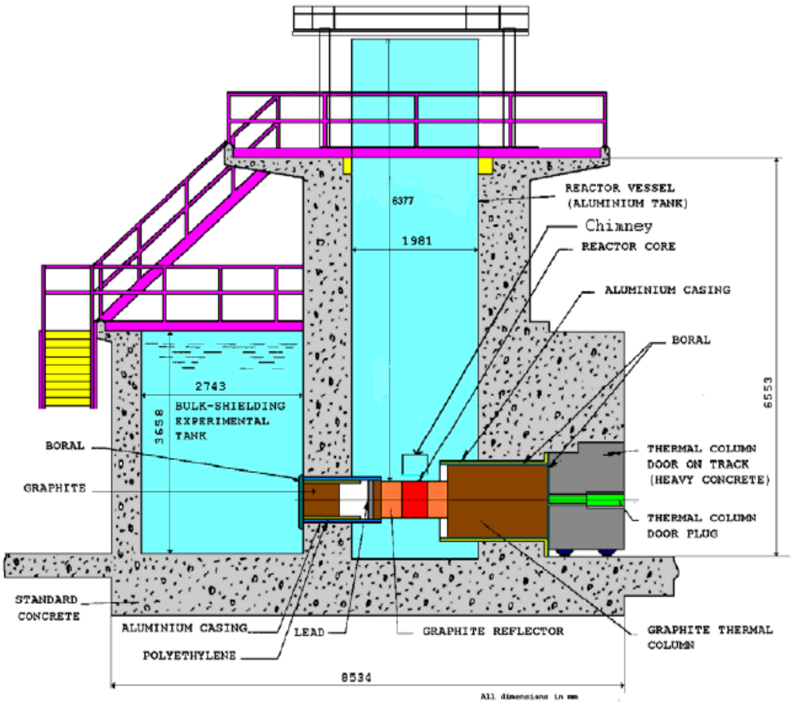


FIG. 1. Vertical cross section of the Bandung TRIGA research reactor.

The major operational challenges of the Bandung TRIGA research reactor are burn-up some FFCR already passed 50%. According to the rules of regulatory authorities, these FFCR has to be removed from the reactor core. Beside of that, approximately 50% of the fuel elements have burn-up more than 40%. The irradiation facility (rotary specimen rack/lazy Susan) was damage and the structure of the building is old and needs to be strengthened because of seismic condition. The important work should be done in 2014 is strengthening of the reactor building (retrofitting) and its implementation will begin in mid-2014 and is planned to finish by the end of 2014.

2. REACTOR COOLING SYSTEM

The reactor cooling system of the Bandung TRIGA research reactor is shown in Fig. 2. It consists basically of reactor pool, primary cooling system with a circulation pump and heat exchanger, and secondary cooling system with a circulation pump and two cooling tower. The design temperature of the Bandung TRIGA research reactor specified in the SAR is 32°C at the heat exchanger outlet or primary water supply to the reactor pool.

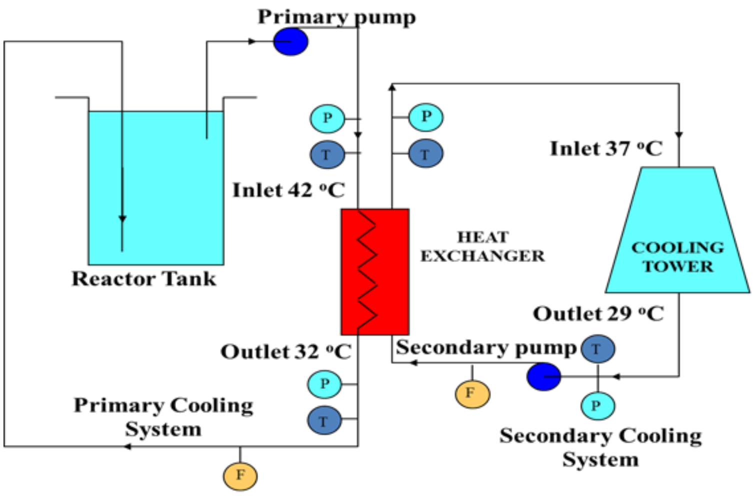


FIG. 2. Reactor cooling system.

