# ACTIVATION ANALYSIS OF DECOMISSIONING OPERATIONS FOR RESEARCH REACTORS

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ABSTRACT

By the end of 2016, there are 241 operational research reactors all over the world. More than 70 % of them are over 30 years old and more than 50 % are over 40 years old. In this scenario, it is foreseen that a major number of them, in the near future, will be decommissioned or will receive life-extension or upgrades.
Decommissioning, life-extension and upgrading involve manipulating and removing equipment and components located close to the reactor core which have suffered material activation. These activated materials could induce dose rate expositions for workers or electronics tools, mostly via gamma emission decay.
In fission nuclear reactors, material activation is mainly produced by neutron capture. Neutron capture rates are a function of material cross section and neutron flux in each component.
Neutron capture produces radioactive isotopes during reactor operation which then decay during shutdown periods. Therefore, the reactor operation history has to be considered to calculate the dose rate around the component at the removal/handling time.
The aim of this work is to present a comprehensive study of dose minimization during removal or replacement of in pool components during the development of a 1 $MW_{th}$ MTR research reactor upgrade program.
To avoid exceeding dose limits, the analysed operations are devised considering which components are more activated and have to be handled remotely or with a special shielding.
Impurities play an important role in material activation; thus studies to identify the most relevant of them are carried on.
Neutron capture rates vary with neutron flux spectrum, therefore this work pursues to find in what situations a better knowledge of the neutron spectrum is needed for activation calculations.
The calculation methodology includes using a stochastic transport method code to calculate the neutron flux in the different beams and in pool components. These neutron fluxes, plus the material compositions and the operational history, are used as input data for the inventory code which is used to determine the isotope activities at the handling time. Finally, the radioactive isotopes produce the gamma source used in gamma transport calculations to obtain the dose rate around the relevant components.

## 1 Introduction

This study is focused on a pool type, low power (1 $MW_{th}$ as a reference), multi-purpose research reactor which is going to receive medium life maintenance and upgrade program. The upgrading involves manipulating and removing activated in-pool equipment and components and irradiation beams. Workers involved in these manoeuvres are exposed to gamma radiation from these activated materials.

Workers could also receive dose from superficial contamination but the contamination can be significantly reduced using an appropriate cleaning procedure. Therefore, in this study, superficial contamination is not considered.

To minimize expositions, operations have to be devised considering the dose rate field of the place where the tasks are to be carried on. In fission nuclear reactors, the main contribution to material activation comes from neutron capture reactions.

To calculate material activation it is necessary to know the neutron flux received at power operation. Neutron captures create radioactive isotopes during reactor power operation which then decay during shutdown periods. Therefore, the reactor operation history has also to be considered to calculate the dose rate around the components at the handling time.

Most of the neutron captures produce short half-life radioisotopes which rapidly decay after an appropriate cooling time. Their contribution to photon radiation can be neglected. Few months after reactor shutdown, the main dose exposition risk comes from isotopes with medium half-life that decay emitting high energy photons. In general, due to an appropriate material selection process during the reactor design, only material impurities can produce isotopes with medium half-life emitting high energy photons. Thus, studies to identify the most relevant impurities have to be carried out.

With all the above considerations the final dose rate field can be calculated and the analysis of the operation to avoid exceeding dose limits can be discussed.

## 2 Methodology

### 2.1 Reactor Characteristics

The adopted open-pool, multi-purpose research reactor for this study is a Material Testing Reactor (MTR) type of 1 $MW_{th}$ power. The core is inside a 150 cm radius light water pool which is in turn in a 200 cm minimum width heavy concrete hexagonal shape block. It has five neutron extraction beams, each consist of two consecutive stainless steel tube sections, the first section is of 8 in and the second section of 10 in diameter. A schematic layout of the reactor and the extraction beams is shown in Fig. 1.

