Operation and Exploitation of the RPI
Reactor Calculations
RPI Dosimetry

The central objective of the operation and exploitation of the reactor is to be able to satisfy the users needs while conducting all activities with the assurance that a highly competent and motivated staff operates the equipment in a safe and reliable manner. The implementation of such objectives demands a variety of projects some of which are repetitive in objective and variable in content others addressing specific aspects of the same end situation. Safety being a permanent consideration in all work around the reactor interlinks all the activities being or to be performed. The main set of projects, actual and coming, in which the staff is involved is presented below.

At present the RPI operates with its second core which is of fully enriched uranium. The first core which was of 20% enriched uranium was last used in 1987 when it was unloaded and stored in the reactor pool. An agreement for the acceptance of all the spent fuel of US origin was signed with the US during the year. After careful examination and control the fuel of the first loading was returned to the US in August. The transport cask was loaded in a station built outside the reactor hall as there was no capability to perform this operation inside the reactor building. The movements of the fuel from the reactor pool to the transport cask where made with a transfer cask moved securely fasted to a forklift truck.

In the context of foreseeing the behaviour of electronic components and circuits to be used at the LHC facility at CERN, simulations of the expected exposure, for a 10 years period, were performed by irradiating those materials in a special container shielded for thermal neutrons and placed on the top of the reactor grid plate at some 9 cm from the West face of the reactor. To simulate the relatively low dose rate in which the equipment will be used it was required to operate the reactor at reduced power. As this mode of operation inhibits the execution of experiments requiring the full flux that the reactor can provide, a beam tube is being prepared to pursue the irradiations program with the reactor running at full power. The design of the tube is in advanced status with the safety analysis report under preparation and the expectation that it can be implemented in the first quarter of the current year. It is required that the beam has a dominant fast neutron component, with low gamma contamination. Therefore the beam is filtered with a thermal neutron absorber and a gamma radiation shield. The design allows for a rapid change of the thickness of the lead shield obviously at the cost of the reduction of the intensity of the fast neutron beam. To better determine the characteristics and intensity of the flux available, neutron flux and gamma dose rates have been measured inside the tube. A reduction of the beam intensity can be achieved through the distance to the core of the reactor of thickness of the absorbers introduced. However, increasing its intensity is more demanding. For this purpose the introduction of an Al reflector, partially replacing the Be reflector, is being studied and designed as is indicated bellow.

Isotope production is an important activity for the reactor utilisation. Effort is being placed in the production of short-lived isotopes which could be delivered rapidly to the users in the country. As the potential for production is reasonable, this activity could advantageously replace some imports. To answer the questions that such activity will imply improvements in handling and irradiation of materials are being considered and have to be implemented.

The operation of the reactor requires updated reactor physics calculations of parameters such as effective multiplication, reactivity margins, control rod worth and fuel consumption. This activity which was drastically reduced with the retirement of Dr. Ramalho Carlos (we record with very deep sorrow his passing away only months after his retirement) has been resumed with vigour. The current work is addressing, other than current situations, the use of specific reflectors at specific places to locally enhance the fast neutron flux. In the coming future it will also address 3D calculations, to increase the detail of the information that can be extracted, and the use of more selected programmes for specific objectives and also to other aspects, particularly, thermohydraulics.

The creation of an epithermal neutron beam in one of the beam ports of the reactor is in progress. This beam is designed for multiple applications and for this reason is provided with two openings with diameters of 2 and 5 cm, and with the possibility of leaving or eliminating the thermal component of the neutron spectrum with a Cd filter. At present it is envisaged that the beam will be used, inter alia, in the context of the "BNCT Activities" project, elsewhere referred and for the determination of hydrogen in heavy metals.

