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RADIATION SHIELDING ACTIVITIES AT IDOM

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ABSTRACT

When human activities have to be performed under ionising radiation environments the safety of the workers must be guaranteed. Usually three principles are used to accomplish with ALARA (As Low As Reasonably Achievable) requirements: the more distance between the source term and the worker, the better; the less time spent to arrange any task, the better; and, once the previous principles are optimized should the exposure of the workers continues being above the regulatory limits, shielding has to be implemented.

Through this paper some different examples of IDOM's shielding design activities are presented. Beginning with the gamma collimators for the Jules Horowitz Reactor, nuclear fuel's behaviour researching facility, where the beam path crosses the reactor's containment walls and is steered up to a gamma detector where the fuel spectrum is analysed and where the beam has to be attenuated several orders of magnitude in a short distance.

Later it is shown IDOM's approach for the shielding of the Emergency Control Management Center of *Asociación Nuclear Ascó-Vandellòs-II* NPPs, a bunker designed to withstand severe accident conditions and to support the involved staff during 30 days, considering the outside radioactive cloud and the inside source term that filtering units become as they filter the incoming air.

And finally, a general approach to this kind of problems is presented, since the study of the source term considering all the possible contributions, passing through the material selection and the thicknesses calculation until the optimization of the materials.

1. INTRODUCTION

Organizational culture is a concept often used to describe shared corporate values that affect and influence members' attitudes and behaviours. Safety culture is a sub-facet of organizational culture, which is thought to affect members' attitudes and behaviour in relation to an organization's ongoing health and safety performance.

The optimisation of radiological protection of the workers in nuclear industry is an important part of the safety culture where we are confronted with a radioactive environment that is in the process of constant change. A good shielding study must be performed and should contain predicted doses in the work area and investigate the effects of geometry, material, source or work position changes. This information provides a quantitative basis to select between various alternative work scenarios for a specific operation.

In order to handle this information IDOM has developed a problem dependent methodology based on different calculation tools.

It is worth mentioning that this kind of calculations requires a high level of technical expertise and very efficient computational tools. Some simple calculations can be carried out almost by mean of analytical equations, some others can be performed with the proper code in a laptop and when the accuracy of the model is crucial identifying streaming paths in a huge size problem, coupling of different nuclear codes to be run in high speed servers may be needed.
The paper will show the current status of the methodology and its application to the Jules Horowitz Reactor, the Alternative Emergency Control Centre and other installations. Ongoing developments will also be presented regarding recently awarded projects.

2. RADIATION TRANSPORT CALCULATION OF THE UGXR COLLIMATORS FOR THE JULES HOROWITZ REACTOR

Jules Horowitz Reactor (JHR), a major infrastructure of European interest in the fission domain, will be built and operated in the framework of an international cooperation, including the development and qualification of materials and nuclear fuel used in nuclear industry. For this purpose UGXR Collimators, two multi slit gamma and X-ray collimation mechatronic systems, will be installed at the JHR pool and at the Irradiated Components Storage pool. The expected amounts of radiation produced by the spent fuel and X-ray accelerator imply diverse aspects have to be verified in order to ensure an adequate radiological zoning and personnel radiation protection. A computational methodology was devised to validate the Collimators design by means of coupling different engineering codes. In summary, several assessments were performed by means of MCNP5 to fulfil all the radiological requirements in Nominal scenario (Total Effective Dose Equivalent, TEDE < 25µSv/h) and in Maintenance scenario (TEDE < 2mSv/h) among others, detailing the methodology, hypotheses and assumptions employed.

2.1. From the Technical Specification

The necessity to design and build new material testing reactors to support operation of the existing power reactors fleets and qualification of future technologies systems have led to the development of precise mechatronic systems compliant not only with the experimental requirements but also with the very demanding nuclear safety constraints, such as UGXR Collimators. These two multi slit gamma and X-ray collimation systems, located at the JHR pool (RER) and Irradiated Components Storage pool (EPI) Fig-1 and Fig-2 respectively, will allow to carry out the collimation of the hard X-rays towards the underwater part of the transmission imagining measurement system and the gamma beams between the immersed object to be measured and the experimental equipment located in the out of pile section, Fig-3.

These two Collimators, as well as support and hoisting structures for the gammametry detector, shall ensure proper shielding and shall also provide second water barriers behind the pile liners.