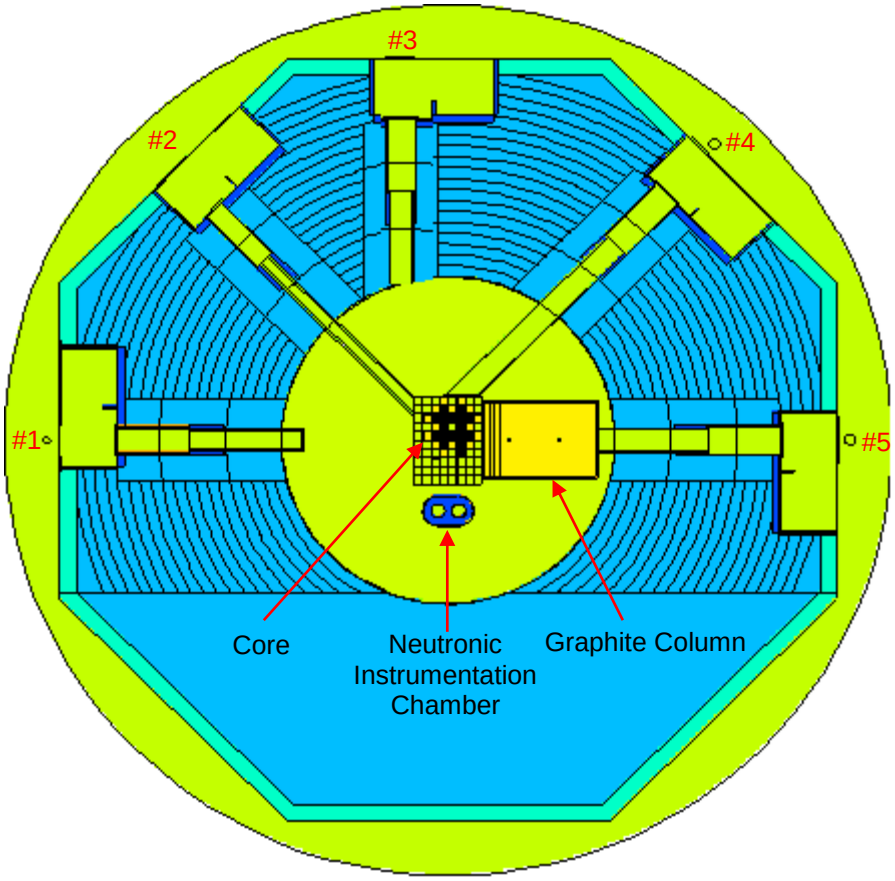


Fig 1. Layout of the core and extraction beam tubes in the reactor pool, and heavy concrete block.

The core is composed of 12 $U_3O_8$ 19.75 % enrichment MTR type fuel assemblies with 290 g

of $U^{235}$ per fuel assembly. Also it has 5 control fuel assemblies, which are $U_3O_8$ 19.75 % enrichment MTR type fuel assemblies with a gap for introducing absorber plates (less fuel plates in these than in the standard fuels). The five control fuel assemblies have 214 g of $U^{235}$ each.

## 2.2 Flux, Activation and Dose Rate Calculation

The study is compound of 3 main successive steps:

a) Neutron fluxes are calculated by modelling the whole reactor core and the main components in the reactor pool with MCNP5 v1.6 [1]. The reactor core is calculated at full power assuming fresh fuel, all control rods out (no axial perturbation), at the hot with xenon state.

b) The neutron fluxes calculated are used as input for material activation which is calculated using the ORIGEN-S code from the SCALE 6.1 software package [2]. Given the particular operation history of the reactor to be upgraded, an operation cycle of a few hours per week is considered, during 30 calendar years. The result of this step is the photon emission spectra of the materials.

c) Then, these photon emission spectra are used to calculate equivalent ambient dose rate from activated material (from now on dose rate), using MCNP5 v1.6. The ICRP-107 [3] flux-to-dose conversion factors are used to translate photon flux into dose rate.

## 2.3 Components Studied

The main contribution to the dose rates comes from the activation of stainless steel and aluminium components. This study is going to be focused on the following components:

a) Aluminium structure of the thermal column
b) Irradiation beam tubes
c) Reactor pool wall
d) Structural supports for neutronic instrumentation
e) Core grid
f) Core support girders
g) Suction funnel

# 3 Results and Discussion

## 3.1 Neutron Flux Calculation

Tab. 1 shows the results of the neutron flux calculation in the different components for a 1 MW fresh core, all control rods out, hot with xenon operation condition.

All the neutron flux calculations presented in Tab. 1 have results with relative statistical error of less than 10 % at 1σ.

| Component | Part | Material | Average Neutron flux ($n \cdot cm^{-2} \cdot s^{-1}$) | | | |
|---|---|---|---|---|---|---|
| | | | Thermal (< 0.625 eV) | Epithermal (0.625 eV - 821 eV) | Fast (> 821 eV) | Total |
| Thermal column structure | Front face | Aluminium | 4.2E+12 | 1.9E+12 | 5.0E+11 | 6.5E+12 |
| | Rest of the structure | Aluminium | 5.1E+11 | 6.4E+10 | 1.4E+10 | 5.9E+11 |
| Irradiation beam #1 | Window | Stainless steel | 1.3E+07 | 2.2E+07 | 1.8E+07 | 5.4E+07 |
| | 8" tube | Stainless steel | 4.5E+06 | 2.8E+06 | 1.0E+06 | 8.3E+06 |
| | 10" tube | Stainless | 3.8E+04 | 1.4E+06 | 2.9E+05 | 1.7E+06 |