The activity developed under the general designation of RPI Dosimetry is closely related with the work performed around the reactor. It has to address the characterisation of the irradiation facilities, control experiments and to obtain values for the validation of the calculated values, only to refer some examples. Besides these objectives, dealing with an optimised exploitation of the reactor, an important goal is also the implementation of specific dosimetric procedures. Work is being developed in the context of the "BNCT Activities" project, partly financed by an European contract, aiming at the establishment of a "Code of Practice for BNCT Dosimetry". The epithermal neutron beam referred above is one of the irradiation facilities to be used in the intercomparison of methodologies. It is worth mentioning that the ITN, through the RPI team, is one of the partners of a proposal submitted to the 5th Framework Program on "Infrastructural and Technical Support to Implement Clinical and Pre-clinical Programmes Throughout Europe".

Fast Neutron Irradiation of Electronic Circuits for the LHC/CERN

A.J.G. Ramalho, J.G. Marques, I.C. Gonçalves, A.P. Fernandes[1], I.F. Gonçalves, A. Vieira[2], M.J. Prata[1], F.M. Cardeira, J.A. Agapito[3], J.P. Santos[1], P. Gomes[4], J. Casas[4]

Objectives

The operation of the Large Hadron Collider (LHC) at CERN will require the use of electronic components in a weak field of fast neutrons, with an energy similar to the average energy of the thermal neutron fission. The circuits are expected to receive a neutron fluence of the order of 5×10^{13} n·cm^{-2} during a 10 year period. To simulate these conditions, irradiations with fast neutrons were performed at the Portuguese Research Reactor (RPI).

Results

A dedicated irradiation device was built to perform the irradiation in the RPI's pool due to the significant volume occupied by the materials to be irradiated. The irradiation device consists mainly of a 30 cm long cylindrical Al container, with 15 cm diameter, lined internally by 0.1 cm Cd and surrounded by a 2 cm thick Pb shield [1]. It rests on the core grid plate, starting at about 10 cm from the core and going to the edge of the grid. The bottom part of the handling frame of the irradiation device is provided with holes, through which the pins used to fix the fuel can penetrate, thereby assuring that the device does not slide and can be positioned in a reproducible way.

The material to be irradiated included full circuits and components placed in up to 11 assembling boxes. The circuits and components were continuously monitored during the irradiation. A 40 m long, 100-conductor cable was used to connect the boards and components to the on-line monitoring equipment installed near the reactor's pool. The reactor was operated at a reduced power of 2 kW so that the central circuit would receive a fluence of 5×10^{13} n.cm^{-2} over a period of 60 hours. Three campaigns have been performed so far, measuring passive and active components: precision and standard thin film resistors, different types of capacitors, discrete active components and operational amplifiers from several manufacturers. On-line measurements were performed before, during and after irradiation and stand-by periods, to evaluate the neutron and gamma irradiation damages as well as possible annealing effects. Integration dosimeters placed on each assembling box have revealed a total γ dose in the 3.5-17 kGy range, which is higher than expected for the LHC. A good behavior was found for precision resistors and capacitors, while the behavior of operational amplifiers depended strongly on their manufacturing technology. A detailed description of the obtained results can be found in ref. [2].

References

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2. Agapito, J.A., Cardeira, F.M., Casas, J., Duarte, A., Fernandes, A.P., Franco, F.J., Gil, M.J., Gomes, P., Gonçalves, I.C., Hernández-Cachero, A., Jordung, U., Martín, M.A., Marques, J.G., Paz, A., Ramalho, A.J.G., Rodríguez Ruiz, M.A., Santos, J.P., Preliminary Test for Radiation Tolerant Electronic Components for the LHC Cryogenic System, *Proc. 5th Workshop on Electronics for LHC Experiments,* Snowmass, 1999, in press.

Further work

Experience has shown that it is desirable to perform the irradiation with the reactor operating at 1 MW, thereby allowing the access to other users and also that the γ field should be decreased. Both objectives can be achieved using a dedicated facility built around a beam tube, where it is easier to modulate the neutron and γ fields. The proposal for such a facility has been recently funded (contract CERN/P/FIS/15181/1999) and the facility will be built during 2000.

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Examination of spent fuel of the Portuguese Research Reactor

A.J.G. Ramalho, J.G. Marques, F.M. Cardeira

Objectives

The main objective of this work was to conduct a detailed examination of the fuel assemblies from the first core of the Portuguese Research Reactor, in preparation for their return to the USA under the "Reduced Enrichment for Research and Test Reactors" program. The tests centered on a visual inspection of the fuel assemblies and a determination of the fission product leakage.

