INSPECTION OF THE TRIGA REACTOR TANK

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Abstract.

Nuclear components are under strict supervision of operators and safety authorities. The Reactor Centre of the Jožef Stefan Institute decided to make an inspection of its TRIGA Mark II research reactor to verify the conditions for long-term future operation within the on-going periodic safety review. Two main inspection methods were used: ultrasonic and visual inspection. Ultrasonic inspection was selected to prove that there is no significant reduction of wall thickness anywhere in the tank. The inspection confirmed that the reactor tank has not been degraded or corroded. In the future such inspection will take place every 10 years within the periodic safety review in order to monitor every 10 years the reactor tanks condition.

1. INTRODUCTION

Almost every piece of industrial equipment has a projected life-time from the very beginning of its operation. The life-time is normally defined on the basis of operational experience and is always determined in a conservative manner. On one hand, poor maintenance shortens the life-time and on the other hand, good maintenance and good operation can prolong the predicted period of individual component operability. Very important facts are also received from new research results in the field of aging of different materials.

Nuclear components are under strict supervision of operators and safety authorities. The Reactor Centre of the Jožef Stefan Institute (JSI) decided to make an inspection of its TRIGA Mark II research reactor to verify the conditions for long-term future operation within the on-going periodic safety review (PSR). Main information for PSR comes from the in-service inspection. In-service inspection contains a program of examinations, testing, and inspections to prove adequate safety and to manage deterioration and aging effects [1].

The inspection of the reactor tank was planned to be performed within the PSR from the beginning as the reactor tank is critical for normal and safe operation of the reactor. In addition it is the structure that is the most difficult to replace. The tank is made of aluminium and holds de-mineralized water under normal pressure at temperatures below 37 °C. It was not expected to be significantly degraded or corroded. In order to verify this assumption, the operator decided to perform detailed inspection of the reactor tank wall. Q Techna d.o.o. was selected to perform the task; mainly due to professional references on other similar nuclear installations (e.g. inspection of the nuclear power plant (NPP) Krško spent fuel pool).

Two main inspection methods were used: ultrasonic and visual inspection. Ultrasonic inspection was selected to prove that there is no significant reduction of wall thickness anywhere in the tank. Detailed visual inspection confirmed that there are no visually detectable defects like cracks or any other unacceptable surface defects. The main challenge of the inspection was that it had to be done under water from the inner side of the tank and, especially at the bottom of the tank, very close to a strong source of radiation, as the core was not removed during inspection. The challenge was met by selection and professional use of appropriate equipment and techniques.

Procedures, approach and the main findings are presented in this paper.

2. PERIODIC SAFETY REVIEW

The purpose of the PSR is to systematically review ageing effects, effects of various changes in the facility, operating experience, new developments in the field, changes in characteristics of the reactor site and all other possible effects on nuclear and radiation safety. In addition it should be proved that the reactor facility is still compliant with the newest safety standards, legislation and international recommendations. All this is needed to confirm that the reactor is at least as safe as at the beginning of operation and that it is capable of future safe operation.

The PSR programme of the JSI TRIGA reactor was prepared in compliance with the valid Slovenian legislation [2], practical guidelines prepared by the Slovenian Nuclear Safety Authority (SNSA) [3] and with the IAEA guidelines for the review of the research reactor safety [4]. In addition we used IAEA safety standards [5], [6] and [7]. The programme was approved by the SNSA in November 2011. The reference date was determined to be January 1st 2011. The estimated duration of the PSR was three years and the financial costs were estimated to 700,000 €. The most important task within the PSR was the inspection of the reactor tank, as it had never been inspected before. In addition this component is critical as it is the one which cannot be replaced easily.

3. CONSTRUCTION OF TRIGA REACTOR

1.1. General

The TRIGA Mark II research reactor at the JSI in Ljubljana, Slovenia was built in 1962-1966 and achieved first criticality on 31st of May 1966. It is a pool-type light water reactor with a annular graphite reflector and cooling by natural convection. The side view of the reactor is shown in Figure 1.

It is of essential importance to know and understand the construction of the TRIGA reactor when performing a PSR. This is the basis for evaluation of possible problems that could occur during the operation. The reactor is an open cylindrical vessel with a flat bottom end. It is 4870 mm high and 1982 mm in diameter. It is made from aluminum alloy 5052 H34. The minimum thickness of the vessel is 6.35mm. It was welded with fusion welding. All welds were inspected with radiographic examination (RT), with liquid penetrates and with bubble tests. The vessel as a whole was tested with a pressure test. The reactor is not stamped but it fulfils applicable portions of ASME Boiler & Pressure Vessel Code Section VIII requirements. The reactor tank during the construction is shown in Fig. 2.

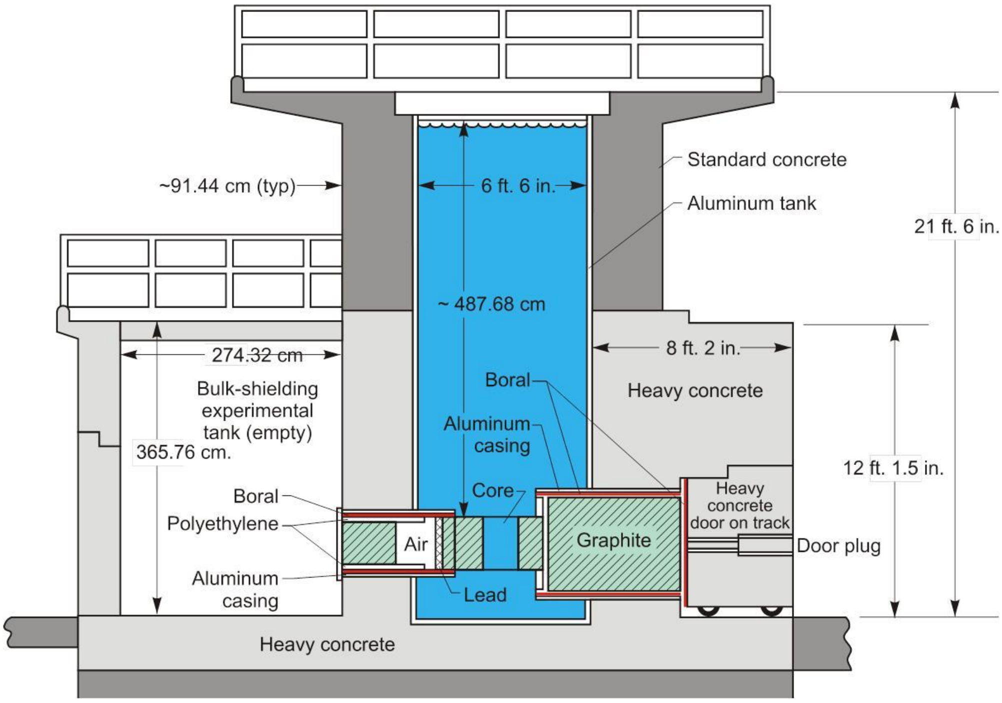

FIG. 1. Side view of TRIGA Mark II reactor.

The reactor tank is externally protected with two layers of bituminous #15 saturated felt wrapped around it. The tank is placed in the heavy concrete, and it is not accessible from the outer side.

The weight of the empty tank is 1080 kg and the weight of the reactor vessel filled with water is 18,780 kg. It was by designed by General Atomics and manufactured by Slovenian company Hidromontaža.

FIG. 2. Photos of the TRIGA reactor tank during the construction.

3.2. Analyses of possible degradation processes

Demineralized water has been used as the reactor coolant and for radiation protection since the very beginning. Aluminum alloys are resistant to this fluid and from a design point of view corrosion was not expected. But during the operation unexpected situation could occur like: an unintended change in water chemistry or contact with some other metals like stainless steel. These could provoke galvanic corrosion which is many times connected with submerged and embedded structures. In the case of the TRIGA reactor we have both situations. Since the structure is visible from the inner side larger degradation processes could be seen from the platform. Much more problematic is the embedded side. Galvanic corrosion is a local corrosion and could occur on a very small area that is not accessible or it is hidden. A typical area is the bottom of the reactor from the inner side, where a support construction for the reactor core is located.