| Component | Part | Material | Average Neutron flux ($n \cdot cm^{-2} \cdot s^{-1}$) | | | |
|---|---|---|---|---|---|---|
| | | | Thermal (< 0.625 eV) | Epithermal (0.625 eV - 821 eV) | Fast (> 821 eV) | Total |
| | | steel | | | | |
| | Shutter box | Stainless steel | 6.1E+03 | 2.3E+05 | 5.4E+04 | 2.9E+05 |
| Irradiation beam #2 | Window | Stainless steel | 3.7E+11 | 1.8E+11 | 9.2E+10 | 6.4E+11 |
| | 8" tube | Stainless steel | 3.9E+10 | 2.3E+10 | 1.2E+10 | 7.5E+10 |
| | 10" tube | Stainless steel | 2.4E+07 | 1.2E+08 | 2.9E+07 | 1.7E+08 |
| | Shutter box | Stainless steel | 4.6E+06 | 1.8E+07 | 3.9E+06 | 2.7E+07 |
| Irradiation beam #3 | Window | Stainless steel | 6.0E+09 | 7.2E+09 | 1.9E+09 | 1.5E+10 |
| | 8" tube | Stainless steel | 7.5E+07 | 2.2E+08 | 5.1E+07 | 3.5E+08 |
| | 10" tube | Stainless steel | 3.7E+06 | 2.0E+07 | 2.1E+06 | 2.6E+07 |
| | Shutter box | Stainless steel | 5.3E+05 | 2.6E+06 | 1.9E+05 | 3.3E+06 |
| Irradiation beam #4 | Window | Stainless steel | 2.3E+12 | 1.7E+12 | 6.6E+11 | 4.7E+12 |
| | 8" tube | Stainless steel | 2.8E+11 | 2.1E+11 | 7.7E+10 | 5.7E+11 |
| | 10" tube | Stainless steel | 1.6E+08 | 7.5E+08 | 1.3E+08 | 1.0E+09 |
| | Shutter box | Stainless steel | 3.2E+07 | 1.3E+08 | 1.7E+07 | 1.8E+08 |
| Irradiation beam #5 | Window | Stainless steel | 1.6E+10 | 2.7E+08 | 3.3E+07 | 1.6E+10 |
| | 8" tube | Stainless steel | 5.7E+09 | 1.1E+08 | 2.9E+07 | 5.8E+09 |
| | 10" tube | Stainless steel | 1.1E+07 | 1.2E+06 | 6.7E+04 | 1.2E+07 |
| | Shutter box | Stainless steel | 7.2E+05 | 3.1E+04 | 1.1E+03 | 7.5E+05 |
| Reactor pool | Wall | Stainless steel | 9.9E+06 | 1.1E+07 | 1.9E+06 | 2.3E+07 |
| Supports for neutronic instrumentation | Support plate | Aluminium | 1.3E+09 | 6.7E+08 | 3.0E+08 | 2.2E+09 |
| | Support body | Aluminium | 3.9E+09 | 1.5E+09 | 8.7E+08 | 6.3E+09 |
| Core grid | Grid | Stainless steel | 1.6E+11 | 3.4E+11 | 1.1E+11 | 6.2E+11 |
| Core support | Girders | Stainless steel | 4.0E+09 | 2.0E+10 | 5.6E+09 | 3.0E+10 |
| Suction funnel | Funnel | Stainless steel | 9.9E+09 | 9.7E+09 | 3.0E+09 | 2.3E+10 |

Tab 1: Calculated neutron fluxes in different components.

## 3.2 Activation Calculation and Impurities Contribution Analysis

The analysis takes into account the reactor has been operated, in average, for 3 hours a week during 30 calendar years. A cooling time equal to 60 days before handling is also considered.

Activation rate is determined by the magnitude and spectrum of the activating neutron flux. Tab. 1 presents the magnitude but it is still necessary to know the spectrum of the neutron flux. Spectrum calculations at different distances from the core are calculated using MCNP5.1 and the results are shown in Fig. 2.

