EPJ Nuclear Sci. Technol. **3**, 5 (2017)

DOI: 10.1051/epjn/2017004

EPJ Nuclear Sciences & Technologies

Available online at:
http://www.epj-n.org

Regular Article

Open Access

Methodology for the nuclear design validation of an Alternate Emergency Management Centre (CAGE)

César Hueso[1*], Marco Fabbri[1,3], Cristina de la Fuente[1], Albert Janés[1], Joan Massuet[1], Imanol Zamora[1], Cristina Gasca[2], Héctor Hernández[2], and J. Ángel Vega[2]

[1] Idom Ingeniería y Consultoría, Avda. Zarandoa, 23, Bilbao-Vizcaya 48015, Spain
[2] ANAV Asociación Nuclear Ascó-Vandellòs II, L'Hospitalet de l'Infant, Tarragona 43890, Spain
[3] Fusion for Energy, C/Josep Pla, 2, Torres Diagonal Litoral, Edif. B3, Barcelona 08019, Spain

Received: 18 November 2016 / Accepted: 30 January 2017

Abstract. The methodology is devised by coupling different codes. The study of weather conditions as part of the data of the site will determine the relative concentrations of radionuclides in the air using ARCON96. The activity in the air is characterized depending on the source and release sequence specified in NUREG-1465 by RADTRAD code, which provides results of the inner cloud source term contribution. Known activities and energy spectra are inferred using ORIGEN-S, which are used as input for the models of the outer cloud, filters and containment generated with MCNP5. The sum of the different contributions must meet the conditions of habitability specified by the CSN (Spanish Nuclear Regulatory Body) (TEDE <50 mSv and equivalent dose to the thyroid <500 mSv within 30 days following the accident doses) so that the dose is optimized by varying parameters including CAGE location, flow filtering need for recirculation, thicknesses and compositions of the walls, etc. The results for the most penalizing area meet the established criteria, and therefore the CAGE building design based on the methodology presented is radiologically validated.

1 Introduction

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 has subsequently required, the creation of a centre to safely manage an emergency, called an Alternate Emergency Management Centre (CAGE), located at the sites of Nuclear Power Plants [1]. 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 <500 mSv within 30 days following the accident [2]. 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. 1). 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 accumulation of radionuclides in the filters.
- Determination of dose due to proximity to the containment.

* e-mail: cesar.hueso@idom.com

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 h), 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 knowledge of the radiation transport mechanism and of the site meteorological data. To help solve this problem, the ARCON96 (Atmospheric Relative Concentration in building wakes) [3] and RADTRAD 3.03 [4] are applied.

On the other hand, the fact of the radioactive cloud standing around the CAGE becomes a shielding problem where the source term is outside and the people to protect are inside. Therefore, a shielding has to be designed: mainly the concrete walls and doors. After assuming the geometry and applying the radionuclides activity released to the environment (RADTRAD), ORIGEN-S [5] is used to "translate" activities into gamma radiation energy spectra. These spectra are introduced as input data in a Monte-Carlo radiation transport calculation by means of MCNP [6]. This code delivers the outer cloud contribution.