The function of the primary cooling system is to transfer the energy or heat contained in the reactor tank to the secondary cooling system through the heat exchanger. The primary cooling system should have the capability to transfer energy as much as 2000 kW in maximal, safety and effectively condition during the continuous operation. A plate type of heat exchanger is used to transfer heat from the primary cooling system to the secondary cooling system.

The function of the secondary cooling system is to transfer heat from the primary cooling system into the air around the cooling tower. This system should also have the capability to transfer the energy as much as 2000 kW in maximal, safety, and effectively condition during the continuous operation. The reactor cooling system of the Bandung TRIGA research reactor also functions as to remove heat decay when the reactor operates or shut down. The reactor cooling system is also connected to other related system with reactor cooling water, such as, emergency core cooling system, purification water system, water supply system for primary and secondary cooling system and diffuser system. The core of the Bandung TRIGA research reactor consists basically of fuel rod arranged in bundle on two grid plates at ends, control rods, graphite reflector wall, and several research set ups. The reflector wall acts as neutrons reflector of the reactor. The flow pattern in the pool of the Bandung TRIGA research reactor is actually complex. Coolant exiting from the reactor core rises toward the top of the pool where some part of it is removed to the primary cooling system. Meanwhile, the coolant from the primary system's heat exchanger is returned to the reactor core at the point near the level of bottom grid plate.

3. PHYSICAL MODEL AND NUMERICAL PROCEDURE

The physical model is shown in Fig. 3 and consists of a reactor tank, a reactor core structure, thermal and thermalizing column, reflector, rotary specimen rack and beam ports. Initially, the fluid inside the reactor tank is at rest, and the wall and water inside are at the same uniform temperature, T = 300 K. At time t = 0, heat is supplied to the outside surface of fuel element by means of an imposed heat flux along the surface of fuel element. It initiates a heat transfer process from the fuel element wall to the fluid in the coolant channel. Meanwhile, the flow is supplied by the primary cooling system to the reactor tank by means of an imposed velocity v = 2.8 m/s. The power distribution for the core at 2000 kW is based on the neutronic calculation and Fig. 4 shows standard configuration of the Bandung TRIGA research reactor core with 116 standard fuel elements (FE) and 5 fuel followed control rods (FFCR). Table 1 shows the power distribution for the core at 2000 kW [7].

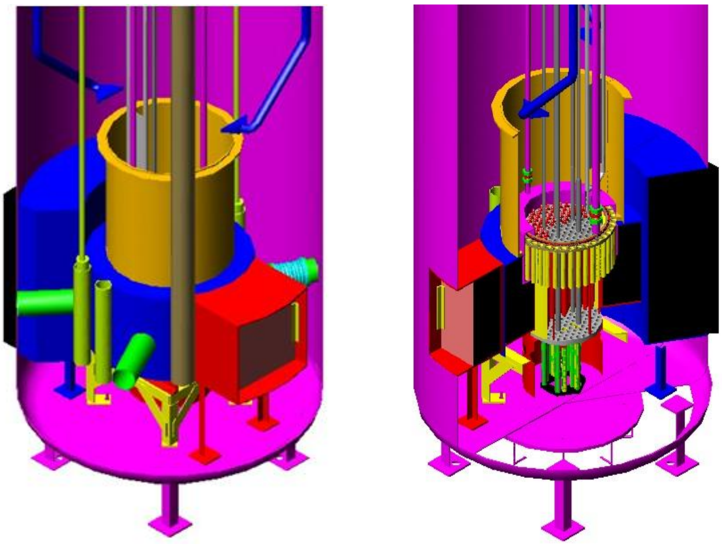


FIG. 3. Reactor core components.