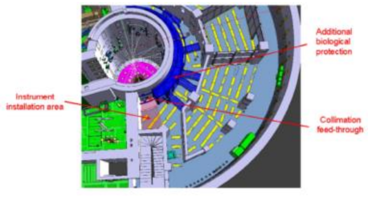

Figure 1: JHR pool Collimator.

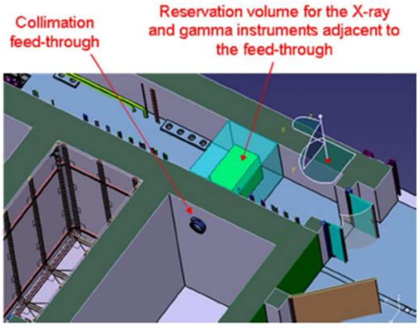

Figure 2: Irradiated Components Storage pool Collimator.

Several assessments are performed to fulfil all the radiological requirements, detailing the methodology employed as well as hypotheses and assumptions. Dose rate compliance is evaluated by mean of radiation transport calculations through MCNP5 [1] and the use of SCALE6.1/ORIGEN-S [2] for the source term definition, both intensity gamma and energy spectra.

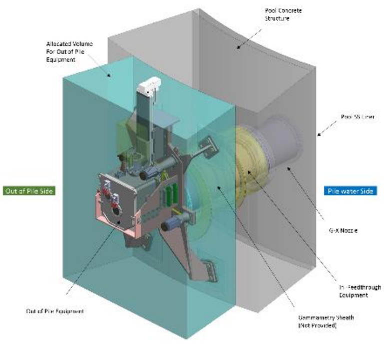

Figure 3: UGXR Collimators.

2.2. The requirements

Large amounts of radiation are expected to be produced by the spent fuel and X-ray accelerator, which implies that several aspects need to be verified to ensure adequate radiological zoning and radiation protection of the personnel. Three main requisites are defined that shall be taken into account during the design phase:

- A TEDE dose rate lower than 25 μSv/h in the technical gallery concerning UGXR equipment in the out of pile section under normal conditions;
- Existence, under normal conditions, of a significant difference between the shielded flux and the collimated flux, considering the Lanthanum-140 energy peak as a reference value. As well, a non-significant scattered component in the detected flux;
- In Maintenance situation, a TEDE dose rate lower than 2 mSv/h in the area concerning UGXR equipment in the out of pile section under normal conditions.

Multiple scenarios and hypotheses are defined that allowed to consider all the possible casuistry, so that even in the worst cases the Collimator adjacent galleries will be safe from the radiological point of view.

2.3. Model definition

The first step, avoiding the CAD generation by scratch, is to simplify and defeature UGXR Collimators models using SpaceClaim2015 [3] package. The purpose of this is to allow successful conversion of the solid CAD bodies to MCNP [1] through SuperMC [4], obtaining a fast conversion process and also reducing the effort required to repair the resulting MCNP model.

It is convenient to remove details which are not important for radiation transport calculations, such as fillets, chamfers, bolt holes, etc. as shown in Fig-4. Whilst convertible to MCNP, the presence of such details significantly increases calculation times.

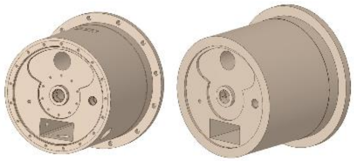

Figure 4: Comparison between original (left side) and simplified model.

The degree of simplification which is acceptable is a function of the photon importance of the features being considered. Generally, it is desirable to retain more detail of channels in the photon gradient direction, since photons would stream down them.

In order to maintain the correct mass of materials, such simplifications are always accompanied by homogenizations of materials and density corrections. Thanks to SuperMC [4], the geometry is modelled in a very detailed way.

Once the model is simplified and checked, the complex solids shall be decomposed into a combination of simple convex solids like boxes or spheres to allow the proper conversion to MCNP5 [1] by means of SuperMC [4] Fig-5.

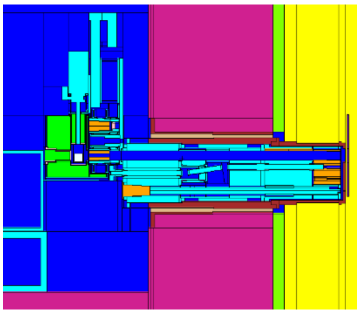

Figure 5: MCNP nominal model vertical cross section.