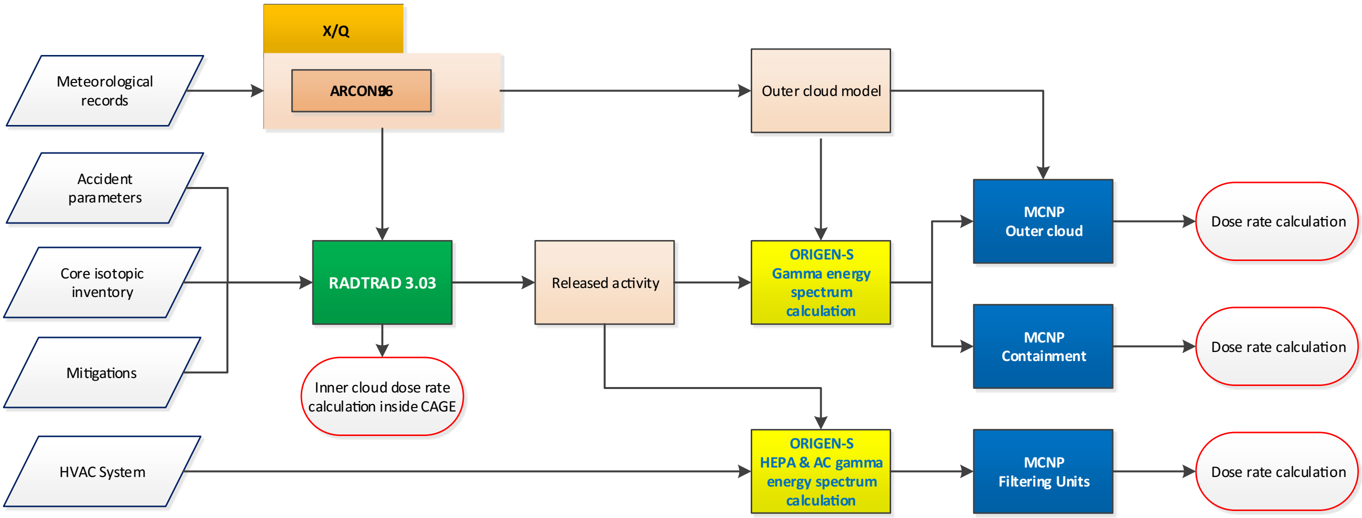

Fig. 1. Methodology applied to determine different dose contributions.

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.

And last but not least, 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 as the filters work, resulting in a strong source term. To perform this calculation, the activities of filtered radionuclides, and their daughter's activities, are taken into consideration. Again thanks to ORIGEN-S, these activities are "translated" into gamma energy spectra to be introduced as input data in a new radiation transport calculation that delivers the dose contribution due to the filtering units.

2 Assumptions and input data

In order to carry out the necessary calculations by coupling the various codes that perform the methodology used to determine dose rates, we must first consider the 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.

2.1 Diffusion factors

According to RG (Regulatory Guide) 1.23 Rev. 1 [7], a Nuclear Power Plant should be able to get the weather information it requires to determine the potential spread of radioactive material from an accident (among other objectives), so the amount of radionuclides resulting from the release into the environment of the considered source term can be deduced. The ARCON96 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 [8].

The following structure summarizes the different steps carried out to calculate the atmospheric relative concentrations (X/Q) of radioisotopes:
- Obtaining meteorological data.
- Process meteorological data.
  - Calculation of hourly averages.
  - Calculation of atmospheric stability.
  - Generation of meteorological files for ARCON96.
- ARCON96 execution.

2.2 Obtain and process meteorological data

Weather information are provided by specific files, including the matrix of hourly frequencies, defined from the following time averages:
- Wind speed (in m/s) at different heights.
- Wind direction (in degrees) at different heights.
- Category stability (Pasquill, from A to G).

2.2.1 Hypothesis

RG 1.194 considers representative hourly weather observations for more than 5 years.
- Height measurement data at 10 m and 29 m.
- An emission at ground level is assumed; conservative assumptions at the selected location distance.
- Conservatively, a height equal to intake 0 m is assumed.
- A 'terrain elevation difference' is taken equal to 0 m, since no data are available about it.
- Building line perpendicular to the direction of the release section.
- The angle between the CAGE and the emission source, taking care not to locate the building in a predominant wind window.
- 90-degree wind window is taken.
- Distance from the emission point to CAGE: measured on the ground.
- Minimum wind speed 0.5 m/s.
- Surface roughness of 0.20 m.
- The initial values of σ_y and σ_z are equal to 0, as advised in RG 1.194.