Results

Thirty nine spent fuel assemblies (enriched to 19.8% ^{235}U) from the initial core of Portuguese Research Reactor were visually inspected for corrosion and sipped for determination of fission product leakage. For a visual inspection all assemblies were individually transferred to an examination station located 2 m deep in the water and photographs were taken using a digital camera. Once the visual examination was completed, the fuel assembly was placed in a dedicated washing and sipping system. This system, detailed in ref. [1], consisted essentially of a 3 m long aluminum pipe which could be moved up and down along guides fixed on the reactor's pool. The procedure was to place the assembly inside the pipe, isolate the pipe from the pool's water and wash the assembly with demineralised water. After washing the water was homogenized by passing compressed air and a sample of 1.5 liter was taken for background determination. After a resting period of, at least, 4 hours the water was homogenized again and another sample was taken. The ^{137}Cs activity was determined using a coaxial HPGe detector with 25% efficiency (relative to NaI(Tl)). One liter samples were used in all cases, placed on an aluminum positioning guide attached to the shield surrounding the detector. They were measured for at least 3 hours. The analyses of the collected spectra were done using a commercial code (SAMPO 90, from Canberra), which determines the intensities of the peaks and identifies the isotopes present. In various cases no ^{137}Cs photopeak was identified. However, an observation of the spectra showed that there were increases in the counting of the channels located in the 662 keV peak region. It was therefore decided to perform a manual determination of the excess counting in this region for every spectrum, calculating both a net peak counting rate and its statistical error.

Only small ^{137}Cs leakage rates of less than 50 Bq/h were found in 5 of the 39 tested assemblies. The washing and sipping procedures were repeated for these assemblies. In these cases the second and following samples yielded lower leakage values, suggesting that the values measured in the first sample were due to surface contamination that was essentially removed during the first washing and sipping period [2]. The visual inspection has shown that all assemblies were in good condition after more than 30 years in the pool. This can be attributed to the purity of the water and to the fact that the fuel assemblies were always stored in aluminum racks. From these measurements it could be concluded that the total ^{137}Cs leakage rate expected from the 39 fuel assemblies loaded in a single transport cask would be less than 1% of the acceptance criteria of DOE/SRS.

References

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Further work

A procedure similar to the one above described should be repeated with the fuel elements currently being used prior to their shipment to the USA until 2009.

BNCT Activities at RPI *

A.J.G. Ramalho, I.C. Gonçalves, A.P. Fernandes[1], J.G. Marques, A. Vieira[2], M.J. Prata[1], I.F. Gonçalves, M. Castro[3], N.G.Oliveira[1], J.G.Toscano Rico[4]

Objectives

The objective of this project is to develop activities in the field of Boron Neutron Capture Therapy, BNCT, both in the specific domain of Dosimetric Procedures and with the objective of testing the behaviour of biological samples in complex radiation fields.

The work to be conducted will use as main irradiation facility the epithermal neutron beam that is being implemented at one irradiation tube of RPI.

Results

The first objective is the optimisation of the filters to be installed in the neutron beam so as to have it available with the most feasible adequate spectral characteristics.

The MCNP program is being used to define the composition and geometry of the materials to be used as neutron filters. The results already obtained are in the right direction but some more effort will be developed to try to ameliorate the characteristics of the beam [1].

The second objective is the characterisation of the components of the radiation field of the beam. Work is being conducted in the available RPI irradiation facilities, aiming at the implementation of specific dosimetric procedures that will be used in the new beam.

Concerning the characterisation of the neutron component of the field, the main effort is being done to solve the problems related with the use of proton recoil detectors, besides the utilisation of standard techniques like the multi-foil activation for the unfolding of the spectrum.