2.4. Source term & Radiation transport assessment

2.4.1. Source term

The source term considered is based on the maximum fission product activities from an experimental PWR MOX fuel cooled for 3 days. The total number of photon/s per rod emitted by the irradiated fuel is $1.60 \cdot 10^{15}$ with an average energy of around 0.150 MeV. The source term is modelled as a cylindrical fuel pin facing UGXR Collimators.

It has to be remarked that all the mentioned cases are assessed assuming the same conservative source term defined through the fuel activity concentration, determining the energy spectra by means of SCALE6.1/ORIGEN-S [2].

2.4.2. Radiation transport

Radiation transport calculations, which are very computational demanding, are fundamental to accomplish with the dose rate zoning requirements. The radiation transport assessments, which allowed performing the dose rate evaluation to the JHR personnel, are carried out by means of MCNP5 [1].

The MCNP model contains the UGXR Collimator, the technical gallery, the pool and its concrete wall, the fuel pin source, the X-accelerator and the support Fig-6.

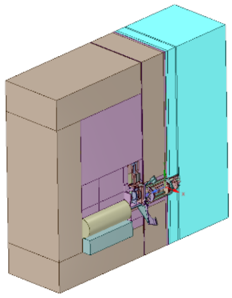

Figure 6: Slice view of the model.

Considering different tallies outside and inside the Collimator allows to determine photon flux, thermal power and dose rate at different points of the Collimator but also in the technical gallery, where the JHR personnel is expected to be. Furthermore, the radiological compliance has been assessed by the use of different MCNP tally typologies (e.g. F15/F25/F24) giving robustness to the analysis Fig-7.

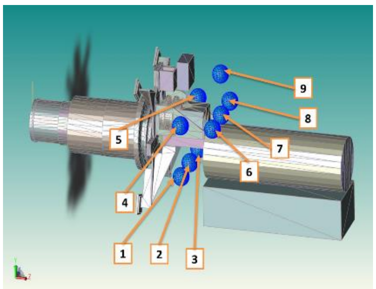

Figure 7: Some of the used tallies.

For the calculations exponential transform variance reduction techniques are employed and optimized, as well as a mono-energetic weight window, obtained by MCNP weight window generator, with a reduced density model. An iterative procedure allowed the optimization of the variance reduction parameters, being lately checked via ADVANTG 3.0.3 code [5] for each of the analyzed models.

2.5. Results

For the "Nominal Situation" scenario, the dose rate estimated in the technical gallery is smaller than 25 µSv/h required, except for a small not-accessible area. The higher dose rate values are located in the UGXR Collimators lower area due to a downside primary photon streaming caused by the scattering of the collimated gamma flux against the post-collimator wheel and collimator as shown in Fig-8 and Fig-9.

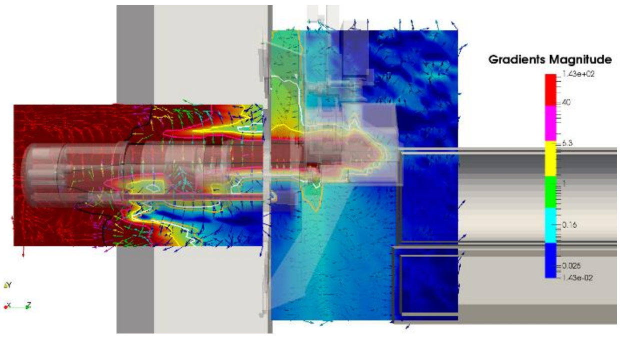

Figure 8: Dose Rate Gradient for Nominal situation.

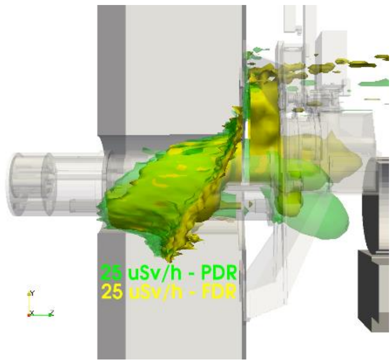

Figure 9: 25 µSv/h Dose Rate 3D contour plot comparison between Final Design (FDR) and Preliminary Design (PDR) for Nominal situation.