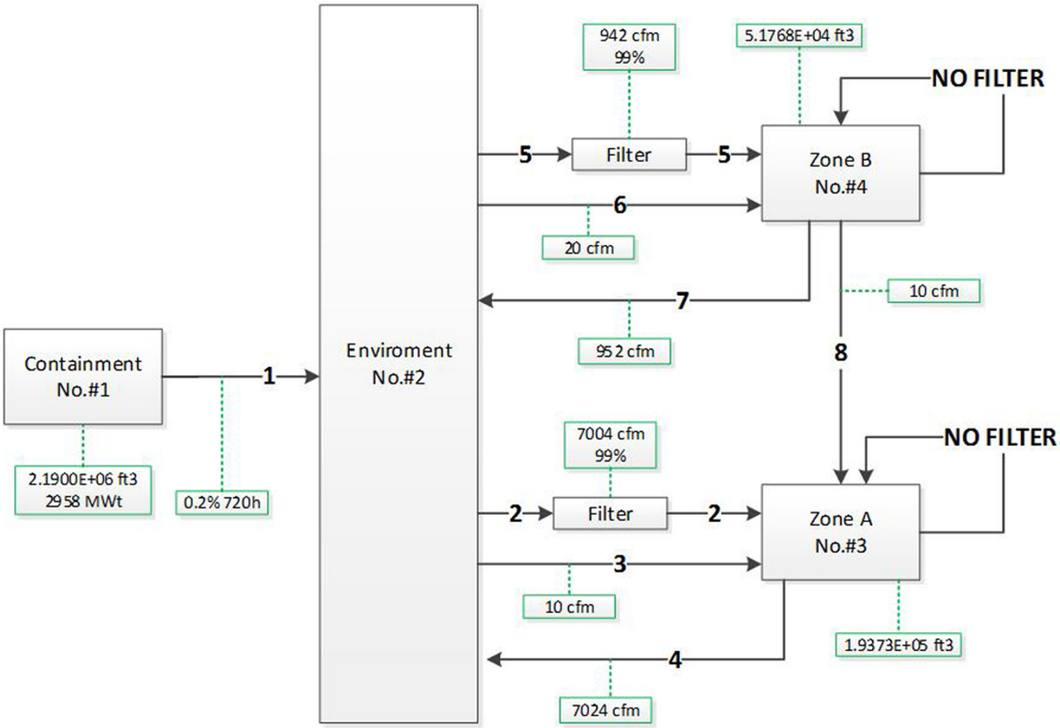

Fig. 2. Model of RADTRAD.

2.3 Determination of dose

Once the diffusion factors have been obtained, and therefore the relative concentration of radionuclides known in the points to 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.

2.3.1 Inner cloud contribution. Input data and assumptions

Determining dose inside the cloud will take place through the software RADTRAD, as shown in Figure 1. As specified in NUREG-1465 [9] and the RG 1195 [10], it is a code that incorporates adequate methodologies to meet dose determination.

Then the necessary data and hypotheses considered are as follows:
- Diffusion factors or X/Q obtained through the ARCON96 code.
- Flow diagram of HVAC system.
- Decontamination factor by natural deposition of elemental iodine.
- Chemical composition of radio-iodine, extracted value from NUREG-1465.
- Containment volume.
- Thermal power of the reactor.
- Reactor core inventory, assumed to be consistent with the inventory from the post-Fukushima stress test project.
- Overpressure flow calculation to define the radiological classification of CAGE.
- Net volumes of each of the areas of the CAGE.
- FGR 11: Limit values of Radionuclide Intake and Air concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion. 1988 [11].
- FGR 12: External Exposure to Radionuclides in Air, Water and Soil [12].

Furthermore, with regard to assumptions, the following are considered:
- Two different zones are considered in the CAGE. Zone A, which will be in excess of pressure compared to Zone B. Note that both of them are in overpressure relative to the atmospheric pressure (Fig. 2).
- Release rates in the event of a severe accident in reactors PWR/BWR are introduced into the RADTRAD code. The severe accident definition is in line with the post-Fukushima stress test accident definition. Note that the release fraction and timings for severe accident in RADTRAD are the ones from NUREG-1465, Tables 3.12 and 3.13.
- Isotopes of the source term are introduced by the corresponding external .nif file.
- Loading factors defined in RG 1.195 are used.
- Breathing rates according RG 1.195, being $3.5\mathrm{E}^{-4}\ \mathrm{m}^3/\mathrm{s}$ at 720 h.
- Consideration is given to radioactive decay.
- Inflow of air and recirculation values established by the HVAC system are set for Zones A and B.
- In the compartment defined as containment credit is given to the natural deposition.
- A release of radionuclides to the environment is estimated corresponding to 0.2% of the containment volume per day during the 30 days of the postulated accident (not only during the first 24 h as specified in RG 1194 [8]) to add conservatism to the calculations.

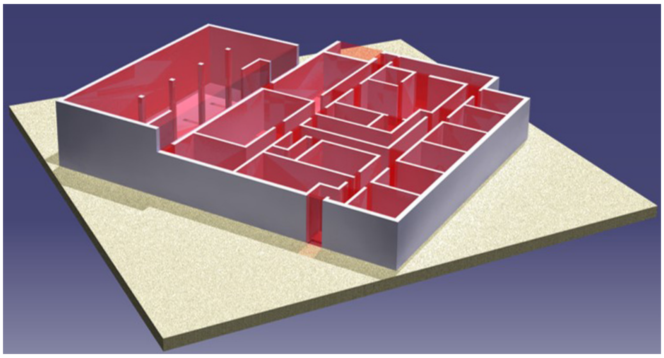

Fig. 3. Simplified geometry CAGE.