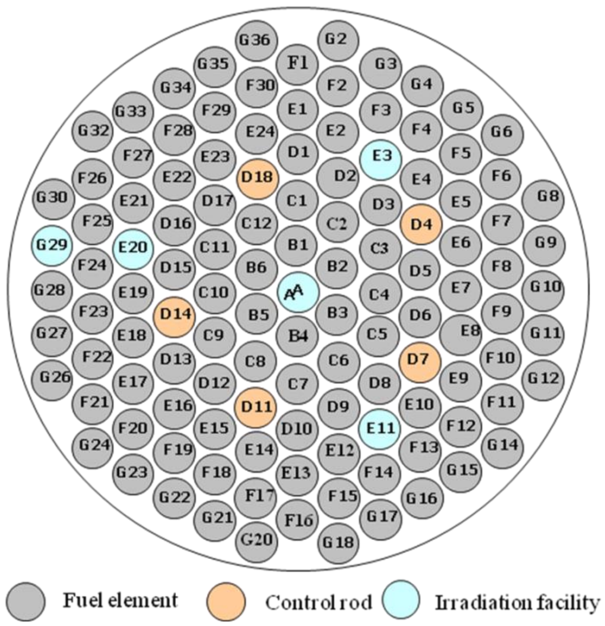


FIG. 4. Core configuration of the Bandung TRIGA research reactor.

In the present calculations, the flow and heat transfer are assumed to be three-dimensional. The CFD-3D code, a general-purpose program for simulating laminar and turbulent flow and heat transfer, is utilized. The schematic of geometry and refine mesh is shown in Fig. 5 and there are 380,000 non-uniformly spaced meshes in the reactor tank volume.

TABLE 1. POWER DISTRIBUTION IN THE STANDARD CORE

Fuel Position	Power (kW)	Fuel Position	Power (kW)	Fuel Position	Power (kW)	Fuel Position	Power (kW)	Fuel Position	Power (kW)
A	0	B1	30.5	B2	30.5	B3	30.3	B4	30.2
B5	30.3	B6	30.1	C1	28.7	C2	30.2	C3	30.7
C4	30.3	C5	28.8	C6	29.8	C7	29.9	C8	30.0
C9	28.5	C10	30.0	C11	30.0	C12	30.2	D1	27.4
D2	26.5	D3	27.0	D4	22.6	D5	26.4	D6	26.0
D7	15.5	D8	26.1	D9	26.0	D10	27.2	D11	20.1
D12	26.0	D13	27.1	D14	16.5	D15	25.9	D16	27.8
D17	26.0	D18	16.8	E1	15.3	E2	19.5	E3	0
E4	20.1	E5	16.0	E6	19.3	E7	24.9	E8	18.9
E9	15.3	E10	17.6	E11	0	E12	19.2	E13	15.2
E14	19.0	E15	20.2	E16	19.0	E17	15.1	E18	18.7
E19	19.9	E20	0	E21	17.1	E22	19.0	E23	20.2
E24	19.0	F1	10.3	F2	12.3	F3	14.2	F4	14.3
F5	12.5	F6	10.3	F7	12.6	F8	13.3	F9	13.2
F10	12.2	F11	9.97	F12	11.9	F13	13.7	F14	13.8
F15	12.1	F16	10.3	F17	12.2	F18	13.5	F19	13.5
F20	12.2	F21	10.3	F22	12.0	F23	13.1	F24	13.4
F25	12.5	F26	17.1	F27	12.2	F28	13.5	F29	13.5
F30	12.2	G2	8.40	G3	9.30	G4	9.66	G5	9.36
G6	8.53	G8	8.54	G9	9.37	G10	9.62	G11	9.25
G12	8.38	G14	8.29	G15	9.06	G16	9.41	G17	9.14
G18	8.32	G20	8.37	G21	9.26	G22	9.57	G23	9.25
G24	8.36	G26	8.29	G27	9.06	G28	9.51	G29	0
G30	8.53	G32	8.34	G33	9.22	G34	9.54	G35	9.24
G36	8.37	-	-	-	-	-	-	-	-