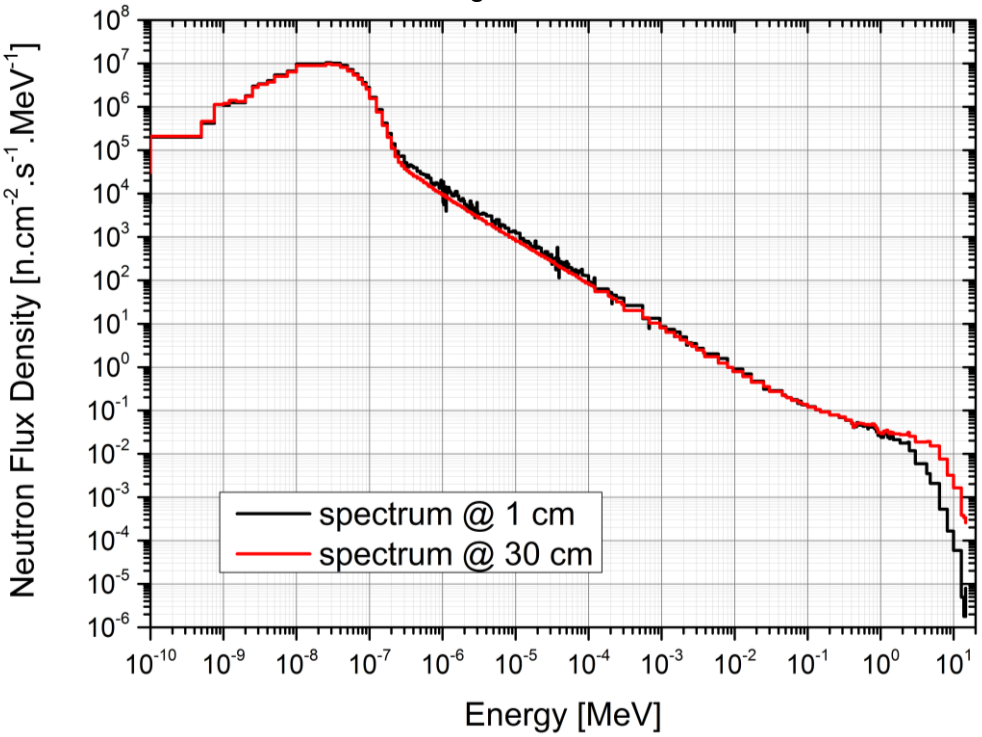


Fig 2. Normalized neutron spectrum density at 1 cm and 30 cm of the core.

Fig. 2 shows that the neutron spectrum at 1 cm of the core differs to the neutron spectrum at 30 cm of the core mainly in the high energy region. A subsequent analysis will show that, in a light water moderated reactor as the case study, thermal neutrons are more important for activation than high energy neutrons.

Material compositions are the other important magnitudes for activation calculations. The materials considered in this study are stainless steel 304L (SS304L, composition in Tab. 2) and Al6061 aluminium (composition in Tab. 3). The compositions used are the ones from [4] where no impurities are presented but its importance has to be analysed.

| **Density:** | 8.00 g/cm$^3$ |
|---|---|
| **Element** | **wt %** |
| Fe | 70.3155 |
| C | 0.15 |
| Si | 0.5 |
| Mn | 0.01 |
| P | 0.023 |
| Si | 0.0015 |
| Cr | 19.0 |
| Ni | 10.0 |

Tab 2: SS304L composition.

| **Density:** | 2.70 g/cm$^3$ |
|---|---|
| **Element** | **wt %** |
| Al | 97.3745 |
| Cu | 0.275 |
| Cr | 0.0195 |
| Mg | 1.0 |
| Si | 0.6 |
| Zn | 0.146 |
| Fe | 0.409 |
| Mn | 0.088 |
| Ti | 0.088 |

Tab 3: Al6061 composition.

In order to study the impurities contribution, a sensitivity analysis with different compositions is presented. Three cases for each material are calculated. Maximum impurities for SS304L are taken from [5]. Maximum impurities for AL60661 are taken as the maximum allowed for other elements not listed (500 ppm) in [6]. Standard amount of traces are taken for INVAP's reactor design experience. The impurities in each case are shown in Tab. 4.

| | **Element (ppm)** | | | | | | |
|---|---|---|---|---|---|---|---|
| **Alloy** | **Case 1: No impurities** | | | | | | |
| | **Co** | **N** | **Nb** | **Mo** | **Sn** | **Ag** | **Cl** |
| SS304L | - | - | - | - | - | - | - |
| AL6061 | - | - | - | - | - | - | - |
| **Alloy** | **Case 2: Maximum amount of impurities** | | | | | | |
| | **Co** | **N** | **Nb** | **Mo** | **Sn** | **Ag** | **Cl** |
| SS304L | 2600 | 1000 | 300 | 5500 | 200 | 20 | 130 |
| AL6061 | 500 | - | - | - | - | - | - |
| **Alloy** | **Case 3: Standard amount of traces** | | | | | | |
| | **Co** | **N** | **Nb** | **Mo** | **Sn** | **Ag** | **Cl** |
| SS304L | 1200 | - | - | - | - | - | - |
| AL6061 | 60 | - | - | - | - | - | - |

Tab 4: Impurities cases considered.

Activation is calculated with 1500 cycles of 3 hours of irradiation and 165 hours of decay time per cycle and 60 days cooling time. The material reference is 1 g and the total flux is 1E10 $n.cm^{-2}.s^{-1}$. The neutron spectra used are taken at 1 cm from the core.