The determination of gamma doses in mixed fields, neutrons and gamma radiation, presents some problems that need to be addressed. It's worth referring, for instance, that the thermoluminescent materials used for gamma detectors, the so-called TLD detectors, may have an important sensitivity to neutrons. A new material, developed by a Hungarian team, that is supposed to be relatively insensitive to neutrons, is being studied and the conditions of its usage will be established. The points referred before are part of the work being performed in the frame of a PhD thesis in "BNCT Dosimetry".

The third objective is the study of the behaviour of biological samples under special irradiation conditions. The work so far performed has been centred in the use of the Thermal Column of RPI, where boron loaded melanoma cells have been irradiated to study the induced genetic effects [2]. This work is the subject of a PhD thesis in Genetics.

References

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Further work

Work will be pursued aiming at the optimised utilisation of proton recoil detectors and TLD detectors, and other active neutron detectors to be used for beam monitoring and quality control.

* Partialy financed by European Contract SMT4-CT98-2145.

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Return of spent fuel from the Portuguese Research Reactor

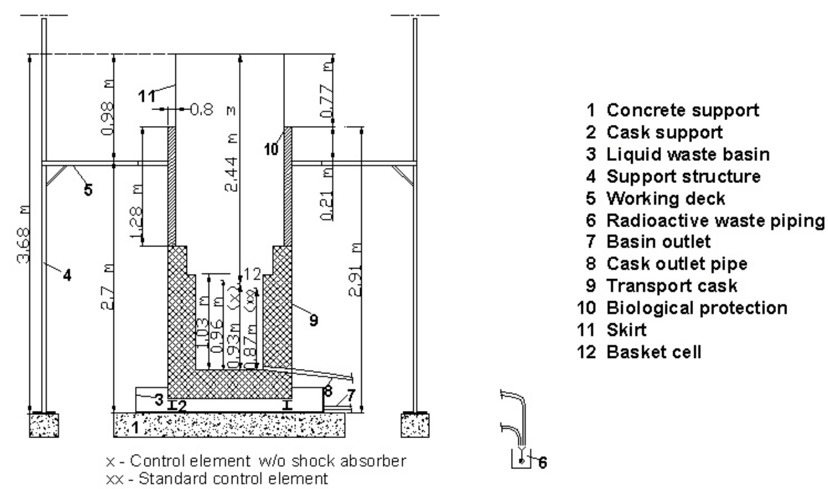
A.J.G. Ramalho, J.G. Marques, F.M. Cardeira

Objectives

The main objective of this work was to return spent fuel assemblies from the first core of the Portuguese Research Reactor to the USA under the “Reduced Enrichment for Research and Test Reactors” program.

Results

Thirty nine spent MTR fuel assemblies from the Portuguese Research Reactor (RPI) were recently returned to the US under the RERTR program. Prior to the shipment all assemblies were visually inspected for corrosion and sipped for determination of fission product leakage. Albeit the long time in the pool (up to 38 years) this work showed that only small ^{137}Cs leakage rates existed in 5 of the 39 tested assemblies (A.J.G. Ramalho, J.G. Marques, F.M. Cardeira, Examination of spent fuel of the Portuguese Research Reactor, this report). A Transnuclèaire IU04 cask with capacity for 40 assemblies was selected for the shipment. Limitations on the floor loading of the reactor building and on the capacity of the central crane prevented the placement and loading of the cask inside the containment building. The transport cask was thus placed outside, under permanent surveillance, in a support structure around which the loading and safety structure shown below was built.



A small transfer cask was used to carry individually the fuel assemblies from the storage racks to the transport cask. The transfer cask was handled inside the building using the central crane of the RPI and outside the building by a mobile crane. A forklift was used as a shuttle between the pool and the transport cask. The whole transfer operation was performed in two days. Samples of water from the cask taken at regular intervals after loading have shown a maximum ^{137}Cs leakage rate of 63 Bq/h, well below the limit imposed by DOE/SRS for acceptance of the shipment. The IU04 cask was put inside an industrial container and taken to a commercial ship bound for the Charleston Naval Weapons Base (SC). The RPI will cease using HEU fuel until May 2006 [1] and will return this fuel to the USA until May 2009 under the conditions of the RERTR program.