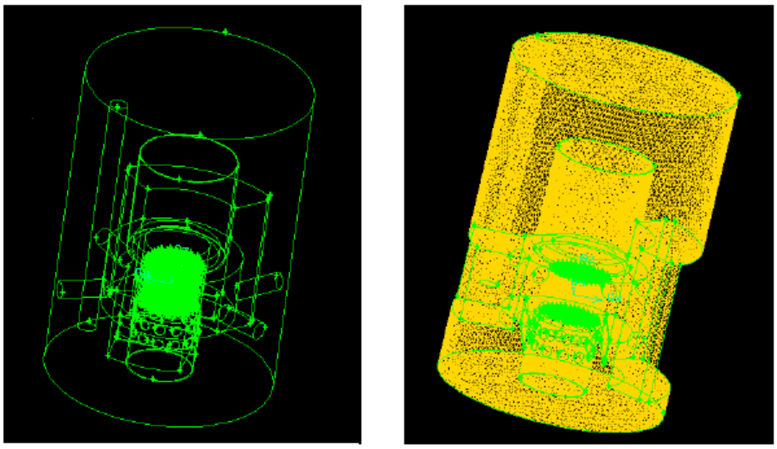


FIG. 5. The schematic of geometry and refine mesh.

4. NUMERICAL RESULTS

In discussing results from the numerical simulation, emphasis is placed on the velocity and temperatures fields around the reactor core in order to clarify the flow and heat transfer mechanism. Figure 6 shows a velocity vector in the outlet of primary cooling system. The velocity fields in the fluid show that a forced convection dominates the heat transfer process in this region. The numerical prediction also show that the flow patterns in the outlet of primary cooling evolve from one-dimensional (flow to negative *z*-direction) in the primary pipe to a vortex flow in the reactor tank.

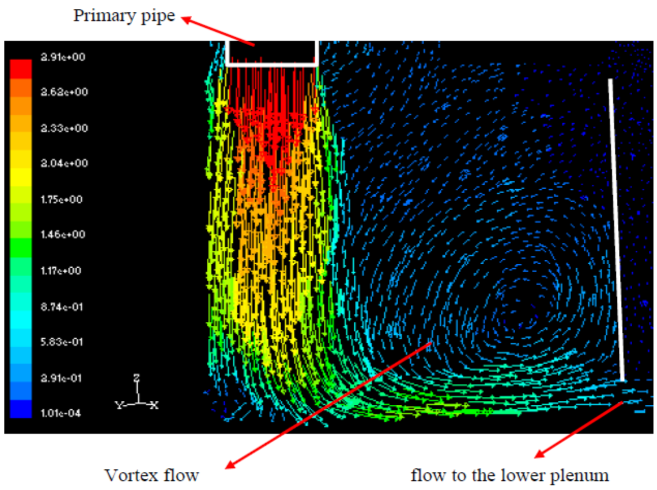


FIG. 6. The velocity vector in the outlet of primary cooling system.

Figure 6 also shows that some of water from the primary cooling flows to a lower plenum of the reactor core and causes the second vortex flow in this region. The vortex flow pattern, shown in Fig. 7, causes the axial flow in the lower plenum and gives an initial velocity to the reactor core. It means, the forced-convection coming in from the primary cooling system affects the flow pattern around the reactor core, and yields small influence to the natural flow rate and heat transfer characteristic in the reactor core.