3. ALTERNATIVE EMERGENCY MANAGEMENT CENTRE (CAGE)

After the earthquake and tsunami on March 11, 2011 in Fukushima Dai-chi, all nuclear plants in the European Union have been subjected to "stress tests". The Spanish nuclear sector has proposed, and the CSN (Spanish Regulatory Body) has subsequently required, the creation of a centre to safely manage an emergency, called Alternative Emergency Management Centre (CAGE) located at the sites of the Nuclear Power Plants [6]. Living conditions of the occupants of the CAGE in the event of a Severe Accident imply that TEDE must be <50 mSv and the equivalent dose to the thyroid is <500 mSv within 30 days following the accident [7]. Given the weather conditions of each plant, the calculations are analogous to those supporting the Control Room and the different ways of radiation exposure or contamination are simulated Fig-10. These paths that contribute to the dose are:

- Determination of dose due to inner radioactive cloud (within the CAGE).
- Determination of dose due to the presence of the radioactive cloud outside the CAGE.
- Determination of dose due to the accumulation of radionuclides in the filters.
- Determination of dose due to the proximity to the containment.

The variety of contributions to the dose has to be approached in an integral way. Each contribution is due to a different source term or a different interaction with the human body (i.e. external exposure, internal contamination, etc.) that have to be taken into account.

Considering that a radioactive cloud stands around the CAGE during the duration of the accident (720 hours), different situations arise.

Regarding the consequences of radioactive materials being incorporated inside the CAGE atmosphere, external exposure and inhalation of radionuclides contributions have to be evaluated. This contribution requires the knowledge of the radiation transport mechanism and of the site meteorological data. To help solving this problem, the ARCON96, (Atmospheric Relative Concentration in building wakes) [8] and RADTRAD 3.03 [9] are used.

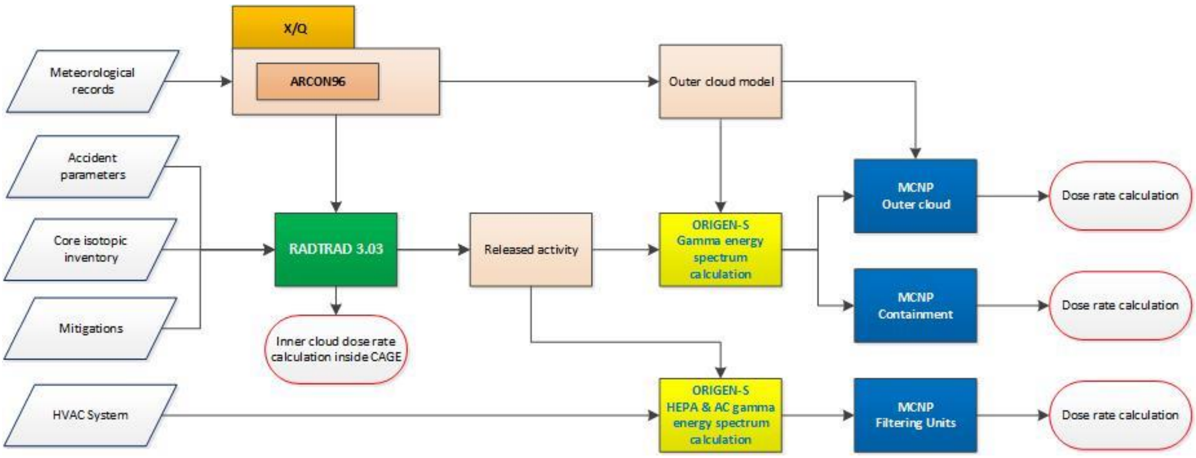

Figure 10: Methodology applied to determine different dose contributions.

On the other hand, the radioactive cloud standing around the CAGE becomes a shielding problem where the source term is outside and the people to protect is inside. Therefore, a shielding has to be designed, mainly the concrete walls and doors. After assuming a geometry and applying the radionuclides activity released to the environment (RADTRAD), ORIGEN-S [2] is used to "translate" the activities into gamma radiation energy spectra. These spectra are introduced as input data in a Monte Carlo radiation transport calculation by means of MCNP [1]. This code delivers the outer cloud contribution.
In a similar way, the containment direct radiation is assessed. The only difference is that in this case, the inside containment activities (RADTRAD) are considered. This problem is highly demanding from a computational point of view because of the thicknesses of shielding (containment and CAGE concrete walls) and distance involved.

Finally, the outer radioactive cloud is being filtered by the filtering units. These HEPA and active carbon filters are not perfect and inner cloud contribution is due to their small inefficiency. Nevertheless, most of the radionuclides are accumulated while the filters work, resulting in a strong source term. To perform this calculation, the activities of filtered radionuclides, and their daughter activities, are taken into consideration.