The calculated photon emissions per each activated material are shown in Fig. 3 and Fig. 4.

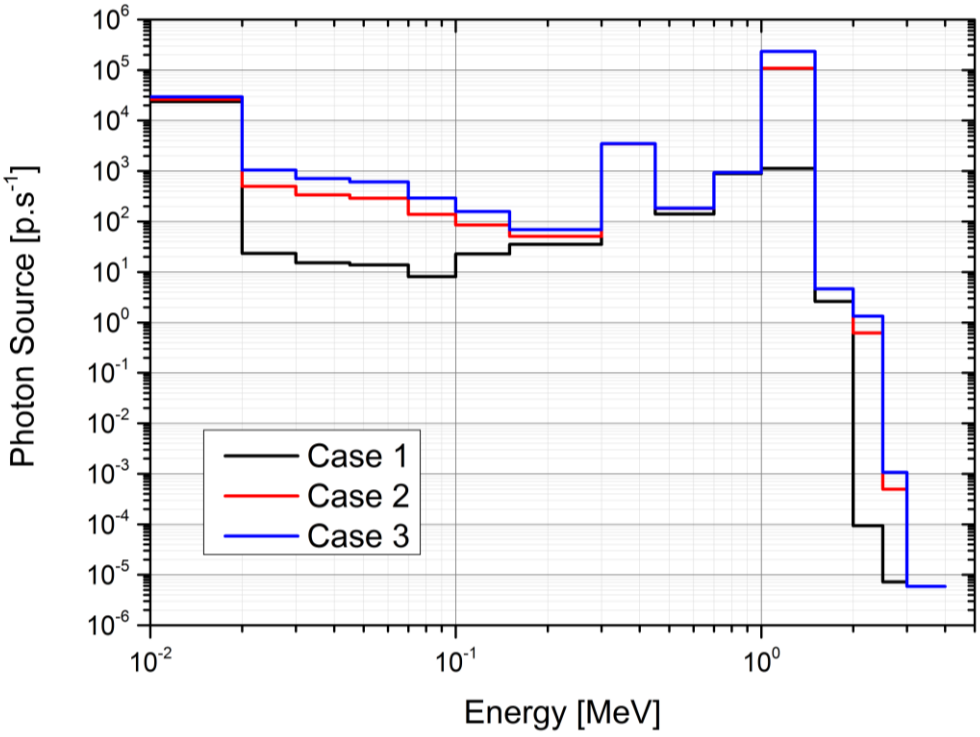


Fig 3. Photon emission for 1 gram SS304L for different impurities concentrations after irradiation and 60 days of cooling time.

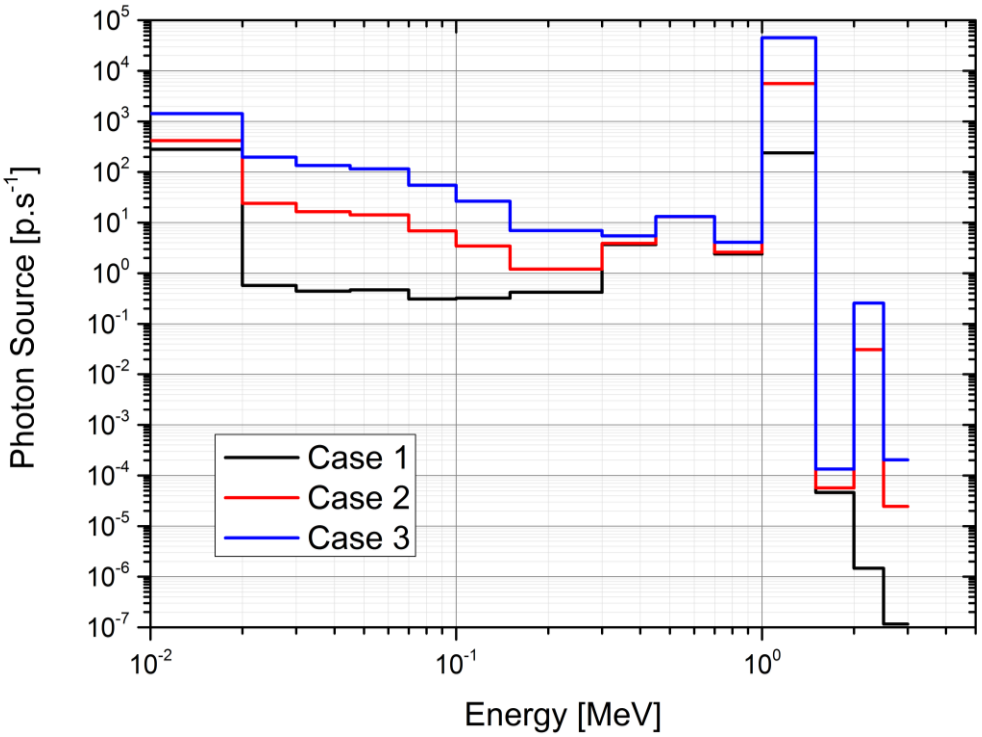


Fig 4. Photon emission for 1 gram Al6061L for different impurities concentrations after irradiation and 60 days of cooling time.