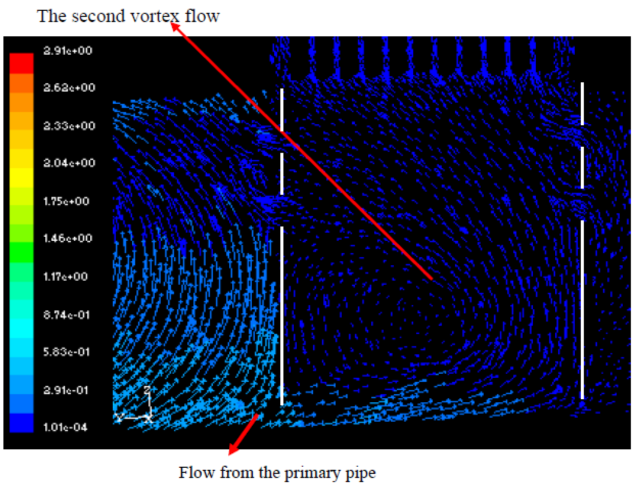


FIG. 7. The vortex flow pattern in the lower plenum.

5. DESCRIPTION OF EXPERIMENT

The purpose of the experimental study is to verify the numerical analysis, especially the temperature distribution in the hottest coolant channel of the Bandung TRIGA research reactor core. In this experiment, a special probe for temperature detection has been designed and inserted to central thimble (CT). The temperature is measured by calibrated thermocouples installed in the probe. In the experiment, eight thermocouples were used to measure the bulk temperature of the water at different levels in the cooling channel and simultaneous quantitative measurement of the temperature distribution were done by using a data acquisition cards system. The experimental setup and the position of thermocouples in the core are shown in Fig. 8.

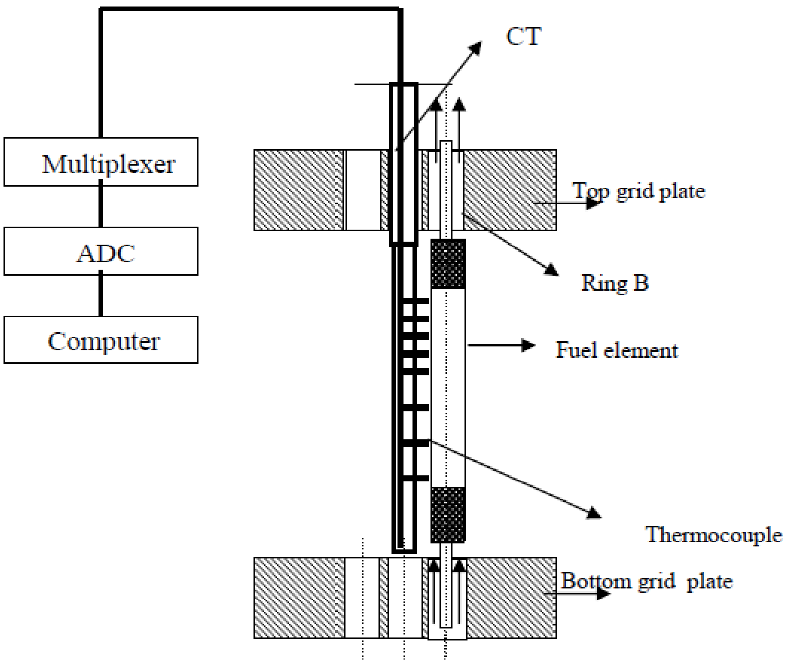
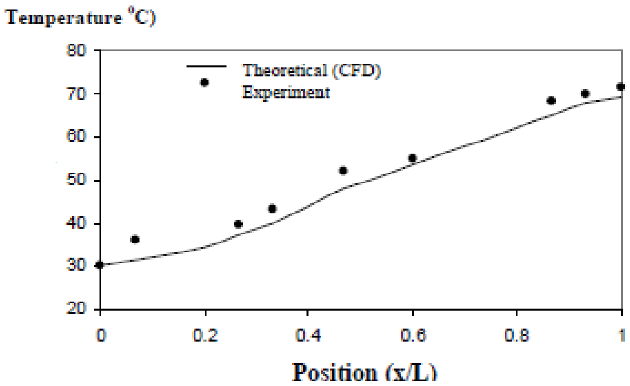