3.1 Assumptions and Input data

To carry out the necessary calculations by coupling the various codes that perform the methodology used to determine dose rates, there have to be considered situations and initial data that will determine the suitability of the resulting solutions. Then the input data and assumptions depending on the location of the CAGE are presented, as well as for each of the contributions to the final dose rate.

3.1.1 Diffusion factors

According to RG (Regulatory Guide) 1.23 Rev. 1 [10], a Nuclear Power Plant should be able to get the weather information it requires to, among other objectives, determine the potential spread of radioactive material from an accident, so it can be deducted the amount of radionuclides resulting from the release into the environment of the considered source term. The ARCON96, it is a tool developed by the Nuclear Regulatory Commission to perform calculations of diffusion factors for habitability analysis of Control Rooms of Nuclear Power Plants in compliance with RG 1.194 [11].

3.1.3 Determination of dose

Once the diffusion factors have been obtained and therefore, the relative concentration of radionuclides known in the points of study, the analysis of the different routes of contribution to the dose within 30 days of accident principles is studied.

The aforementioned diffusion factors will be introduced as input data in the codes to be used for calculations of radiation transport.

Inner cloud contribution

Determining dose inside the cloud will take place through software RADTRAD, as shown in Fig-10, since as specified in NUREG-1465 [12] and the RG 1195 [13] it is a code that incorporates adequate methodologies to meet dose determination.

External cloud contribution

Determining dose provided by the outer cloud to CAGE is carried out by coupling software as indicated in the flow diagram of Fig-10. RADTRAD 3.03 was used for estimating the release of radioactive materials into the environment in case of a severe accident. Then, using the diffusion factor determined by ARCON96, the average isotopic activity contained in the radioactive cloud in which is immersed the CAGE is obtained. Likewise, the corresponding gamma energy spectrum is determined by software ORIGEN-S, in which are entered as input data the activities obtained for each time interval. Finally, these gamma spectra are introduced in their respective MCNP5 simulations in order to characterize the cloud corresponding to the outer volume source surrounding the CAGE geometry Fig-11.

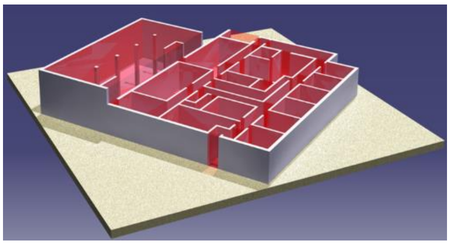

Figure 11: Simplified geometry CAGE

Contribution accumulation in filters

All input data necessary for the definition of spectra by external cloud are necessary for determining activity and gamma energy spectra of radionuclides accumulated in the filters, noting that should generate new geometry models to get the amount retained in filters Fig-12.

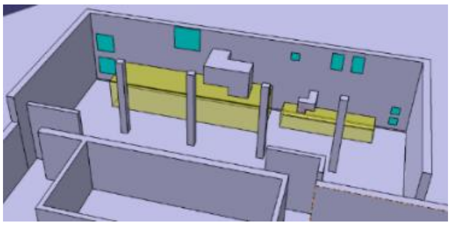

Figure 12: Simplified Geometry Filtration Units

This activity retained in the filters thus becomes the source term for the Monte Carlo calculation, allowing the estimation of the thickness of shielding required or even the definition of the strategy for filter maintenance and management of the relevant waste.

Contribution by direct radiation from containment. Input data and assumptions

To carry out this simulation, it is proceeded in a similar way to that explained for the above cases. Most of baseline data and hypotheses considered, coincide with those already set out throughout this document, so that only those that are particular to this model are mentioned:

- Drawings for determining the geometry of the simplified containment;
- It is considered conservatively that there is no leakage to the environment.

3.2 Simulations

3.2.1. Location

Although the location of the buildings that house the CAGE obey multitude of conditions, among them is clearly identified, and so states the CSN in their design requirements, that it should not be located in areas of predominant winds.
The utilization of ARCON96 code to determine the relative concentrations of radionuclides after a severe accident, allows to know the place where the concentration will be smaller. Especially sensitive to this situation would be the HVAC system, which may relax its demands in comparison to other place where concentrations were higher.

Once executed several cases, the X/Q are determined at different time intervals, providing the necessary data in the next phase.

3.2.2 Inner cloud

Once the input data and the assumptions are introduced, the implementation of the necessary simulations proceeds.