Results from Fig. 3 and Fig. 4 show that the major photon contribution is in the energy group between 1 and 2 MeV. Those photons came from the decay of Co-60 (1.17 and 1.33 MeV photons). Co-60 appears as a result of activation of cobalt impurities. Cobalt production rate is mainly due to thermal neutron capture, this behaviour allows having accurate enough results using the 1 cm distance to the core spectrum.

### 3.3 Dose Rate Calculation

To obtain the final dose rates at the handling time (60 days after last power operation) the model used is the same as for Fig. 1 but with the fuel assemblies and the reflector assemblies completely removed.

The photon sources of the problem are obtained scaling the total fluxes of Tab 1 and multiplying the total mass of the components by the photon emission per gram of Fig. 3 and Fig. 4 for standard impurities concentration cases (1200 ppm Co in SS304L and 60 ppm, Co in Al6061). The results are shown in Tab. 5.

| **Component** | **Part** | **Material** | **Density (g/cm$^3$)** | **Volume$^{(1)}$ (cm$^3$)** | **Total photon emission (photon·s$^{-1}$)** |
|---|---|---|---|---|---|
| Thermal column structure | Front face | Aluminium | 2.70 | 5040 | **5.4E+10** |
| | Rest of the structure | Aluminium | 2.70 | 33588 | **3.2E+10** |
| Irradiation beam #1 | Window | Stainless steel | 8.00 | 130 | **7.1E+05** |
| | 8" tube | Stainless steel | 8.00 | 5594 | **4.1E+06** |
| | 10" tube | Stainless steel | 8.00 | 5130 | **9.2E+05** |
| | Shutter box | Stainless steel | 7.85 | 13438 | **4.3E+05** |
| Irradiation beam #2 | Window | Stainless steel | 8.00 | 212 | **1.5E+10** |

| Component | Part | Material | Density ($g/cm^3$) | Volume[(1)] ($cm^3$) | Total photon emission ($photon \cdot s^{-1}$) |
|---|---|---|---|---|---|
| | 8" tube | Stainless steel | 8.00 | 10320 | **4.9E+10** |
| | 10" tube | Stainless steel | 8.00 | 5127 | **1.1E+08** |
| | Shutter box | Stainless steel | 7.85 | 13440 | **3.8E+07** |
| Irradiation beam #3 | Window | Stainless steel | 8.00 | 825 | **1.4E+09** |
| | 8" tube | Stainless steel | 8.00 | 15501 | **1.1E+11** |
| | 10" tube | Stainless steel | 8.00 | 5124 | **1.8E+07** |
| | Shutter box | Stainless steel | 7.85 | 13447 | **5.0E+06** |
| Irradiation beam #4 | Window | Stainless steel | 8.00 | 214 | **1.1E+11** |
| | 8" tube | Stainless steel | 8.00 | 10676 | **3.9E+11** |
| | 10" tube | Stainless steel | 8.00 | 5132 | **7.4E+08** |
| | Shutter box | Stainless steel | 7.85 | 13452 | **2.7E+08** |
| Irradiation beam #5 | Window | Stainless steel | 8.00 | 150 | **2.7E+08** |
| | 8" tube | Stainless steel | 8.00 | 5352 | **7.6E+08** |
| | 10" tube | Stainless steel | 8.00 | 5130 | **8.2E+06** |
| | Shutter box | Stainless steel | 7.85 | 13452 | **1.0E+06** |
| Reactor pool | Wall | Stainless steel | 8.00 | 236836 | **6.0E+08** |
| Supports for neutronic instrumentation | Support plate | Aluminium | 2.70 | 5398 | **2.1E+07** |
| | Support body | Aluminium | 2.70 | 5126 | **5.5E+07** |
| Core grid | Grid | Stainless steel | 8.00 | 45620 | **3.1E+12** |
| Core support | Girders | Stainless steel | 8.00 | 167465 | **1.2E+11** |
| Suction funnel | Funnel | Stainless steel | 8.00 | 8947 | **2.0E+10** |

**(1) Volumes are conservatively calculated, for more accurate calculations better volumes estimation has to be done.**

Tab 5: Total photon emissions for the different reactor components.

Sources of Tab. 5 are isotropic and homogenously distributed inside each component volume.