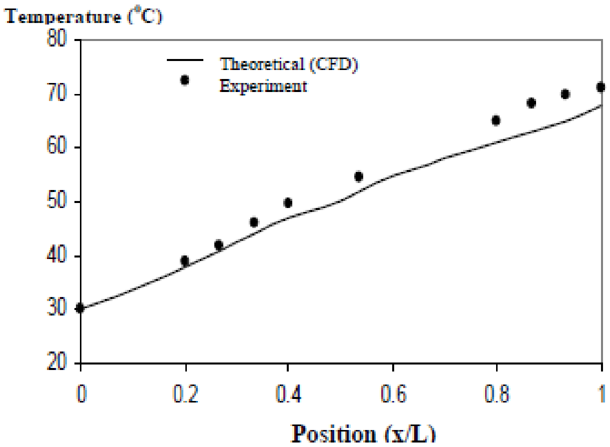
FIG. 8. Schematic of experimental setup

6. COMPARISON OF NUMERICAL AND EXPERIMENTAL DATA

The experimental study has provided the temperature distribution of the coolant water along the coolant channel of the Bandung TRIGA research reactor. The numerical results are compared against our experimental data in Fig. 9(a) and (b) which local temperature of fluid at the coolant channel between CT and fuel element in B1 position and the coolant channel between CT and fuel element in B2 position, respectively. It is seen that agreement is satisfactory. The experimental data along the vertical channel in the reactor core indicated that, for the reactor power is 1000 kW, the temperature of fluid in the hottest channel still below the boiling point and the calculated temperatures reflect this behavior. There exists some discrepancy in the top of coolant channel and this may be caused mainly by the existence of grid plate in the top of coolant channel. In the numerical analysis, the existence of the top grid plate isn't considered, resulting in a different flow resistance from the experimental study.



(a)



(b)

FIG. 9. Comparison of predicted local temperatures with experimental data for 750 kW power.

CFD analysis has also been carried out to calculate the maximum surface temperature of fuel elements and the fluid temperature in the coolant channel for the higher reactor power. The results show that for the initial condition (116 fuel elements in the core) and for the water inlet temperature of 32°C and the primary velocity of 2.8 m/s, the exit clad temperature for maximum powered fuel rod (~135°C) exceeded the boiling point of reactor coolant. It means boiling phenomena occur in the coolant channel. Meanwhile, for the water inlet temperature of 24°C and the primary velocity of 5.6 m/s, no boiling occurs in the coolant channel. The temperature pattern around the reactor core is shown in Figure 10.

7. FUTURE ACTIVITIES

7.1. Facility improvement program

The government of Indonesia (BATAN) expects the Bandung TRIGA research reactor can be operated more safely and strongly supports the continued operation of the Bandung TRIGA 2000 research reactor. For this purpose, BATAN Upper management have promoted a marketing team and a technical team for the Bandung TRIGA 2000 research reactor. The marketing team have been preparing the strategy plan (SP) in order to gather all potential stakeholders (academics, industry, medical application, environmental surveillance). Meanwhile, the technical team have been assessing the different technical options and related economical costs for bringing the Bandung TRIGA research reactor back in operation.