Due to the non corrosive medium and properties of aluminum there is almost no possibility for general corrosion. But in the case that by accident some mercury comes in contact with the vessel, intergranular corrosion occurs. Mercury pollution of aluminum provokes severe irreversible degradation processes [8].

Special emphasis always has to be put on welds. Aluminum alloy 5052 H34, or with ISO designation AlMg2.5, contains 2.5% of magnesium as a principal alloying element. Such a material has good weldability. Since the welds were examined by RT during construction it is presumed that there are no unacceptable volumetric irregularities in the welds. But despite this aluminium is sensitive to lack of fusion which cannot always be detected with RT. Welds have different structure as a base material; residual stresses, irregularities and discontinuities etc., are present. For that reason degradation processes like cracks could occur in welds or in the heat affected zone.

3.3. Inspection methods

On the basis of analyses of possible degradation processes an inspection plan and scope of inspection was defined. It was foreseen that two main methods would be used:

- Visual inspection
- Ultrasonic inspection

Detailed visual inspection of all inner surfaces was performed. This included base materials, welds, bolting materials and surfaces of other internal components. The main purpose of this inspection was to detect possible degradation processes like corrosion, cracks and mechanical deformations.

Ultrasonic inspection gives information about processes from the outer side. If the wall thickness is not different from at the time of construction, this indicates well, that there are no corrosion processes from the outside. It is of essential importance that scanning is detailed enough, i.e. measuring points are not more than 500 mm apart.

Q Techna has much experience in in-service inspection on commercial NPP`s locally and abroad, but this was its first activity on a nuclear research reactor. Approach for inspection was the same as in NPP`s, meaning that all prerequisites had to be fulfilled especially from the points of nuclear safety and quality.

4. VISUAL INSPECTION

4.1. General

Visual inspection is the basic method to detect discontinuities, defects, degradation and similar undesired conditions or processes. It reveals a great deal of very important information and in many cases it is essential information for interpretation of results obtained by other methods. Of course it is desired to have a possibility for direct visual inspection, but in many cases that is not possible. Especially in the nuclear field remote visual inspection is used very often. The main obstacles for direct visual inspection are accessibility and radiation.

Visual inspection is very often underestimated, but it requires much theoretical knowledge, practical training with different materials and defects and a capability to use different equipment. It is essential to have skills to interpret images in a proper way.

4.2. Execution of visual inspection

In the case of the TRIGA reactor it was obvious that only remote visual inspection could be applicable. The reactor is filled with water continuously. For this reason it was decided to use a special underwater camera that could be used also very close to sources of radiation like fuel elements. For such an application charge coupled device (CCD) cameras could not be used.

Camera Mirion Technologies IST-REES R90 MK 3 CCU was used. It has built in an additional source of illumination. It is connected to the control unit with a cable. To operate the camera guiding tubes as a manipulator were prepared. The system was verified in laboratory with a performance demonstration.

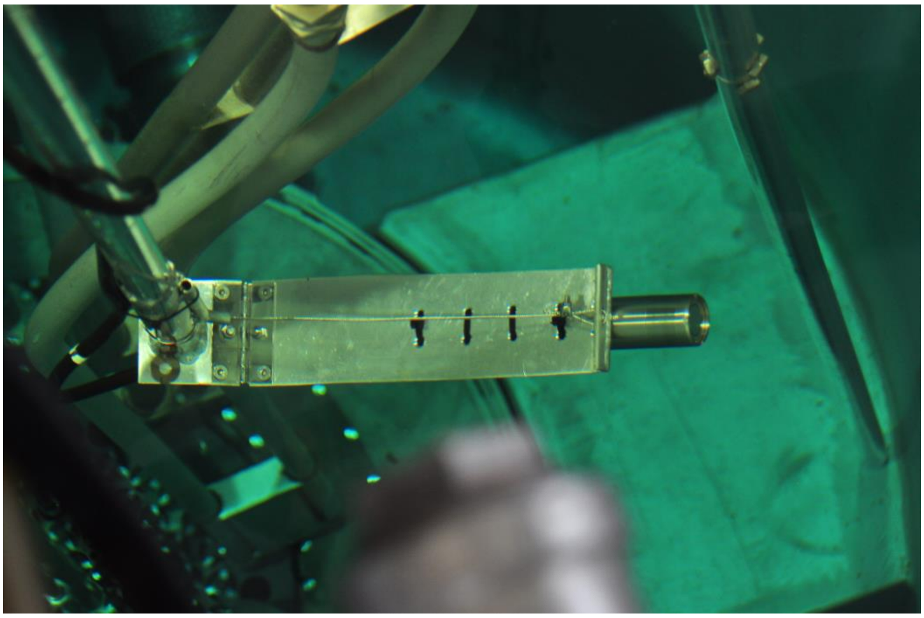

FIG. 3. Camera for visual inspection.

Before inspection began the whole system (camera-monitor-recorder-text generator) was properly calibrated on-site. The system was calibrated with a calibration chart (line with thickness of 0.8mm) recorded at a distance of 150mm, 500 mm and 1000 mm. The calibration procedure was documented and enclosed in the final report [9].

The camera did not have a fixed connection to the guiding tube. Connection permitted rotation of the camera in a vertical direction allowing for examination of surfaces from different angles. An operator was moved the tubes to position the camera and moved the camera`s angle with metallic rope. A picture of camera fixation is in Figure 3.

Inspection requires at a minimum three persons present on site. Two were responsible to manipulate the camera and one to interpret and evaluate. It was also essential that the coordinator was present at all times. As the reactor is used mainly for research and education, the reactor tank is full of various tubes, channels, fuel racks and other components that make the manipulation of the camera more demanding. For this reason the coordination of the camera path was needed.

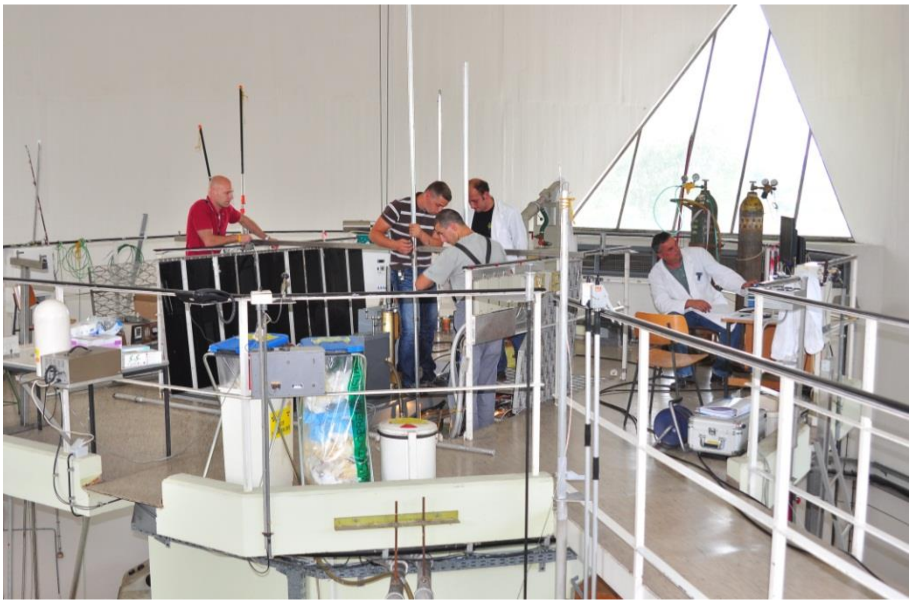

FIG. 4. Work on the platform during visual inspection.