Throughout the project, there have been various adjustments that have enabled IDOM to optimize the design of ventilation systems and sealing requirements of the building in general.

3.2.3 Outer cloud

The determination of this contribution can determine the thickness of the outer walls Fig-13, in charge of providing the shielding necessary to maintain the habitability inside.

It is worth noting, that there may be cases where radiation limitation exceeds the limitation required from the seismic standpoint, prevailing over each other depending on the chosen location.

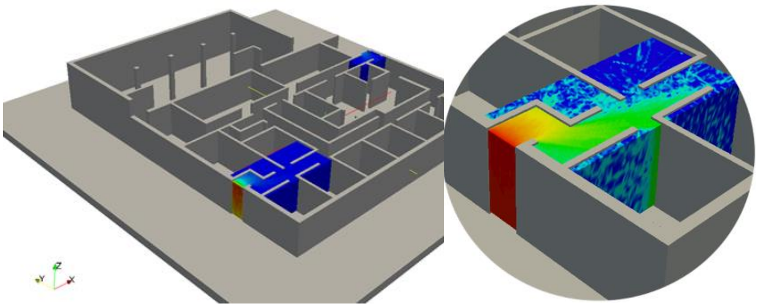

Figure 13: Main entrance detail

3.2.4 Filtering units

Similarly to the previous case, characterization of the source term is required, with the particularity that in this case, the concentration of radionuclides inside the filtration units increases over time, becoming a source term of great contribution to dose.

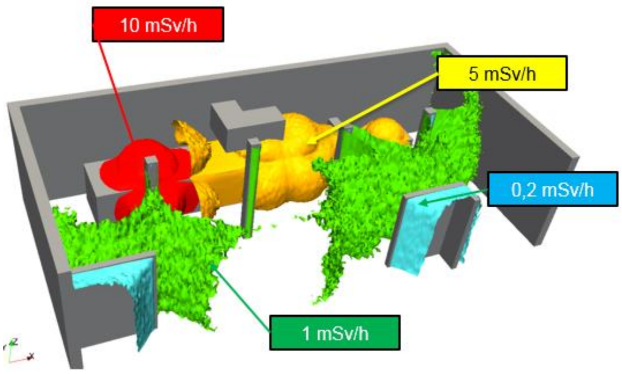

Figure 14: Filtering Units iso-surface dose rates

Because of the magnitude of their contribution in this case the model is more detailed, simulating the materials of the filter housing and filter media. As shown in the previous Fig-14, it has also been necessary to include labyrinthine accesses that reduce the dose.

3.2.5 Containment

This simulation is similar to the outer cloud, with the proviso that the "cloud" is contained within the containment building.
Its contribution is found to be negligible after its evaluation.

4. ONGOING PROJECTS

4.2 Proton-therapy for Quirón Hospital in Madrid

4.2.1 Development of the nuclear design of the facility

Once the project was awarded, some preliminary analyses were carried out:

Documentation of the manufacturer of the proton therapy machine (IBA):

The documentation provided by the manufacturer IBA on its commercial reference Proteus I was analyzed in detail in the specific search for data to characterize the source term to be used (the proton beam in the present case); Information on beam energy (MeV), beam particle fluence and various machine characteristics (details on beam production, beam collimation and modulation detail, maximum and firing energies, materials used in the construction of the equipment, presence of own shields, specific studies, ...) will be necessary for the subsequent 3D design and nuclear modelling.

Documentation on the design of the installation (architectural design):

Architectural designs will be analyzed in detail to determine permissible design ranges and significant constructional details, such as maximum / minimum wall thicknesses, maximum permissible transmission loads to foundations, hollows and penetrations in bunker walls, confinement of radiation, boundaries of adjoining rooms to the bunker, etc.

Documentation on the use of the installation (Quirón Health):

The data related to the use of the installation have been analyzed in detail. Data on the number of expected therapeutic processes, the usual characteristics of the processes (number of pulsations, beam energy, exposure times, ...) depending on their purpose, the expected frequency of each characterized process, degree of occupation of all the rooms, etc., were provided by Quirón Health, in order to evaluate both the individual doses of the different profiles of individuals that are going to circulate through the installation (most exposed individual, compliance with regulatory limits depending on the radiological classification of workers), as the collective doses of the different groups of individuals (workers and public).