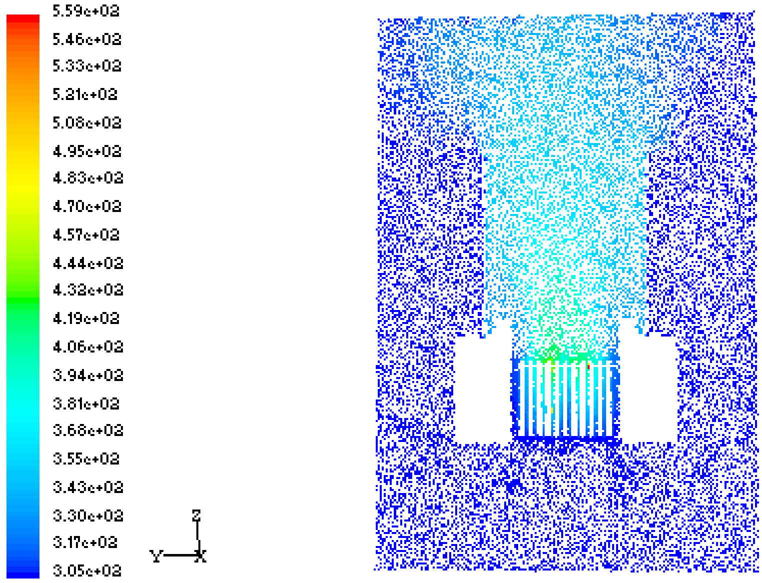


FIG. 10. Temperature pattern of water in the Bandung TRIGA research reactor tank.

The major operational challenges of the Bandung TRIGA research reactor are burn-up some FFCR already passed 50%. According to the rules of regulatory authorities, these FFCR has to be removed from the reactor core. Beside of that, approximately 50% of the fuel elements have burn-up more than 40%. The irradiation facility (rotary specimen rack/lazy Susan) was also damage and the structure of the building is too old and needs to be strengthened because of seismic condition. The important work should be done in 2014 is strengthening of the reactor building (retrofitting) and its implementation will begin in mid-2014 and is planned to finish by the end of 2014.

Recently, we are also planning several options for operating reactors again such as procurement of fuel element standard TRIGA, utilize the existing TRIGA fuel and create a new control rod without fuel follower and conversion of the standard TRIGA fuel into the plate type fuel.

7.2. Reactor operation programme

To operate the Bandung TRIGA research reactor for near future, by utilizing existing TRIGA fuel, BATAN just use the 103 existing fuel elements, one fuel follower control rod (FFCR) and four control rods without fuel follower. Our goal to operate of the reactor is to improve the ability of the supervisors and operators of the reactor. Meanwhile, for the long term operation, we know that the TRIGA fuel elements are not produced anymore by the manufacturer, so that it is necessary to find a solution for the Bandung TRIGA reactor. One proposed solution is to replace the all of fuel elements with plate type fuel elements which are used in the multipurpose reactor (RSG-GAS) in Serpong, Indonesia. This solution is taken in order to reduce dependence on the TRIGA fuel.

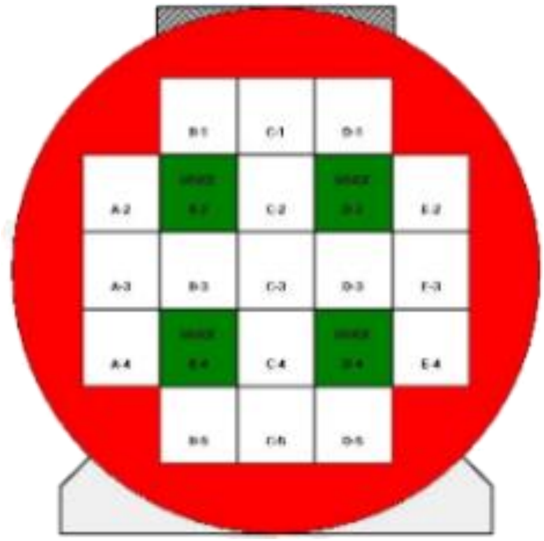


FIG. 11. The shape of the reactor core.