## 3.4 Planned Tasks and Dose Fields at Operator Position

In a medium life maintenance and upgrade program the identified manoeuvres with radiation exposition risk are:

a) Irradiation beam tubes refurbishments: manoeuvres inside the shutter boxes are expected in order to clean the beam tubes and to add instrumentations and drivers for shutter remote operations.

b) Reactor pool components handling: it is expected to change in pool instrumentation and to add or upgrade irradiation facilities.

In the case of tasks inside the beam tube shutter boxes, the beam tubes will be drained, in consequence no water will be present as biological shielding. Therefore, it is important to know if these operations could be carried out without high operator's expositions.

For the in-pool components repair or replacements tasks, to ease the operator's manoeuvres, it is programmed to lower the pool water level. In this case it is necessary to know the minimum water depth that allows safe operations.

Dose rate fields are calculated using MCNP5.1 with the 3D model of Fig. 1 (fuel and reflector removed) and the photon source of Tab. 5. Results are shown in Fig. 5 and Fig. 6.

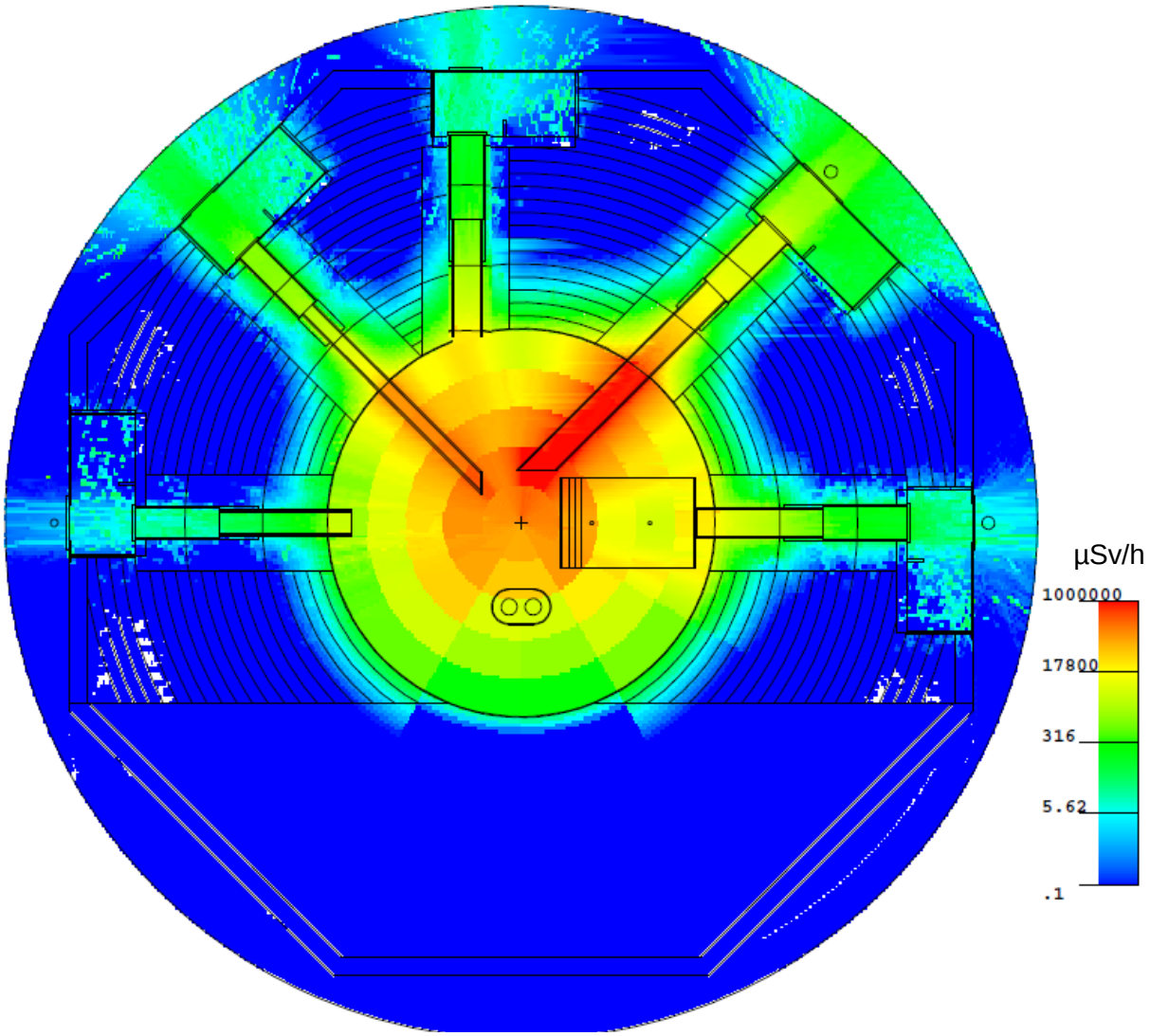


Fig 5. Calculated dose rate field at core centre level (µSv/h)