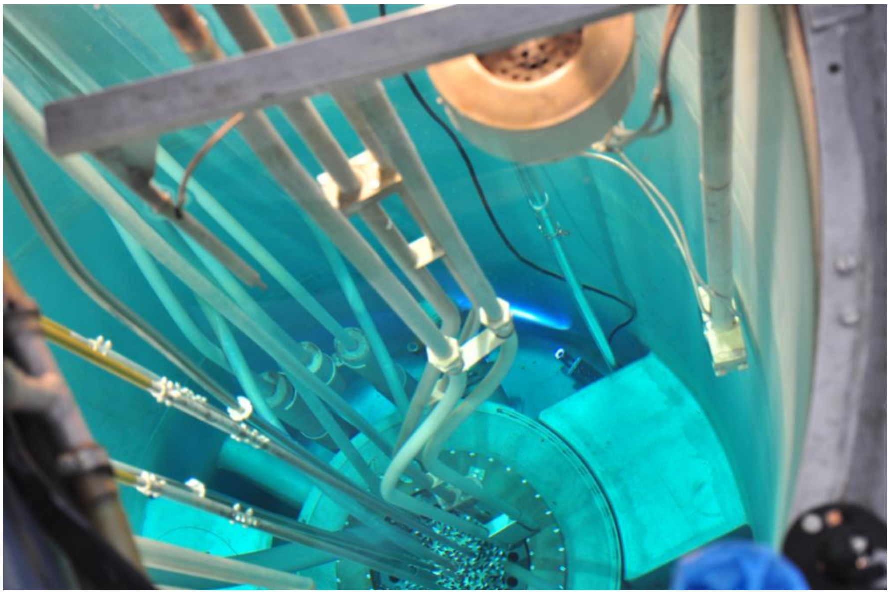

FIG. 5. Bottom of TRIGA reactors with different internals.

The scanning path was planned in the advance. Inspection was done at levels all around the tank. The camera does not permit a very far view and for this reason much scanning was needed. The upper part is not so complicated relatively for inspection because there are not so many components and structures. At the bottom of the tank the situation is completely different. There are guide tubes, irradiation channels, nuclear instrumentation, fuel element storage racks, the reactor core, thermalizing column, thermal column, pipes (thruports), etc. It can be seen in Figure 5 that the configuration at the bottom of the vessel is quite complicated for visual inspection.

There was a requirement that at a minimum 95% of all vessel surfaces were to be examined, which was fulfilled. Furthermore due to the construction of the manipulator 99% of the surface was inspected.

During the inspection no degradation processes like cracks or major corrosion areas were observed. There were also no other indications like mechanical damage due to the fall of heavy loads or collision with a sharp hard object. On the bottom of vessel some small foreign material and small local corroded areas were observed. It is assumed that corrosion areas appeared from foreign material which had been removed from vessel in the past.

All sections and positions were marked on recordings. This permits traceability for this inspection and also a possibility to compare recordings with new ones recorded in the future.

5. ULTRASONIC INSPECTION

5.1. General

Ultrasonic inspection was an additional method for visual inspection. In the case of any degradation processes it would be necessary to further investigate those areas.

In our case that was not necessary. The plan of inspection was prepared in advance. It was foreseen to measure the thickness of the vessel along eight verticals 45° apart. Measuring points were along a vertical line 300 to 500 mm apart. At least 20 measurements on each line on the bottom were performed.

5.2. Execution of ultrasonic inspection

Ultrasonic inspection was performed under water and required a specific approach. The underwater technique requires some different preconditions from inspection outside water. The procedure was basically prepared as an immersion ultrasonic technique. First of all the ultrasonic probe as well as cable connections must be water tight. The ultrasonic probe must be at a distance from the metal sheet and this distance must be maintained precisely during the entire measurement. Employment of water is as a contact mean and as a delay line that one element transducer can be used for thickness measuring. A very similar guiding tube was used for visual inspection. In addition this guide had a scale which showed vertical position. The ultrasonic probe was fixed to the guiding tube with a tilting connection. For the vertical part the probe was in one position; along the bottom the probe was turned in another position and another was selected along the transition radius between the vertical and bottom horizontal part. The ultrasonic probe and system of guidance can be seen in Figure 6.

For measuring the following equipment was used: ultrasonic equipment Krautkramer USN 58L and a probe K5K with 5mm vibrator in diameter and a nominal frequency 5MHz.

In laboratory’s performance demonstration was conducted with identical material that was used for construction of the vessel. Calibration was performed in equivalent conditions to those on site. The basic parameter for calibration and later thickness measurements taken on-site was the measuring distance from the water/metal interface echo to the first back wall echo using the ultrasonic system. In addition the repeatability of measurements was confirmed.

The measuring system was again calibrated on-site. Calibration was done with a step calibration block.

All personnel were appropriately trained in the same approach applied for visual inspection. Training was performed on a mock-up in laboratory.

Inspection on-site was done from the platform. Scanning was done along the same vertical lines as for visual inspection i.e. eight lines. Due to the geometry in the transition between the wall and bottom of the vessel it was not possible to maintain a constant distance to the water column. For this reason measurement was done between the first and second echo from the back wall.

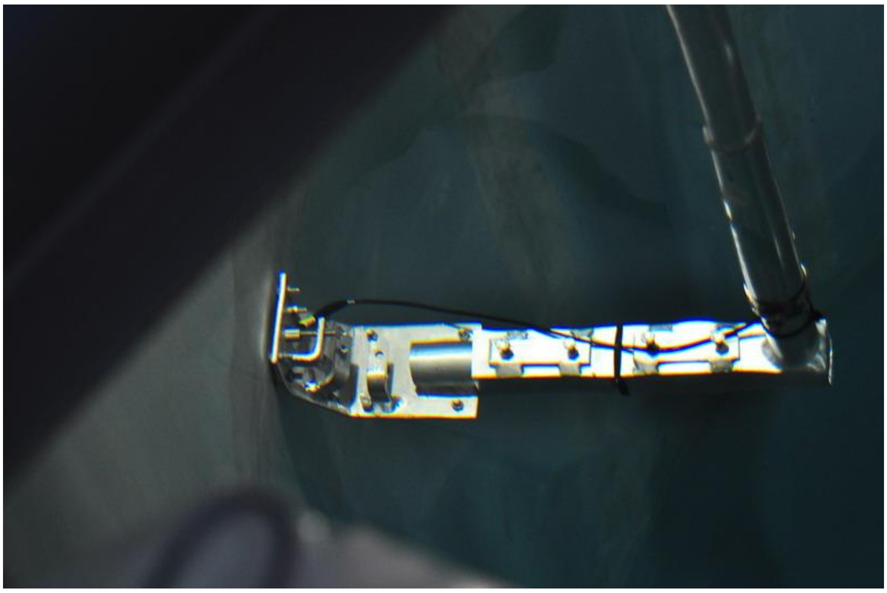

FIG. 6. Ultrasonic probe.

Locations where thickness was measured were positioned as a grid on the cylindrical and bottom parts. Thirteen measuring point were made in the longitudinal direction of cylindrical part 300 to 500mm apart, and eight divisions were made in the circumferential direction. The bottom was divided at four locations in the radial direction and the cylindrical part in eight divisions in circumference direction. Positions of measuring points were dependent sometimes on obstruction configuration in the vessel. Wall thicknesses of the cylindrical part were between 5.6 and 6.7mm and on the bottom 5.8 and 6.5mm, and no essential deviation from the nominal thickness was detected. Also in between the measuring points the thickness was scanned to discover any changes.

6. CONCLUSION

An appropriate method for visual and ultrasonic testing of the JSI TRIGA Mark II reactor vessel was successfully developed and applied.

Visually the vessel is in good condition, and there were also no indications that wall thickness has diminished.

All inspections performed show that there are no significant degradation processes taking place in the reactor tank.