References

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Further work

The return of the LEU spent fuel closed a cycle initiated in 1961. Another cycle will be closed until 2009 with the return of the HEU fuel assemblies from the second core.

Preparation of a Fast Neutron Irradiation Facility*

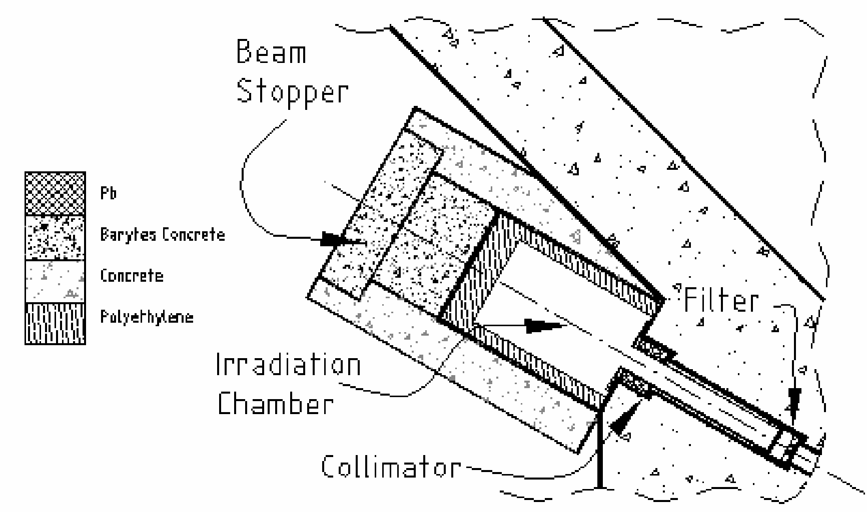
A.J.G. Ramalho, J.G. Marques, N.P. Barradas, A. Vieira[1], M.J. Prata[2], I.C. Gonçalves, A.P. Fernandes[2]

Objectives

In order to evaluate and design radiation tolerant electronic systems for the cryogenic system of the LHC facility at CERN, irradiations with fast neutrons were performed at the RPI, as described elsewhere (A.J.G. Ramalho *et al.*, Fast Neutron Irradiation of Electronic Circuits for the LHC/CERN, this report). The components were placed inside an Al container immersed in the pool, resting on top of the core grid plate. To reach the desired fluence of $5 \cdot 10^{13}$ $n \cdot cm^{-2}$ in 5 days of operation the reactor was operated at 2 kW. A considerable amount of irradiations is now foreseen and it is desirable to have a more versatile setup. This can be achieved using a facility built around a beam tube were it is not only easier to obtain a more favorable n/γ ratio but it is also possible to operate the reactor at 1 MW. Additionally there is more room for the simultaneous test of more boards. The goals to achieve with the new facility are a fast neutron flux of 1×10^8 $n \cdot cm^{-2} \cdot s^{-1}$ and a γ dose rate of less than 10 Gy/h.

Results

A cut of the proposed arrangement is shown below. A dry irradiation cavity with 60x60x100 cm will be created at the end of beam tube E4 outside the pool wall. This ensures a good available volume for circuits and cables running to the measuring instruments. Additionally, a cylindrical cavity will be available inside the first portion of the beam tube (1 m long) where the materials to be irradiated can also be placed. In this way, the materials can be placed inside the cylindrical cavity where the fast neutron flux differs by a factor of 2. The neutron beam size is kept as the one given by the tube's smallest diameter, i.e., 16 cm.



The n/γ ratio is changed via a Pb shield, of which a 5 cm thick disk will be installed in the last section of the irradiation tube. Such a shield will attenuate the fast neutron flux by a factor of 2 and the γ intensity by a factor of 15. Additional Pb disks can be placed at the end of the cylindrical cavity. An optimization of the core configuration is also under study. The objective is to increase the fast neutron flux in the E4 tube by replacing the closest core Be reflector elements by an Al reflector. Regarding the shielding of the facility, MCNP calculations show that the maximum total dose is less than 2 mR/hr for the intended neutron flux, using a combined shield of 20 cm of polyethylene and 80 cm of high density concrete. Measurements of the neutron and γ fields were done inside the beam tube to calibrate the MCNP calculations.