4.2.2 Nuclear design

With the extracted data from the previous phases, the problem of shielding and radiation transport is modelled, using specific nuclear codes such as MCNP [1] or similar, approved and validated by the CSN. The tasks associated with modelling are, among others:

- A simplified 3D model is made in the Catia V5 design software for both the main components of the machine and the building. In this model geometry and materials will be characterized, in order to obtain a simplified model, but enough detailed to launch the nuclear calculations;
- The source term associated with the beam (protons and neutrons) is modeled;
- The source term associated with the cyclotron / synchrotron (photons, secondary radiation) is modeled;
- The 3D model will be migrated to the nuclear calculation codes, using specific conversion software;
- The radiation detectors will be positioned in the most characteristic places for the study of the operation of the installation;
- Shielding materials and thicknesses (successive iterations between the 3D model and the nuclear calculation codes) will be characterized.

In addition, the severity of the neutron activation processes of materials are studied in order to discern the importance of these processes and, therefore, the need to repeat the shielding calculations or to determine the exclusion of the use of materials.

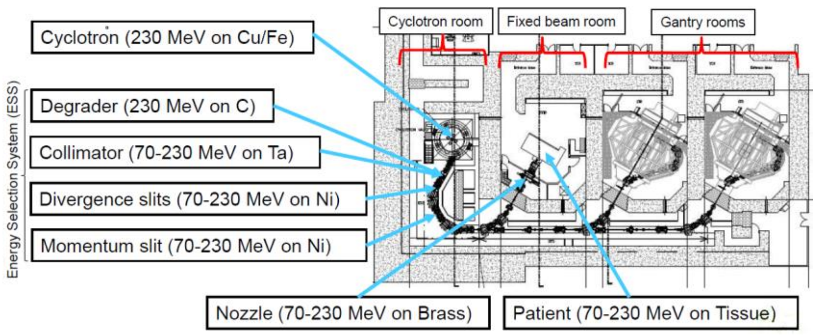

Figure 14: Scheme of a proton-therapy facility

4.3 Shielding assessments for Centro Tecnológico da Marinha em São Paulo (CTMSP)

The shielding has been designed to ensure that neutron and photon radiation dose rates achieve acceptable values under regulatory requirements. This shielding ensures obtaining a controlled area once the source has been shut down.

Shielding assessment has taken into account:

- Source term determination;
- Radiation transport calculation;
- Dose rate assessments;
- Materials' activation by mean of Rigorous Two Step (R2S) methodology.

4.4 Spent fuel casks

In 2014 ENRESA (Spanish Radwaste Management Company) awarded the Engineering Services for the design of the Cask Maintenance Facility to IDOM.
Among the different analyses requested by ENRESA to IDOM, is the design of the Void Spent Fuel Casks Parking (VSPCP), as well as its location in the Centralized Temporary Storage Facility (ATC).

Once its location is established, the dose contribution should be assessed both within and outside the ATC site (<2.5 μSv / year).

The purpose of this study is to evaluate the dose rates as well as the integral doses, which are expected due to the contribution of the empty casks stored in the VSPCP.

The expected doses will be taken at different points around the casks, the arrangement of which is a matrix of 2 x 6 casks. With this it is possible to know the contribution of doses at different distances of the casks matrix, as well as to 100 m, corresponding to the limit of the Controlled Area Fig-15.

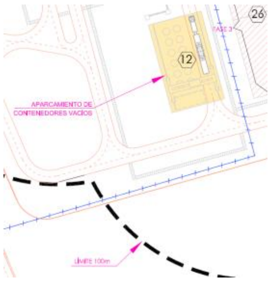

Figure 15: Void Spent Fuel Casks Parking

The following activities have been carried out in order to derive the requested dose rates:

- Calculation of the maximum activity that can contain a cask that is placed in the VSPCP, in order not to exceed 2.5 μSv / year to 100 m of the cask matrix;
- Calculation of the dose rate to 5 cm (in contact) of one of the central casks of the matrix.
- Calculation of the dose rate at 2 m from the cask matrix;
- Calculation of the maximum dose rate from the cask matrix;
- Radiological classification at different distances of the casks;
- Analysis of results.

4.4.1 General methodology:

Firstly, the geometric model (dimensions and materials) is carried out starting from the existing documentation concerning the ENUN 32P cask Fig-17.