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TRIGA RESEARCH REACTOR CALCULATIONS

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Abstract.

The first part of the paper describes typical computer codes for research reactor calculations, development of a computational model and its verification and validation. The majority of the paper is devoted to presenting various applications of research reactor calculations related to a typical 250 kW TRIGA type reactor, supported by numerous examples.

1. INTRODUCTION

The utilization of reactor calculations at research reactor started with appearance of cheap small computers in the late 1980s. It shows rapidly growing trend ever since [1]. Several centres have already passed the basic level and perform reactor calculations which are of practical value such as:

- detailed burnup determination,
- core design and fuel management,
- core optimization for production of particular isotopes,
- power distribution analysis of mixed cores,
- design of experiments,
- safety analyses,
- etc.

Reactor calculations have become standard part of the research and development programmes at research reactors and are often initialized at the same time with the construction of a new reactor. In this paper we present research reactor calculations, which can be performed by the staff of a small research reactor with the computer codes available through the IAEA, NEA and RSICC. A general overview of research reactor calculations has already been made by M. Ravnik in 1996 [1]. Since then the field of research reactor calculations has experienced important changes due to significantly increased computer power and access to it. Some applications of RR calculations that were once in a domain of big computer clusters can nowadays be performed on a personal computer. The purpose of this paper is to provide an update of the old paper and to give an overview of the modern applications of RR calculations.

The paper is organised as follows: in the first part we present typical computer codes for research reactor calculations, development of a computational model and its verification and validation. In the second part we present various applications supported by numerous examples of research reactor calculations related to a typical 250 kW TRIGA type reactor. The paper deals only with reactor neutronics codes and does not include thermal hydraulics calculations.

1.1. Computer codes for research reactor calculations

Practically, all standard reactor physics codes can be installed on modern PC computers without simplifications of the mathematical models.

Classification with respect to the practical requirements from the aspect of the user:

- physical model (adequacy of geometry, group structure, etc. for particular application),
- method of solution (complexity of the physical model),
- flexibility (application for non-standard problems),
- performance (running time, numerical accuracy, user friendliness, presentation of results).

In general the computer codes for research reactor core calculations can be roughly divided into two groups; deterministic and stochastic or Monte Carlo (MC). Deterministic codes are based on numerical or rarely analytical solving of neutron transport or diffusion equation. The computing errors are mostly systematic: uncertainties in the nuclear data, discretization of time-space-energy phase space and geometrical simplifications. Commonly they are computationally less demanding than stochastic methods. Stochastic or commonly refer to as Monte Carlo codes are based on Monte Carlo transport of neutrons (and other particles). They are capable of treating very complex three-dimensional configurations. Continuous treatment of energy, as well as space and angle eliminates discretization errors. The computing errors are systematic (uncertainties in the cross section data) and random which are inherently present due to stochastic nature of the method itself.

Generally MC codes are computationally more demanding than deterministic codes. However with appearance of faster computers and computer clusters, utilization of Monte Carlo methods is becoming much easier to perform. Typical representatives of the most common Monte Carlo neutron transport codes are, MCNP [4], SERPENT [4], KENO [6], TRIPOLI [7], OPENMC [8], MCBEND [9], MONK [10], PHITS [11], MVP [12].

Although Monte Carlo codes are extremely powerful, majority of them are still inferior to deterministic codes, when it comes to core design and fuel management. In addition many of the Monte Carlo codes feature poor user interface, making them a bit more difficult to start working with them. An example of a user friendly and fast running core management code is a deterministic diffusion code TRIGLAW [13], which features a graphical user interface [14] and is very easy to learn. Hence, it is used for education and training purposes by universities [15] and at international training courses on research reactors, such as EERRI training course [16].

It is interesting to note that the TRIGLAW code was developed by a team of researcher around the Slovenian TRIGA Mark II reactor at the Jozef Stefan Institute, who needed a core management tool for efficient reactor operation and utilization. This was the first step towards development of the NPP core design code CORD-2 [17], which has been used for the core design of the Slovenian NPP Krško (PWR, 700 MWe, Westinghouse) since the 1980s. The latter is a nice example of how knowledge and experience gained at research reactors can lead to their application at the power plants. Another application that was developed at the RR and then used and applied at the NPP was the digital reactivity meter [18].

1.2. Building a reactor model

After the user has defined the scope of ones calculations and the code to be used, one has to develop and build a computational model of the system (usually research reactor) under investigation.

The first step is to collect material, geometry data of the reactor. For certain application on has to collect operational data as well. This task is not trivial if we try to collect “as built” and not just typical or generic data of particular reactor. The set of data required for the calculation depends also on the computer code and the problems which are addressed to be solved. For example diffusion codes require only general reactor geometry and dimensions, Monte Carlo codes however require detailed geometry and materials.

All relevant geometry and material data should be in principle contained in the Safety Analysis Report (SAR) of the reactor. In practice, only part of those information can be found there. It is also not very reliable and accurate since the reactor description in SAR is often based on generic and not on “as built” data. The most reliable source of practical data is the design documentation of the reactor (plans, blueprints, drawings, fabrication specifications, etc.). It contains normally detailed data on geometry but only general data on material specifications.

The material data are normally found in internal reports of the reactor manufacturer or in general literature. Such data are, however, also mainly generic and normally approximately correspond to the particular case. The exception are the data about the enrichment and mass of uranium which are part of the safeguard documentation and are for this reason in details provided together with the fuel elements. The rest of the material data (e.g., material density, metallurgical composition in case of alloys, impurities important for neutrons, concentration of burnable poisons,) are normally not available for the particular reactor, especially, if the reactor is old. Developing a complete and consistent material and geometry database for reactor calculations of a particular reactor is therefore a tedious and time consuming task.

1.3. Verification and calidation OF RR calculations

The user of any computer code should not only know how the code works but has to be familiar also with the validity and the limitations of the code. Therefore, one has to verify and validate the code and the computational model by performing a comparison of the calculated results with the reference value. It is very common to use analytical solutions or results of Monte Carlo calculations as a reference for diffusion or transport calculations. The most rigorous approach, however, is to use benchmark experiments also called benchmarks as a reference.

The most comprehensive compilations of criticality and reactor physics experiments are the International Handbook of Evaluated Criticality Safety Benchmark Experiments [19], prepared within the International Criticality Safety Benchmark Evaluation Project (ICSBEP) Working Group [20] and the International Handbook of Evaluated Reactor Physics Benchmark Experiments [21] prepared within the International Reactor Physics Experiment Evaluation (IRPhE) Project [22]. Both compilations are updated with new experiments every year.

In addition, one has to validate also the input data, in our case nuclear data, used in the calculations.

2. APPLICATION OF RESEARCH REACTOR CALCULATIONS

The extent and scope of reactor calculations depends on particular needs and experience of the reactor staff [1]. However, main needs for reactor calculations stem from operational problems which are very similar for all research reactors. In the recent years the needs for calculational support of experiments and efficient utilization of research reactors have significantly increased.

There are many different research reactors around the world, each having its own purpose and set of applications. Information about world RRs are collected and updated by the IAEA and are available at the RR database [23]. One of the most common types of research reactors is a TRIGA type RR featuring inherently safe UZrH fuel.

In this paper we present some examples of applications of RR calculations on a TRIGA reactor that were performed by a small team of researchers gathered around the 250 kW TRIGA Mark II reactor at the Jozef Stefan Institute (JSI) in Ljubljana, Slovenia.

Until recently, there were practically no publicly available experiments of benchmark quality that could be used for verification and validation of the JSI TRIGA model. Hence, a number of well-defined and carefully designed experiments have been performed aimed at establishing a set of benchmarks for the TRIGA reactors. The performed experiments have been thoroughly analyzed and the experimental uncertainties evaluated using the most advanced Monte Carlo neutron transport codes such as MCNP, the Monte Carlo N-Particle Transport Code [4].