- Outside contaminated air inlet is considered to be filtered by HEPA and active carbon filtering units.
- Infiltration flow is introduced from the outside contaminated environment into Zone A at 10 cfm and Zone B at 20 cfm, in line with the RG 1.78 [13].
- The discharge rate compensates for infiltrations and the supply flow.
- Infiltrations flow from Zone B to Zone A at 10 cfm.
- Several time steps are considered for evaluation, according to the diffusion factor or X/Q.

2.3.2 External cloud contribution. Input data and assumptions

Determining dose provided by the outer cloud to CAGE is carried out by coupling the software RADTRAD 3.03, ORIGEN-S and MCNP as indicated in the flow diagram of Figure 1. 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 surrounding the CAGE is obtained. Likewise, the corresponding gamma energy spectrum is determined by the software ORIGEN-S, in which the activities obtained for each time interval are entered as input data. Finally, these gamma spectra are introduced in their respective MCNP5 simulations in order to characterize the cloud corresponding to the outer volume source. Also needed are:

- Simplified but representative model of the geometry of the CAGE.
- Modeling the radioactive cloud as a semi-cylindrical source.
- Defining the parameters of interest of the simulation and the respective locations where these values are extracted (Fig. 3).

Therefore, we consider as input data:

- Isotopic activity released to the outside environment.
- Diffusion factors or X/Q obtained through ARCON96 code.
- Figures 4 and 5 of RG 1.194 are used to determine the diffusion coefficients σ_z and σ_y. Note that once all the parameters in equation (1) from Section 3.2 of the NUREG/CR6331 [3] are known, the distance from the centre of the plume (i.e. parameter y from the equation (1) of the present paper) can be calculated for each time step. This y parameter allows for the cloud volume definition.
- It is assumed that the external environment is in the stability class of type G since thereby stability coefficients are minimized.
- "Still air" speed is assumed to be 0.5 m/s.
- Project drawings for the determination of the simplified geometry of CAGE.
- Gamma energy spectra for the characterization of the volume source term corresponding to the outer cloud over the CAGE.
- Photon libraries MCPLIB84 included in the MCNP5 package.
- Radiological properties of the materials considered in the CAGE.

And the main assumptions are as follows:

- For simplistic effects it has been assumed that the cloud over the CAGE has a semi-cylindrical geometry. Therefore,

$$V_{\text{cloud}} = \frac{1}{2}\pi y^2 L, \tag{1}$$

V_{cloud} (m^3), y (m), L (m).

- At each time step, a uniform concentration is assumed.
- Significantly, it is assumed that the transport of the release to the CAGE is instantaneous.
- The most representative concrete walls of the CAGE are modeled.
- CAGE slabs and the outer ground are modeled to take into account backscattering.
- No accumulation of radioisotope within the CAGE.
- The contribution of the neutron dose determination is neglected.
- In each time step, an intensity corresponding to the selected volume source spectrum previously calculated by ORIGEN-S is assumed.
- The weighting factor for the thyroid equivalent dose due to direct radiation is considered.
- Dose conversion factors are assumed according to ICRP119 [14].

2.3.3 Contribution accumulation in filters. Input data and assumptions

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 this should generate new models to get the amount retained in filters (Fig. 4).

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.

The input data and particular hypotheses of this case would be:

- Volumetric flow HVAC system.
- Project drawings for the determination of the simplified geometry of the filters.
- Gamma energy spectra for characterization corresponding to the accumulation of radionuclides in the filtering units of the CAGE volume source.

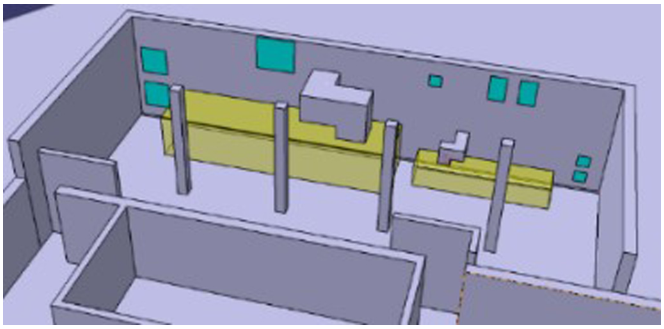

Fig. 4. Simplified geometry filtration units.