For this purpose, a team has been formed to study the possibility of the reactor operate with the plate type fuel elements. Results of the study showed that by making the possible changes, the Bandung TRIGA research reactor core can be converted to a square core, which contains a maximum of 21 fuel elements, including 4 element controls and one in core irradiation facility. Control elements can occupy the position B2, B4, D2 and D4, while the in core irradiation facility located at the C3 position. It seems that the Bandung TRIGA research reactor can operate in a long period and the shape of the reactor core is shown in Fig. 11 [8]. However, it is necessary to consider that some of the problems to be found with this option are the difficulty in designing and creating and installing a new core. Even so, there are the benefits that the plate type fuel element can be produced by BATAN.

8. CONCLUSIONS

A numerical simulations and experimental study of the thermal-hydraulics characteristic in the Bandung TRIGA research reactor core have been carried out. The main contributions of the present investigation are that:

1. The numerical simulations show that for the initial condition (116 fuel elements in the core) and for the water inlet temperature of 32°C and the primary velocity of 2.8 m/s, the boiling phenomena occur in the coolant channel of reactor core. Meanwhile, for the water inlet temperature of 24°C and the primary velocity of 5.6 m/s, no boiling occurs in the coolant channel.
2. The forced-convection coming in from the primary cooling system affects the flow pattern around the reactor core, and yields small influence to the natural flow rate and heat transfer characteristic in the reactor core
3. The temperature measurement in the coolant channel of the Bandung TRIGA research reactor core has been used to validate the numerical predictions. These confirm that the numerical model is accurately reproducing the principal flow and heat transfer mechanism.
4. In the near future, the reactor will be operated at a power of 1000 kW using existing fuel elements, one FFCR and four control rods without fuel follower. Then, in early 2020 is expected that the Bandung TRIGA research reactor can be operated using the plate type fuel element if the TRIGA fuel elements are not produced anymore by the manufacturer.

REFERENCES

[1] Anonymous, Safety Analysis Report for Upgrade of TRIGA Mark II Reactor, General Atomic Document, San Diego, 1996.

[2] UMAR, E., KAMAJAYA, K., SUWONO, A., TANDIAN, N.P., HARDIANTO, T., “Steady State Thermal Hydraulics Analysis of TRIGA 2000 Research Reactor”, Proceedings of International Conference on Fluid and Thermal Energy Conversion 2003, Bali, Indonesia, 2003.

[3] RAHARDJO, P.R., YAZID, P.I., KAMAJAYA, K., ALFA, T., UMAR, E., SUDJATMI, K., SUBKI, H., “The Study on Core Sub-cooled Boiling on the Bandung TRIGA 2000 Reactor”, Proceedings of the Fourth Technical Meeting of the Coordinators for the IRSRR, Daejon, Korea, 2005.

[4] UMAR, E., KAMAJAYA, K., SUWONO, A., TANDIAN, N.P., HARDIANTO, T., “Theoretical Analysis of Thermal hydraulic Parameter of the TRIGA 2000 Reactor Using STAT Code”, Proc. Science and Nuclear Technology Seminar, Bandung, Indonesia, 2005. (In Indonesian).

[5] UMAR, E., KAMAJAYA, K., SUWONO, A., TANDIAN, N.P., HARDIANTO, T., “The Chimney Effect on Natural Circulation in the Coolant Channel of TRIGA 2000 Research Reactor”, Proceedings of International Seminar on Nuclear Safety 2004, Tokai-mura, Japan, 2004.

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Code on the Steady of Nuclear Research Reactors: Design, Safety Series 35-S1, IAEA,Vienna, 1992.

[7] ALFA, T., “The Neutronic and Power Distributions Calculations for TRIGA 2 MW Using WIMSD/ 4 and Citation Codes”, Proc. Science and Nuclear Technology Seminar, Bandung, Indonesia, pp. 86 – 97, 2000. (In Indonesian).

[8] PRASETYO BASUKI, PUTRANTO ILHAM YAZID, ZAKI SUUD, “Design Neutronic of Bandung TRIGA research reactor core using plate-type fuel elements” Proc. Science and Nuclear Technology Seminar 2013, Bandung, Indonesia, July 4, 2013. (In Indonesian)