The criticality experiments carried out in 1991 have been thoroughly evaluated [25] and are now included in the International Criticality Safety Benchmark Evaluation Project (ICSBEP) handbook [24].

As TRIGA criticality calculations are very sensitive to Zr nuclear data, they were used to study the effect of changes in Zr cross sections from ENDF/B-VI to ENDF/B-VII nuclear data library. It has been shown that our computational model and the Monte Carlo method can well reproduce the multiplication factor of the TRIGA reactor as the differences between the calculated and experimental k_{eff} are within the experimental and calculational uncertainties. The major contribution to the uncertainty in calculated k_{eff} is the difference in evaluations of Zr cross sections, mainly in ^{90}Zr and ^{91}Zr, and in evaluations of thermal scattering data for H in ZrH and Zr in ZrH [26]. The effect of different evaluations on k_{eff} is up to 600 pcm for TRIGA core. However, it strongly depends on the size of the system and approaches zero for low leakage systems indicating that the problem lies in the evaluation of scattering cross section.

The criticality benchmark experiments were later followed by the neutron spectra and neutron flux distribution measurements to verify and validate these calculations as well [28]. The verification of neutron flux distribution was performed by comparing the calculated and measured for ^{27}Al(n,α)^{24}Na and ^{197}Au(n,γ)^{198}Au reaction rates in irradiation channels in the core centre, at the core periphery and in the graphite reflector surrounding the core (Fig. 1). The calculated and experimental normalized reaction rates in the core are in perfect agreement for both reactions indicating that the material and geometrical properties of the reactor core are well modelled (Fig. 2). Somewhat worse situation is in the reflector between the 30rd and 35th irradiation channel, where the calculated values slightly (~2σ) under predict the ^{197}Au(n,γ)^{198}Au reaction rate. The ^{27}Al(n,α)^{24}Na reaction rates, however, match very well (Fig. 3).

As our model well predicts the k_{eff} of the core and neutron flux distribution in the core it was then used for characterization of full core, that is neutron flux and spectra calculation in all irradiation facilities of the reactor [28] in order to support experimental campaigns.

Moreover, it was used also for calculation of various core parameters, such as power peaking factors [29] and kinetic parameters calculations [30]. The results of the analyses and the approach are general and can be applied to other TRIGA reactor. Later the analyses were repeated with the TRIPOLI Monte Carlo code [31], which uses slightly different algorithms than MCNP.

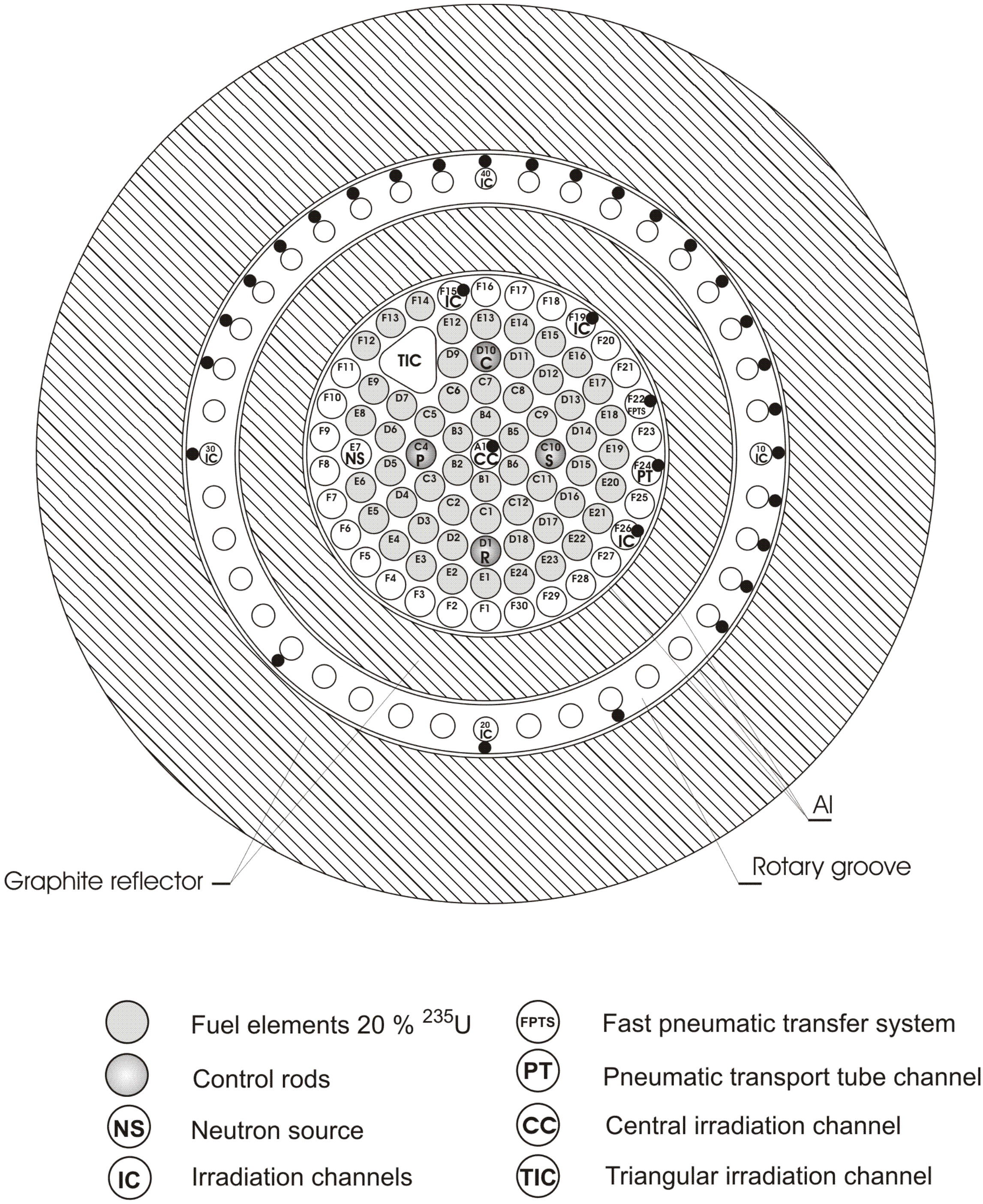

FIG. 1. Schematic top view of the TRIGA reactor. The locations of aluminum-gold foils used in the experiment are marked with black dots.

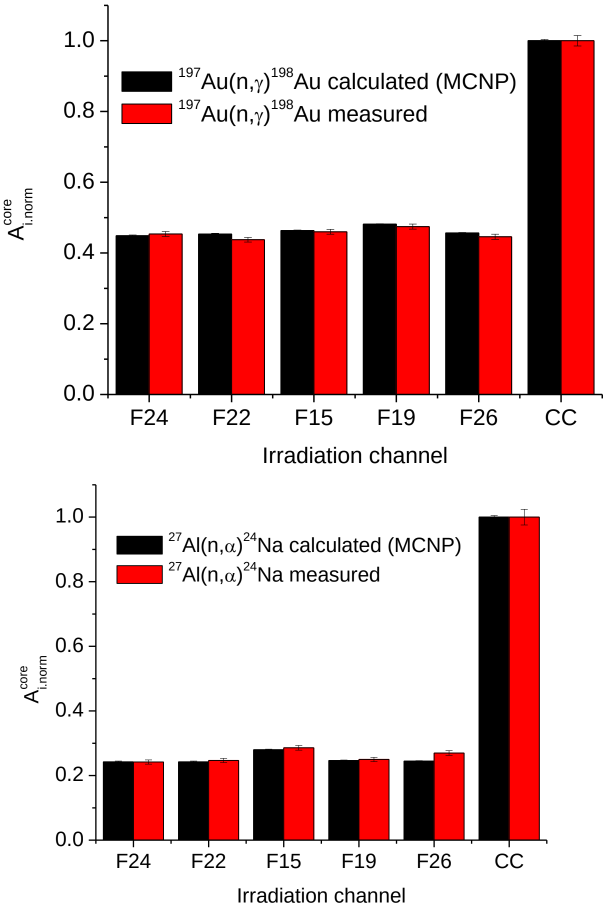

FIG. 2. Calculated and measured $^{197}Au(n,\gamma)^{198}Au$ and $^{27}Al(n,\alpha)^{24}Na$ reaction rates in the reactor core normalized to central channel (CC). The error bars represent 1σ uncertainty.