- All radionuclides will accumulate in the "virtual" filters, decaying in that filter.
- The activity generated by descendants is also considered in the accumulation of radionuclides in the filters.
- Inside the room HVAC, we model only the two filter units, the walls corresponding to the labyrinths and walls defining this room.
- The outer dimensions of the filter units are modeled using the geometric data of the drawings while the inner components are detailed by the typical constructive values.
- In this simplified model, we consider the dose due to filters to be especially represented in three zones: the interior of the room itself, protected areas through out the mazes, and inside the CAGE immediately behind the door in the area adjoining the HVAC room.
- Intensities resulting from the sum of the accumulation of isotopes in each time interval plus its corresponding decay daughters are considered. Thus, the fission products that are decaying in the filter at different time intervals are always considered.

2.3.4 Contribution by direct radiation from containment. Input data and assumptions

To carry out this simulation, we 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 only those that are particular to this model are mentioned:

- Drawings for determining the geometry of the simplified containment.
- We consider conservatively that there is no leakage to the environment.
- Variance Reduction Techniques are used in the MCNP5 code because of the model dimensions and the shielding thicknesses that must be traversed.

3 Simulations

3.1 Location

Although the location of the buildings that house the CAGE obeys a multitude of conditions, one among them is clearly identified and so states the CSN in their design requirements: it should not be located in areas of predominant winds. If setting the building on one of these wind windows is mandatory, the design requirements in other areas will be influenced negatively. Each unit must locate the CAGE taking into account all factors so that it can optimize expenditures.

Using ARCON96 code to determine the relative concentrations of radionuclides after a severe accident allows us to identify the region of minimal concentration. Especially sensitive to this situation would be the HVAC system, which may relax its demands in comparison to other places where concentrations were higher.

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

3.1.1 Inner cloud

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

The required results are TEDE and equivalent dose to the thyroid.

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

As an example, it may be mentioned that filtering recirculation is not required in the case of radiological accident, allowing for cost optimization of the HVAC system.

3.1.2 Outer cloud

As previously stated, the radioactive cloud is a volumetric source term of gamma radiation, so we must consider its contribution to the integral dose. This contribution can determine the thickness of the outer walls, which provide the shielding necessary to maintain habitability inside (Fig. 5).

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.

When it comes to characterizing a source term, one must know its energy spectrum and the emission intensity (γ/s). As to equal activity, the contribution to the dose will depend on the isotopes considered.

This characterization of the source term, along with the model geometry, the definition of materials and measuring points (tallies), defines the "input" of MCNP5. This code allows the determination of the direct radiation dose at different points defined by the user.

It should be noted that the outer cloud model has been hypothesized in various ways before finally opting for a semi-cylindrical representation that is considered conservative because it homogenizes the limiting concentration at the selected location.

3.1.3 Filtering units

Similar 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

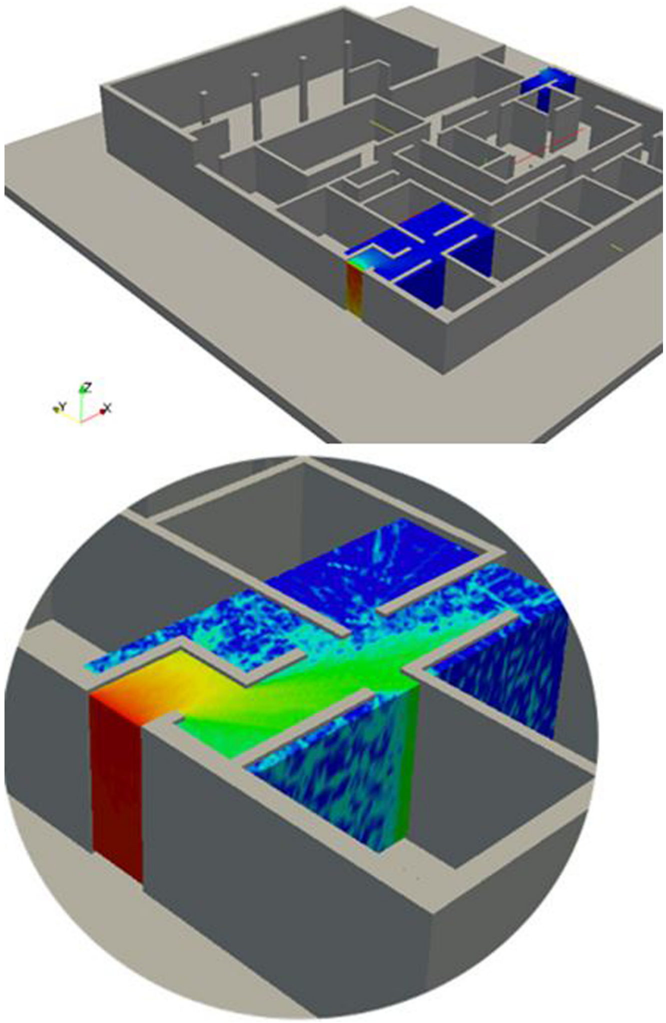

Fig. 5. Emergency exit detail.