The matrix arrangement is then established according to the drawings. In this way the total geometry of the problem is defined, since the models of the 12 casks are arranged in two rows of 6 casks each (2x6 matrix) Fig-16, as well as a simplified model of the surroundings. It should be pointed out that the concrete is also modeled because of its possible effect on the reflection of the γ rays. The surrounding air is also taken into account not to underestimate the contribution of the "skyshine" due to the dispersion by the atmosphere elements nuclei.

In this case, the source term is provided by the technologist based on its experience in La Hague facility. With the isotopic inventory activities, the corresponding spectrum is calculated using the code ORIGEN-S. This spectrum shall be used as input data for the MCNP5 code.

In the first approximation a point source term is established. It is then possible to generate a complete model of radiation transport calculations. With the model of the first approximation the dose corresponding to the simulated energy spectrum is established. It should be noted that the energy spectrum will not change as long as the isotopy is maintained.

While evaluating that the annual dose at 100 m distance does not exceed 2.5 μSv, the corresponding dose rate is also evaluated on contact Fig-17 and at different distances Fig-18.

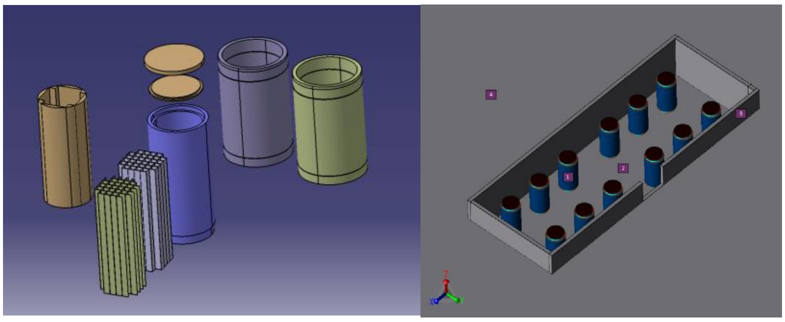

Figure 16: Spent fuel casks elements and matrix configuration

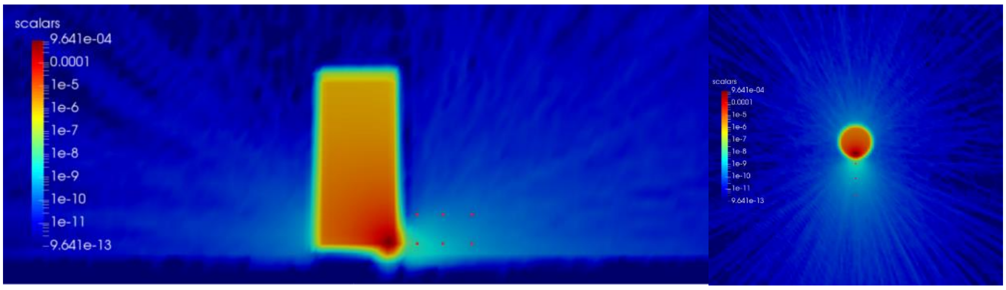

Figure 17: Near field dose rate due to one cask

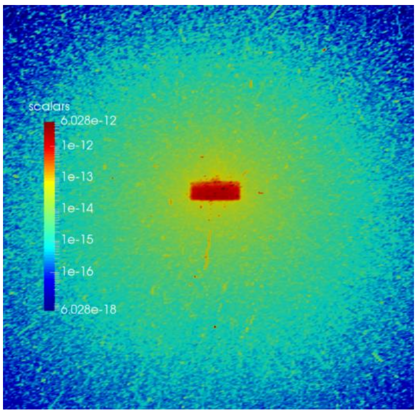

Figure 18: Far field dose rate due to 12 casks

5. CONCLUSIONS

IDOM has applied a general methodology to deal with dose rate assessments where different radiation contributions have to be taken into account. All relevant radiation interactions with matter are considered, including after shutdown dose rate due to neutron activated materials releasing radioactive decay gammas.

Dose rate compliance calculations are assessed, by means of different engineering computational codes, satisfying the simulation verification requirements and being able to verify the radiological design of the models.

Calculation codes selected for the development of this methodology are widely available, internationally used and validated in many studies. Therefore the robustness of the calculations depends primarily on the proper selection of input data and calculation assumptions.

The methodology disclosed herein allows for the modification of any of the parameters, so that it has developed a versatile method of shielding analysis. In-house code development has been needed in order to couple different codes resulting in several innovative tools.

It is worth pointing out that several iterations are required until the final design to imposed dose constraints has been tweaked.

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