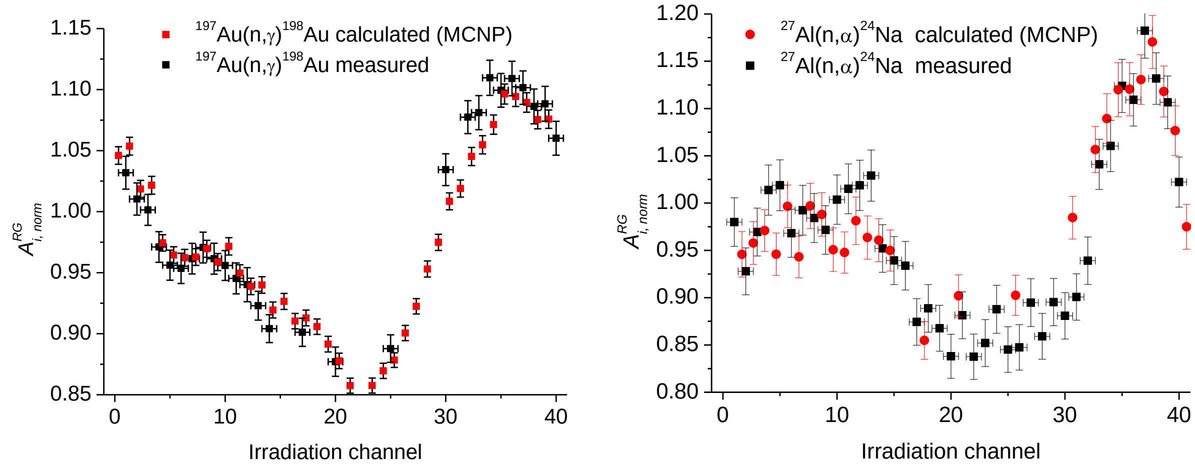

FIG. 3. Calculated and measured $^{197}Au(n,\gamma)^{198}Au$ and $^{27}Al(n,\alpha)^{24}Na$ reaction rates in the carrousel facility. The Y error bars represent 1σ uncertainty in measured or calculated results. The X error bars represent the uncertainty in the irradiation channel position during the experiment.

More comprehensive experiments were performed after 2010, when collaboration was established with the French Alternative Energies and Atomic Energy Commission (CEA). One of the objectives was to analyze and improve the power calibration process of the JSI TRIGA reactor (procedural improvement and uncertainty reduction) by applying a combination of absolutely calibrated CEA fission chambers and the calculations. By using a combination of measurements and calculations we managed to reduce the uncertainty in the reactor absolute power from 15% to less than 5% [32]. This is very important achievement as practically all RR calculations should be normalized to the reactor power in order to get absolute values of the calculated quantities [33], such as neutron fluxes, reaction rates and dose rates. One of the largest contributors to the uncertainty in the measured reactor power was the tilt in neutron flux due to control rods. RR calculations were proposed to eliminate or reduce this uncertainty [34].

One aim of the collaboration with the CEA was also to apply the TRIGA irradiation facilities for irradiation campaign of new activation dosimeters and neutron/gamma flux measurement devices recently developed by the CEA, which would help improve the nuclear data for neutron dosimetry and spectrum calibration. The experiments carried out provided a unique opportunity to compare measurement and calculation results for a pool type reactor and thus help validate the calculation tools and models developed and used at the JSI for neutron transport calculations in the TRIGA reactor. Within these activities the axial reaction rate profiles were measured and calculated with Au foils [35] and miniature fission and ionization chambers [36, 37]. Profiles of axial fission rate at various positions in the core are presented in Fig. 4.

All of the above activities served among others for the purpose of verification and validation of the computational model, which was then applied to various other applications.

It is interesting to note that the experimental equipment used in the above mentioned activities was later used for new practical exercises at the JSI TRIGA [38]. Additionally, the results of Monte Carlo simulations (neutron flux and fission rate 3D profiles) were visualized and used as an advanced tool for teaching reactor physics [39].

The RR calculations are being more and more used for computational support to complex experiments, ranging from improvement of self-shielding factors for neutron activation analysis [40, 41] to fusion applications such as T production for future fusion reactors [42] and feasibility studies of installation of thermal to 14 MeV neutron converters into the reactor [43].

RR calculations are becoming more and more common in various safety evaluations, from introduction of new experiments [45] to accident scenarios [46, 47].

Apart from RR utilization the RR calculations are indispensable part of the end of the reactor lifetime as they are commonly used to estimate the activation of individual components, including the biological shield [48]. By using a combination of measurements and calculations one can significantly reduce the amount of radioactive waste after the decommissioning and make their classification easier.