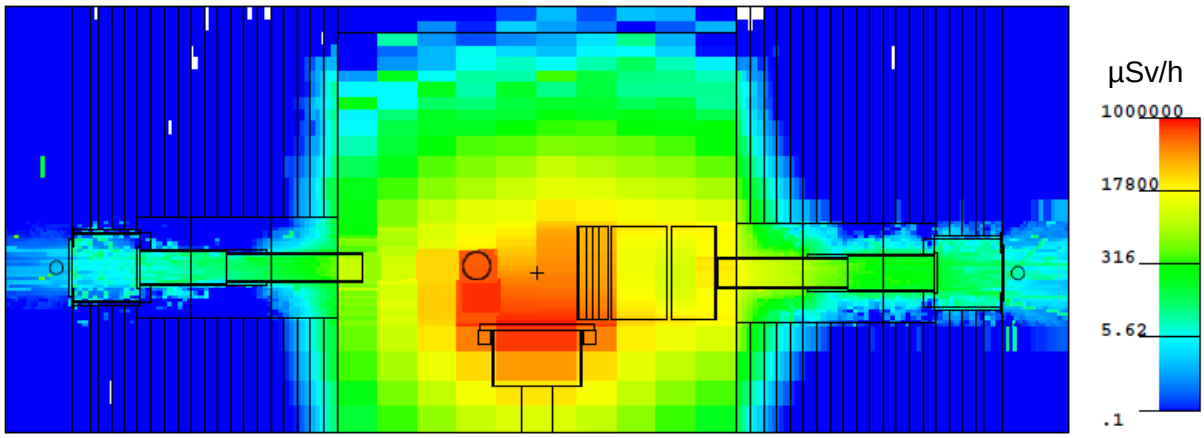

Fig 6. Calculated dose rate field with 200 cm water above the centre of the core (µSv/h)

From the dose rate fields shown in Fig. 5 and Fig. 6 the dose rates at the operator position can be estimated. Results in the relevant position can be found in Tab. 6.

| **Position** | **Position** | | |
|---|---|---|---|
| | **Inside the beam shutter box** | **In contact with the beam shutter box** | **At 10 cm of the beam shutter box** |
| Beam #1 | 50 | 10 | 5 |
| Beam #2 | 500 | 250 | 150 |
| Beam #3 | 200 | 100 | 50 |
| Beam #4 | 5000 | 1500 | 1000 |
| Beam #5 | 150 | 50 | 20 |
| | **Above the pool (with 200 cm water above the centre of the core)** | | |
| Above the pool | 250 | | |

Tab 6: Dose rate estimation at different positions (µSv/h)

From Fig. 5 and Tab. 6 it can be seen that the most compromised beam to work (due to the dose rate) is beam #4.

From Fig. 6 and Tab. 6 it can be seen that with at least 200 cm of water above the centre of the core the operators will receive around 250 µSv/h of dose rate.

## 3.5 Comparison against Experimental Measurements

A similar calculation methodology was previously employed for planning repairing tasks in a 20 $MW_{th}$, MTR-fuel type, open pool reactor. In this case, measurements with an underwater detector were performed in order to validate the calculation methodology.

In general, calculations against experiments (C/E) ratios are around 0.2-1.0, which means a good agreement.

# 4 Conclusions

Neutron flux is calculated using MCNP5.1 in beams and different in-pool components.

With the obtained flux a detailed activation analysis is carried out where it is important to notice:

a) Close core neutron spectrum is appropriate due to the mean contribution to activation of thermal captures.

b) In light water moderated neutron spectrum the main contribution of energetic photon

emission after a few month decay time come from the activation of cobalt impurity.

c) At construction stage the used materials have to be correctly measured to avoid having high dose rates a decommissioning stage

Comparisons done in previous projects against experimental data shows that a good agreement between calculations and reality can be expected.

Operator tasks and positions during an upgrade program were identified and the associated dose rate fields were calculated.

The calculated dose rate field shows that human operators are able to realize the decommissioning manoeuvres with an appropriated radioprotection procedure in the 1 $MW_{th}$ multipurpose research reactor to be upgraded.

## 5 References


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[3] **ICRP Publication 107**. Nuclear Decay Data for Dosimetric Calculations. ICRP Publication 107. Ann. ICRP 38. 2008.

[4] **McConn Jr., RJ et al**., Compendium of Material Composition Data for Radiation Transport Modeling. PNNL-15780 Rev.1. Pacific Northwest National Laboratory, March 2011.

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[6] **SB 210M** - 02 Standard Specification for Aluminum and Aluminum-Alloy Drawn Seamless Tubes, 2013.