Further work

It is expected that this facility will be fully operational in the first quarter of 2000.

* Financed by Contract CERN/P/FIS/15181/1999.
[1] ITN Post Doctoral.
[2] PRAXIS PhD student.

Preparation of a Multipurpose Beam Tube

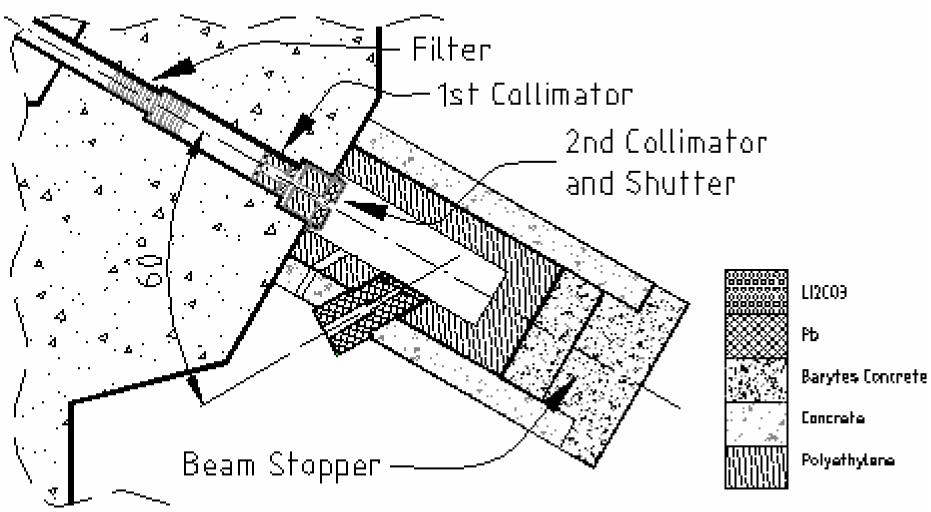
A.J.G. Ramalho, M.J. Prata[1], J.G. Marques, I.C. Gonçalves, A.P. Fernandes[1], I.F. Gonçalves

Objectives

In contrast with the work described in another section (A.J.G. Ramalho *et al.*, Preparation of a Fast Neutron Irradiation Facility, this report) other applications require a neutron beam where the epithermal vs. fast component ratio is maximised. Among those applications are activities connected with Boron Neutron Capture Therapy (BNCT) which may require irradiation with epithermal or both epithermal and thermal neutrons and boron determinations using Prompt Gamma Neutron Activation Analysis (PGNAA). Another envisaged application is the determination of hydrogen at ppm levels in materials with interest for fission and fusion by profiting of the anisotropic scattering of neutrons by hydrogen.

Results

A cut of the proposed arrangement is shown below. The epithermal *vs*. fast component ratio is modulated by a filter inserted in the tube. The aperture and shutting of the beam are generated by two collimators, one of them also containing a fail-safe shutter, placed at the wall end of the beam tube. Two beam openings with 2 and 5 cm diameter are provided, with the possibility of leaving or eliminating the thermal component with a Cd filter. An irradiation cavity with 40×40×125 cm is created outside the tube. The radiation shielding combines 30 cm of polyethylene with 80 cm of barytes concrete in the beam direction and 20 cm of regular concrete and 20 cm of polyethylene in the four side faces.



One of the lateral faces of the shielding has 2 openings, one for an HPGe detector for PGNAA and another for passage of cables such as those of the active detectors used in the determination of H. The HPGe detector is placed at 60°, relatively to the incoming beam, protected by Pb and Li_2CO_3 shields.

Further work

It is expected that the facility will be operational in the first semester of 2000.

[1] PRAXIS PhD student.

Updating and implementation of computer codes relating to the RPI in a personal computer

N.P. Barradas, A.J.G. Ramalho

Objectives

Many of the computer codes used in calculation of working and safety parameters of the Portuguese Research Reactor (RPI) were written for, or available only in, old, slow, and partially not working, computers. Some of the programs were written in command languages of computers that no longer exist at ITN. We have updated the codes, translated into Fortran those written in different languages, and implemented them in a personal computer running MS DOS and Windows 98.