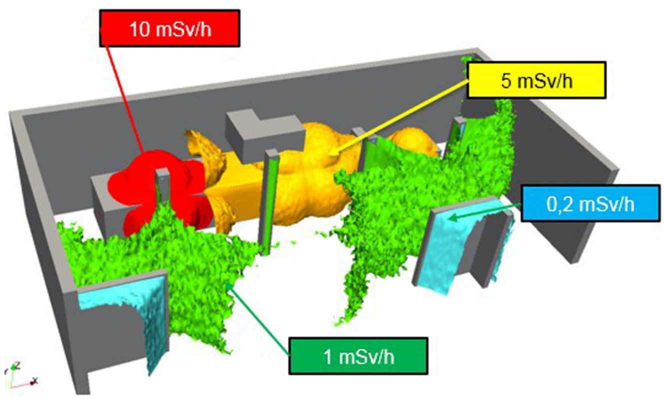

Fig. 6. Filtering units iso-surface dose rates.

contribution to dose. The use of ORIGEN-S code allows us to take into account the daughters; isotopes that are added to the retained radionuclides themselves (Fig. 6).

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 figure, it has also been necessary to include labyrinthine accesses that reduce the dose.

Thanks to the results obtained by these simulations, the dimensions of the mazes and thickness of the separating wall of the room ventilation from rest of the building have been determined.

3.1.4 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.

4 Conclusions

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, making it a versatile method of radiological analysis.

Interaction with other areas' colleagues (HVAC specialists, civil design, etc.) is key in the selection of the required information.

On the other hand, knowledge of the design basis of the NPP where the CAGE is located is essential to avoid incurring contradictions. It is fundamental that there be a fluid communication among all the interested parties.

The HVAC room must be an area of limited access to staff, managed directly by the Radiation Protection Department, due to the dose rates obtained in the HVAC room.

After applying the methodology defined in this paper, the CAGE design is validated in compliance with the CSN requirements.

It is worth pointing out that several iterations are required before designing the building in order to ensure that dose constraints have been tweaked.

References

1. Consejo de Seguridad Nuclear, *Instrucciones Técnicas Complementarias. ITC's a las Autorizaciones de Explotación* (2011)
2. Consejo de Seguridad Nuclear, Acta del Pleno 1.297, Anexo 4 *Criterios de Evaluación del CAGE* (2013)
3. U.S. Nuclear Regulatory Commission, NUREG/CR-6331 ARCON96 *Atmospheric relative concentrations in building wakes* (1997)
4. U.S. Nuclear Regulatory Commission, NUREG/CR-6604: RADTRAD *A simplified model for RADionuclide Transport and Removal and Dose estimation* (1998)
5. Oak Ridge National Laboratory, *SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design* (2011)
6. Los Alamos National Laboratory, *MCNP – A General N-Particle Transport Code, Version 5* – Volume I: Overview and Theory, X-5 Monte Carlo Team. LA-UR-03-1987 (2003)
7. RG 1.23, *Meteorological monitoring programs for nuclear power plants*. Rev. 1 (U.S. Nuclear Regulatory Commission, 2007)
8. RG 1.194, *Atmospheric relative concentrations for control room radiological habitability assessments at Nuclear Power Plants* (2003)

9. U.S. Nuclear Regulatory Commission, NUREG-1465: *Accident Source Terms for Light-Water Nuclear Power Plants* (1995)
10. RG 1.195, *Methods and assumptions for evaluating radiological consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors* (2003)
11. EPA, *FGR 11: Limit Values of Radionuclide Intake and Air concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion* (1988)
12. EPA, *FGR 12: External Exposure to Radionuclides in Air, Water and Soil* (2016)
13. RG 1.78, *Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release* (2001)
14. ICRP, Compendium of Dose Coefficients based on ICRP Publication 60. ICRP Publication 119. Ann. ICRP **41** (Suppl.)(2012)

Cite this article as: César Hueso, Marco Fabbri, Cristina de la Fuente, Albert Janés, Joan Massuet, Imanol Zamora, Cristina Gasca, Héctor Hernández, J. Ángel Vega, Methodology for the nuclear design validation of an Alternate Emergency Management Centre (CAGE), EPJ Nuclear Sci. Technol. **3**, 5 (2017)