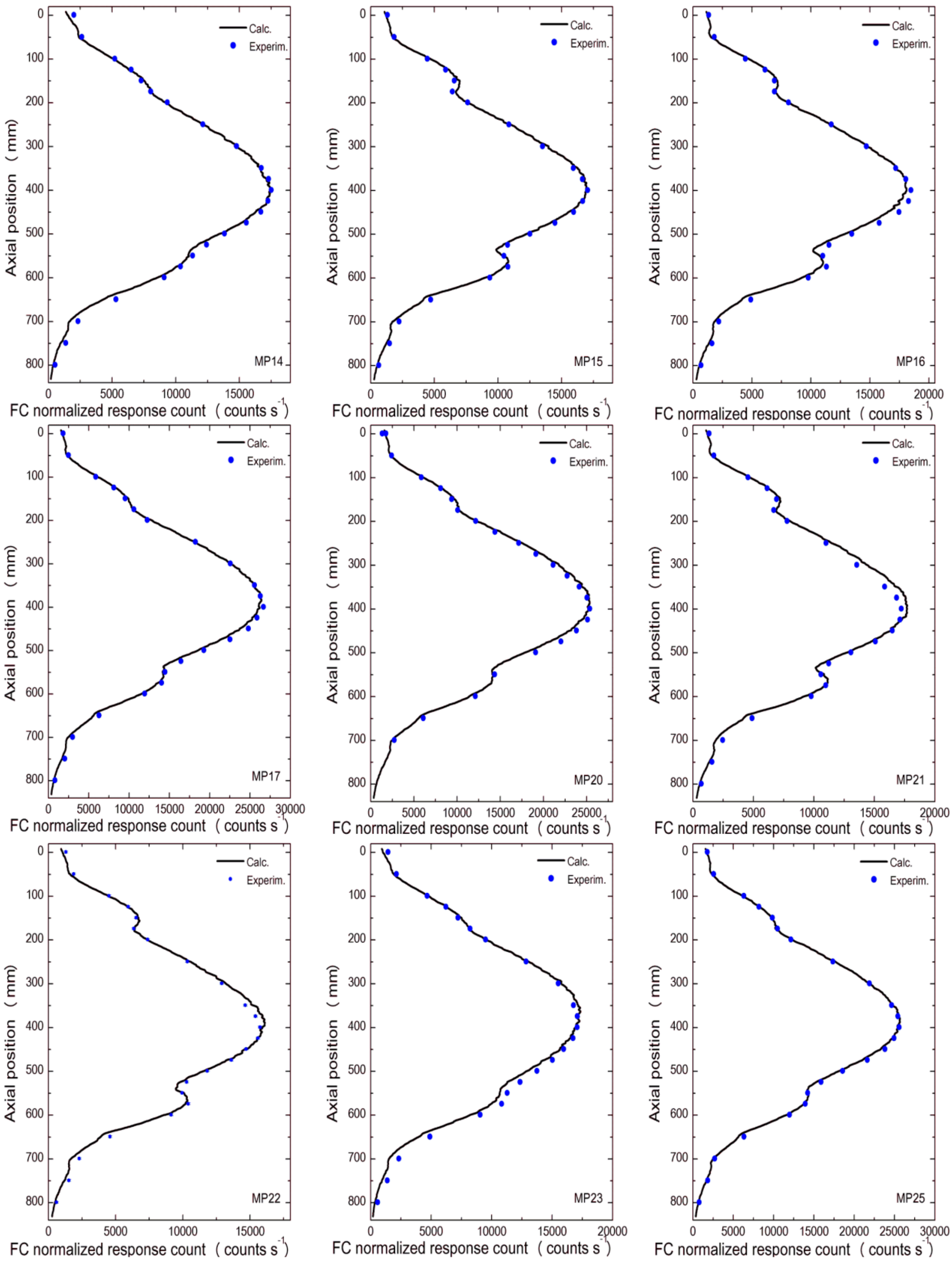

FIG. 4. Calculated and measured axial fission rate distribution at various positions in the JSI TRIGA Mark II reactor core. Dots present the measured fission rates and the line presents the sliding average (over active height of the FC, i.e. 4 mm) of the calculated fission rates. Statistical error of calculated values is under 1 %.

3. CONCLUSIONS

The RR calculations have become indispensable part of safe and efficient reactor utilization not requiring significant financial resources. This is confirmed by a large number and variety of examples of RR calculation utilization, performed by a relatively small team of researchers working at the Jozef Stefan Institute's TRIGA Mark II reactor. A small and relatively old research reactor having rather low neutron flux may even nowadays efficiently support both fundamental and applied research in theoretical as well as experimental reactor physics and significantly contribute to preservation of knowledge on nuclear science and engineering.

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THE CORE CONVERSION OF THE TRIGA REACTOR VIENNA

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Abstract.

The TRIGA Reactor Vienna has operated for many years with a mixed core using Al-clad and stainless-steel (SST) clad low enriched uranium (LEU) fuel and a few SST high enriched uranium (HEU) fuel elements. In view of the US spent fuel return program, the average age of these fuel elements and the Austrian position not to store any spent nuclear fuel on its territory, negotiation started in April 2011 with the US Department of Energy (DOE) and the International Atomic Energy Agency (IAEA). The sensitive subject was to return the old TRIGA fuel and to find a solution for a possible continuation of reactor operation for the next decades. As the TRIGA Vienna is the closest nuclear facility to the IAEA headquarters, high interest existed at the IAEA to have an operating research reactor nearby, as historically close cooperation exists between the IAEA and the Atominstitut. Negotiation started before summer 2011 between the involved Austrian ministries, the IAEA and the US DOE leading to the following solution: Austria will return 91 spent fuel elements to the Idaho National Laboratory (INL) while INL offers 77 very low burnt SST clad LEU elements for further reactor operation of the TRIGA reactor Vienna. The titles of these 77 new fuel elements will be transferred to Euratom in accordance with Article 86 of the Euratom-US Treaty. The fuel exchange with the old core returned to the INL, and the new core transferred to Vienna was carried out in one shipment in late 2012 through the ports of Koper/Slovenia and Trieste/Italy.

This paper describes the administrative, logistic and technical preparations of the fuel exchange being unique world-wide and first of its kind between Austria and the USA performed successfully in early November 2012.

4. INTRODUCTION

The TRIGA Vienna reactor was one of the last remaining TRIGA reactors which still in use High Enriched Uranium (HEU) fuel. Due to the US Fuel Return Program, the Vienna University of Technology/ATI was obliged to return its 9 HEU FE(s) by 2016. Moreover, most of the 102-types LEU FE(s) in the current core were close to reach their maximum burn-up values. Therefore, it was also an option to return these high burned FE(s) along with the HEU fuel. The fuel inventory at the institute before shipment can be seen in the next two tables (Table 1-2).

TABLE 1. FUEL INVENTORY AS PER 01.08.2012

Date	Number of fuel elements	Type	Remarks
05.12.61	+66	Al, 20%	2 instrumented fuel elements
07.07.62	-2 (retour)	Al, 20%	
19.02.65	+2	Al, 20%	
02.08.66	+3	SST, 20%	
21.10.68	+3	SST, 20%	1 instrumented fuel element 5284 TCE
19.10.72	+9	SST, 70%	
02.12.80	+1	SST, 20%	1 instrumented fuel element 8257 TCE
09.08.82	+3	SST, 20%	
15.02.83	+2	SST, 20%	2 instrumented fuel elements 8730, 8731 TCE
21.08.87	+3	SST, 20%	
19.10.88	+3	SST, 20%	
01.02.90	+3	SST, 20%	
14.12.00	+8	SST, 20%	
Total	104		

TABLE 2. FUEL ELEMENT SITUATION AS PER 01.08.2012

Number of FE	Location	Al Cladding	SST Cladding	Enrichment	Remarks
83	core	54	29	75 FE 20% 8 FE 70%	2 instr. FE
4	Storage pits	3	1	3 FE 20% 1 FE 70%	
8	Fresh fuel storage	-	8	20%	2 instr. FE
8	Spent fuel storage	8	-	20%	1 instr. FE
1	Hot storage facility	1	-	20%	Cut into 2 pieces
Total: 104		66	38		

From the 104 FE(s) at the institute it was decided to keep the eight fresh FE(s) and the five FE(s) with the lowest burn-up in the core. Therefore, 91 spent fuel elements were prepared for shipment to the INL.

5. TECHNICAL PREPARATIONS

To be compliant with the NAC-LWT Certificate of Compliance (CoC) for packaging and transport of TRIGA fuel (the fuel to be shipped must cool down for 90 days prior to transport), the reactor was shut down on April 27, 2012. With more than 6 months cooling time prior to the scheduled shipment date all radiation levels are expected to be below the allowed limits. Till April 27, 2012, the TRIGA Mark II reactor Vienna had 10257 days (8 hour working days) of operation. Since the first criticality on March 7, 1962, 12318 MW(th) of energy has been produced.

5.1. Fuel inspection Vienna

On June 9, 2012 staff from the INL travelled to the Atominstitut Vienna, to perform the fuel examination. Ninety-one (91) TRIGA fuel elements were examined at our facility. The elements consisted of sixty-six (66) aluminium clad and twenty-five (25) stainless steel clad TRIGA fuel elements. Of the 25 SST clad elements, nine were high enriched uranium (HEU) TRIGA Fuel Life Improvement Program (FLIP) fuel elements. The elements were inspected to facilitate the preparation of the required documentation for storage at the Idaho Nuclear Technology and Engineering Center (INTEC) in the Irradiated Fuel Storage Facility (IFSF).

Each element was visually examined over 100% of its surface. All but five (5) of the fuel element's identification/serial number were verified using an underwater camera system. Four (4) of the five (5) were verified using a finger camera and the fifth one was done through a lead glass window in our hot cell.

Eighty-one (81) of the elements appeared to be in a condition that would allow them to be shipped to the US without further packaging. The inspection revealed mainly light scratches, dings, scrapes, small dents and scuffing which is common due to handling during removal and insertion into the core. Ten (10) fuel elements appeared to have a damaged cladding. This ten fuel elements required to be placed in sealed failed fuel cans prior to placement in the NAC basket and cask for shipment and storage in Idaho.