Results

Generational change makes replacement of research reactor personnel an important issue. The responsible for calculations at the Portuguese Research Reactor (RPI) has retired in 1997 and passed away shortly afterwards. A replacement could only be engaged in April 1999. Furthermore, the main computer used for the calculations, an obsolete VAX, is no longer used.

In 1999, all codes, including WIMSD5 and CITATION, were implemented on a modern PC running Windows 98 and MS DOS, which required many adaptations and changes to the existing codes. Previously documented calculations have been thereafter successfully repeated. The numerical precision of the calculations has been improved due to the use of modern computers. The history of the fuel burnup in the Core 2 of the RPI (1 MW, 93.2% ^{235}U enriched - MTR, first criticality 18Jan90, first operation at 1 MW 21Feb90) up to the end of 1999 has been studied.

All relevant aspects are documented, allowing anyone to repeat the calculations and perform new ones using the existing codes. Full documentation has been prepared [1].

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Further work

The calculations performed so far were two-dimensional. We shall implement code for three-dimensional calculations. We shall also investigate many test core configurations in order to determine working and safety parameters of proposed new configurations.

Production of radioisotopes in low power nuclear research reactors

M. Neves, A. Kling, R.M. Lambrecht[1]

Objectives

Low power nuclear reactors play a key role in the generation of radioisotopes for the development and production of radiopharmaceuticals. In order to develop effective radiopharmaceuticals it is crucial to evaluate if the available neutron flux provides, or not, the suitable conditions to produce the appropriated radionuclides. We are interested to studied the viability production of several radionuclides with potential interest in therapy, using the open-core swimming-pool type Portuguese reactor (RPI) with thermal neutron fluxes in the range of 10^{11} – 2×10^{13} $n\cdot cm^{-2}\cdot s^{-1}$. In the case of RPI the operation cycle of 12 hr a day operation for 5 days per week, and the possibility of 36 hr continuous irradiation using a neutron flux of $1.6x10^{13}$ $n\cdot cm^{-2}\cdot s^{-1}$ was evaluated.

Results

The expected specific activities for various radioisotopes (^{64}Cu, ^{111}Ag, ^{153}Sm, ^{165}Dy, ^{166}Ho, ^{170}Tm, ^{177}Lu, ^{186}Re, ^{199}Au) produced for the conditions mentioned above were calculated using the NAC program (Weinstein, S. T., NAC - Neutron Activation Code, NASA TM X-5260 (1968)) neglecting the effect of self-shielding. Calculations were performed assuming continuous irradiations as well as repeated cycles of 12 hours of irradiation and 12 hours breaks (the latter one is the common way of RPI operation). For some of the isotopes also the specific activities in the case of using highly enriched targets were studied. For irradiation, the samples sealed in quartz ampoules were encapsulated in a plastic bag and put into a sealed aluminum container.

The calculations pointed out that in many cases the use of highly enriched targets is advantageous despite the higher costs for the material to be irradiated. In order to test the theoretical calculation, samples of samarium and holmium were irradiated in the core position 45 (thermal neutron flux: $1.6x10^{13}$ $n\cdot cm^{-2}\cdot s^{-1}$) of RPI, at several irradiation cycles and cooling times. Using the NAC program for each sample and each experimental irradiation cycle and cooling times, the calculated activities and the measured activities are compared referred to the same date. The activities were measured in a dose calibrator Capintec, Model CRC-15R. The results of the calculations and the measurements agreed very well with each other and show that all major effects influencing the obtainable activity (e.g. self-shielding) have been considered correctly.

This study enables us to choose the most appropriate conditions for the production of radioisotopes in terms of continuous or 5×12 hours cycles operation of the RPI. It also showed in which the cases the use of enriched samples should be preferred. The results show that low power reactor facilities provide the opportunity for cooperative investigations in biomedicine, and the provision of therapeutic radioisotopes for research, radiopharmaceutical development and clinical applications [1,2].