2.2. Fuel inspection Idaho

From August 27 till September 14, 2012, experts from the Atominstitut performed an optical inspection of very low burnt 104-types SST clad LEU elements stored at the INL. Out of a list of one hundred and twenty (120) fuel elements, seventy-seven (77) have been chosen. Seventy-five (75) FE(s) were chosen from the former TRIGA reactor in Musashi, Japan, and two (2) FE(s) from the former TRIGA reactor in Cornell, USA. Each element was visually examined over 100% of the surface through the window of a hot cell. The entire examination was recorded to DVD including a backup DVD recording. Significant scratches or dents were the main criteria to reject two FE(s) from the facility in Musashi. The average burn-up of the chosen 77 FE(s) is below 1%.

6. SITE ASSESSMENT

A meeting was held at the Atominstitut Vienna facility on Friday, March 24, 2012 to discover and discuss the processes, possible methodologies and particularities of handling, unloading and loading the NAC-LWT cask at site with the irradiated TRIGA nuclear fuel inventory at the TRIGA Mark II research reactor.

The company NAC International (NAC) has been awarded a contract from the Vienna University of Technology (VUT) to perform packaging and transport of irradiated TRIGA fuel from the DOE's Idaho National Laboratory (INL) to the Atominstitut Vienna. There to exchange this fuel with the spent fuel inventory currently at the Atominstitut Vienna and to package and transport the spent fuel to INL.

During this site assessment, the outside work area where all the NAC equipment had to be placed and the existing infrastructure outside the building was evaluated. As an output of this evaluation, addition external light was installed to guarantee compressed air during the fuel exchange an external diesel-powered compressor was rented. The outside door was found to be large enough to accommodate the movement of the NAC Intermediate Transfer System (ITS) container through a standard 5 to 10 ton fork lift. The assessment of the inside work area revealed that there is enough floor space and that the ground floor is adequate for handling the load of the fork lift. The capacity of the crane and the 7.7 m of height clearance from crane full up to the reactor platform also fulfilled all requirements. Afterwards the reactor and the fuel storage racks were inspected. All the necessary criteria like water clarity, water purification, water radioactivity and working space were reached. To handle and load the 10 damaged FE(s) a shielding with concrete blocks and a working procedure was developed. At the end of the assessment a detailed list of additional infrastructure supplied by the institute and an assignment of skills between the institute and NAC was developed.

The following support equipment and plant services were defined to guarantee the cask loading.

To be supplied by TU Wien:

- Two 2-man scissor lifts to support cask operations outside;
- Large 10 ton fork lift;
- 80 ton (minimum) mobile crane;
- 3 - 5 ton fork lift;
- Electrical service (220 VAC, 50 Hz);
- Electrical power extension cords and plug adapters;
- Electrical connection for NAC electric chain hoist, 380 VAC, 3 phase;

- Clean compressed air (100 psig, 80 cfm) available outside at cask loading area;
- Helium gas, Grade 4 minimum (99.99 %) with high/low pressure regulator;
- De-min water;
- Shield cave for loading and testing of TRIGA Sealed Failed Fuel Cans;
- Plastic sheeting and bags;
- Decontamination materials;
- RAD waste disposal;
- Four bags of clean sand to assist with cask base plate levelling;
- Site approved solvent (i.e. denatured alcohol) to clean the cask seal area.

Skill sets needed:

- Security – TU Wien;
- Rad Protection - TU Wien;
- Reactor Operators – TU Wien;
- Crane operator and fork lift operator - TU Wien;
- Transportation Specialists – NAC;
- Certified Cask Operators – NAC;
- Helium Leak Test Examiner – NAC.

Afterwards, a detailed project plan covering the project description, the participating organizations, the project interfaces, the quality assurance, the project administration and the task plans were developed by NAC. To control the progress and to avoid possible delays a punch list was developed. During a weekly telephone conference with all the participating organizations, the progress punch list and the on-going progress was discussed.

After all this information a detailed working procedure for the fuel exchange was developed by the Atominstitut. This working procedure was evaluated and approved by the Regulatory Body. After the project was finished, a detailed radiation protection report for the exchange of the reactor core at the TRIGA research reactor in Vienna was submitted to the government.

7. ADMINISTRATIVE PREPARATONS

The Vienna University of Technology (VUT), as the owner of the fuel, signed two contracts. One contract covered the supply of the new fuel, and the other one covered the shipment of the 91 spent fuel elements plus one Pu-Be neutron source. Beside the DOE and the VUT also the European Supply Agency (ESA) had to sign the contracts.

Beside the agreement of the ESA concerning the content of the two contracts, Euratom had to be informed by the Federal Ministry of Science and Research. The Council Directive 2006/117/Euratom of November 20, 2006 on the supervision and control of shipments of radioactive waste and spent fuel, covers shipments of radioactive waste or spent fuel which have a point of departure, transit or destination in an EU Member State if the quantity or concentration are over certain limits fixed by Directive 96/29/Euratom, Article 3(2)(a) and (b). After the process was finished, the Ministry issued the license to the University to ship back the spent fuel and the source to the United States.

In parallel, the Euratom safeguard division was informed about the physical inventory leaving and entering Austria with Annex VI and Annex VII reports using the enmas light format.

- The Federal Ministry of Economy, Family and Youth had to issue an export license for components with dual use.
- The Federal Ministry for Transport, Innovation and Technology had to validate the NAC-LWT cask for the use in Austria.
- The Federal Ministry of Interior had to issue a license to cover the additional amount of U-235 stored at the institute during the time of the fuel exchange. This license forced the institute to increase the physical protection during this time.

To get the authorization to ship the fuel from Vienna, Austria, to the INL, Required Shippers Data (RSD) forms 434.28, 434.28A and 434.28B had to be submitted to the project manager of the INTEC fuel group. In addition, the form 434.30, the proposed shipment content (PSC) has to be submitted.

The shipment of the Atominstitut Vienna fuel was joined with irradiated nuclear material from Italy in the Adriatic Sea. As a result, additional documents for the Italian authorities concerning the spent fuel composition had to be prepared as well as coordination with the Slovenian authorities to use a Slovenian ocean port. To fulfil the requirements of the Slovenian authorities, additional nuclear liability insurance during the cargo transits Slovenian territory was put into force.

8. FUEL EXCHANGE

The fuel exchange was performed based on standard procedures developed by NAC during the last years and IAEA guidelines [1], [2]. For the transportation of fuel on the road and over sea the NAC-LWT cask was used. This cask with a weight of 22.4 tons, could host up to 120 TRIGA fuel elements in 5 fuel baskets. The transfer of the 5 fuel baskets was performed using the NAC-LWT Dry Transfer System (DTS). The DTS was connected with a shield gate adapter to the LWT cask. The handling of the DTS was performed with a 120 t crane. To transfer the fuel in the reactor pool an ITS with an inner and outer shield was used. The transfer of the inner shield in the reactor hall was performed by a 10 t fork lift. In the reactor hall the inner shield was transferred with an overhead crane under water in the reactor pool. Once the inner shield was in the pool, the shield was opened and the fuel was transferred one by one out of the shield into the underwater storage racks.

The transfer of the spent fuel was performed the opposite way. Detailed photo documentation can be found in the corresponding presentation.

The DOE transferred title of the supplied seventy seven (77) LEU Fuel Elements to the Vienna University of Technology upon delivery. The University afterwards transferred the title of these 77 new fuel elements to Euratom in accordance with Article 86 of the Euratom-US Treaty.

9. TRANSPORT

Transport of the DOE supplied LEU Fuel Elements and the irradiated HEU and LEU fuel from VUT spanned Austria, Slovenia, Italy, the global commons over international waters and the United States. Appropriate transportation and security plans were developed and implemented. The ground transport was carried out with appropriate transport companies and with significant security management and oversight. Security responsibility transition between security forces was well managed and occurred seamlessly. In addition to the security management provided by the ocean-going vessel, the Italian government maintained security surveillance of the vessel during its transit in the Adriatic Sea. Then the U.S. Coast Guard provides security escort when the vessel is in or near U.S. territorial waters.

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