References

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2. Neves, M., Kling, A., Lambrecht, R.M., Rosado, M., Fausto, R., Radiolanthanides in Bone Therapy ALASBIMN Journal Vol. 5, ref. 12.20, (1999).

Further work

Besides the radioisotopes mentioned above the possible production of new isotopes and generator systems will be investigated.

[1] Alan J. Blocky Research Reactor Facility, V.A. Medical Center, Omaha, NB, USA.

Dose rates during handling of spent fuel at the RPI

A.J.G. Ramalho, A. Kling, J.G. Marques

Objectives

For the shipment of spent fuel of the RPI the fuel assemblies had to be transferred from the reactor pool to a transport cask located outside the reactor building. The purpose of this work was to estimate the radiation exposure to the persons performing the loading of the fuel assemblies into the transport cask under different possible conditions and to optimize the handling procedure minimizing the exposure dose. The calculations were compared with measurements performed during the actual loading.

Results

The dose rate computations were performed using the Monte Carlo code MCNP 4B (Briesmeister, J.F. (ed.), LA-12625-M, version 4B, 1997). The actual setup for the loading procedure was approximated by a simplified model assuming the transport cask and the loading skirt on top of it to be of cylindrical symmetry. The fuel assemblies were modeled by rectangular slabs with a homogenous mixture of fuel, structural material and water. A uniform distribution along the active fuel assembly length is assumed for the radiation source consisting of 0.2 Ci ^{60}Co and 90 Ci ^{137}Cs. These activities are average values determined using the PHDOSE code (Pond, R.B., Matos, J., ANL/RERTR/TM-25, 1996) taking into account fuel assembly burnup, aluminum composition and the decay time.

Calculations were performed for the fuel assembly residing at the bottom of the loading skirt in its resting position. This was the anticipated situation during the fuel assembly identification cross check. Additional computations were done for the main part of the fuel assembly being in the transfer cask but 15 cm of its active length sticking out from the bottom of it to estimate the dose rates during unloading of the transfer cask. In order to determine the conditions under which this could be done safely the relative position of the bottom of the transfer cask with respect to the top of the biological shield was varied between 0 cm and –30 cm (i.e., 30 cm below the top of the biological shield).

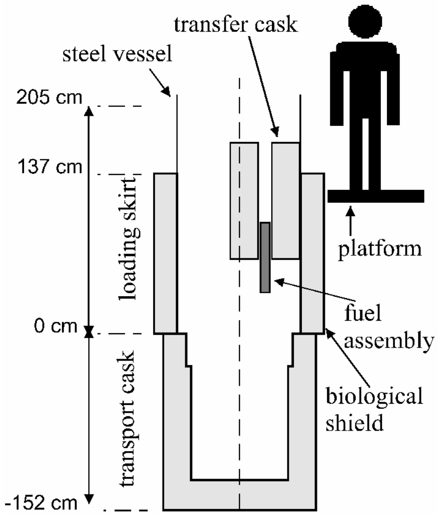

Fig. 1: Sketch of the setup during loading

The maximum calculated dose rates in the abdominal part of the operators' body are 0.07 mrem/h for the fuel assembly resting at the bottom of the loading skirt. For the case of the fuel assembly sticking out of the transfer cask a safe handling can be achieved by positioning the transfer cask 30 cm below the top of the biological shield. The abdominal dose rate in this case was calculated to be 0.02 mrem/h. These values show that a safe handling of the fuel assemblies was possible for the identification cross check and the unloading from the transfer cask to the loading skirt. The calculated values were compared with values measured during the loading procedure with various fuel assemblies resting at the bottom of the loading skirt. Calculation and measurement agreed well for the fuel assemblies that matched the assumed source strength. It was demonstrated that the expected dose rate have been successfully estimated by the MCNP calculations and supply a reliable tool for the planning of handling procedures [1].

Reference

1. Ramalho, A.J.G., Kling, A., Marques, J., Dose rates during handling of spent fuel at the RPI, ITN Internal Report ITN/RPI-R-99-56.