MASTER NUCLEAR ENERGY

Master report:
Dose rate calculation for future ULYSSE reactor dismantling

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[Ref 1]: DEVELOPMENT OF EDUCATION AND TRAINING PROGRAMS USING ISIS RESEARCH REACTOR F. FOULON, B. LESCOP

INSTN, CEA/Saclay, France

[Ref 2]: Study of the temperature effects on ULYSSE reactor for training and qualification of operating personnel, F.Foulon, INSTN/UEIN, CEA/Saclay, France

[Ref 3]: Ulysse – INB 18.Rapport préliminaire de sûreté des opérations de démantèlement. CEA

[Ref 4]: Testing of Different Data Libraries in Activation Analysis of Concrete and Graphite from Nuclear Installations

M. Cometto, D. Ancius, and D. Ridikas* CEA Saclay, DSM/DAPNIA/SPhN, F-91191 Gif-sur-Yvette, France

[Ref 5]: J. F. Briesmeister for Group X-6 "MCNP-A, A General Monte Carlo Code for Neutron and Photon Transport", Version 4A, LANL, report LA-12625-M (1993).

[Ref 6]: W. B. Wilson, T. R. England and K. A. Van Riper "Status of CINDER'90 Codes and Data", Los Alamos National

Laboratory, report LA-UR-99-361 (1999).

[Ref 7]: Offre Technique.Démantèlement et assainissement des locaux constituant l'INB 18 – ULYSSE CEA-SACLAY.STMI, Areva. DPE.IDF.2012.PRO.112.TECH(2012)

[Ref 8]: Critères radiologiques d'acceptation des déchets TFA. Andra. SUR.SP.AMES.02.0007(2013)

[Ref 9]: Approaches used for Clearance of Lands from Nuclear Facilities among several countries, Robert A.Meck

[Ref 10]: Autorité de Sûreté Nucléaire (ASN) [Nuclear Safety Authority].On Transparency and Security in the Nuclear Field.Paris: ASN, 2006. ACT No. 2006-686 of 13 June 2006.

[Ref 11]: ASN Program on the sustainable management of materials and radioactive waste. Paris: ASN, 2006. Act No. 2006-739 of 28 June2006.

[Ref 12]: Ministre d'Etat. Decree No.2007-1557 of November 2, 2007 relating to nuclear installations and control under the Nuclear Safety of transport of radioactive substances. Paris: Ministry for Ecology, Sustainable Development and Spatial Planning, 2007.Official Journal no. 255 of 3 November 2007 page 18026 text no.3.

[Ref 13]:ASN.ASN's Policy on Dismantling and decommissioning of nuclear facilities in France. Paris : ASN, 2009. Version 0.v3

[Ref 14]: Guide de l'ASN n°14

[Ref 15]: « INB 18, Etude déchets dans le cadre des opérations conduisant au déclassement du réacteur ULYSSE », CEA, Saclay (2008)

[Ref 16]: INB 18, Dossier de Demande de prise en charge des déchets TFA, CEA, Saclay (2007)

[Ref 17]:MERCURE : UN CODE 3D INDUSTRIEL DE PROPAGATION DES GAMMA PAR LA METHODE D'ATTENUATION EN LIGNE DROITE. APPLICATIONS EN RADIOPROTECTION, C. Suteau, M. Chiron et L. Luneville

[Ref 18]: MERCURAD® – 3D Simulation Software for Dose Rate Calculation, CANBERRA

[Ref 19]: MICROSHIELD, http://www.radiationsoftware.com/mshield.html

[Ref 20]: MICROSHIELD® User's Manual, Grove Software, Inc. (2009)

[Ref 21]: Notice NARMER-1 version 4.2 DEN/DANS/DM2S/SERMA/LTSD/RT/11-5115/A (2011)

[Ref 22]: POINT-KERNEL METHOD FOR RADIATION FIELDS SIMULATION, I.M. Prokhorets, S.I. Prokhorets, M.A. Khazhmuradov, E.V. Rudychev, D.V. Fedorchenko(2007)

[Ref 23]: Java-based Nuclear Data Information System http://www.oecd-nea.org/janis/

[Ref 24]: http://www.asn.fr/

[Ref 25]: http://www.cea.fr/

[Ref 26]: http://www.areva.com/

[Ref 27]: TRIPOLI-4 Project Team, Tripoli-4 User Guide, Report CEA-R-6169, Commissariat à l'Energie Atomique, (2008).

[Ref 28]: J. P. BOTH, H. DERRIENNIC, B. MORILLON, and J. C.NIMAL, "A Survey of TRIPOLI-4," Proc. 8th Int. Conf. Radiation Shielding, Arlington, Texas, April 24–28, 1994, Vol. 1, p. 373, American Nuclear Society ~1994!

[Ref 29]: "MCNP-4C General Monte Carlo N-Particle Transport Code, Version 4C," J. F. BREISMEISTER, Ed., LA-12625-M, Los Alamos National Laboratory ~2000!

[Ref 30]: C. Suteau, M. Chiron and G. Arnaud, Improvement of MERCURE-6's General Formalism for

[Ref 31]: ETUDE DU DEMANTELEMENT DE L'INB 18 – ULYSSE, LOT 2, SCENARIO DE DEMANTELEMENT, SQP-IG/MD.02.A Révision 1

[Ref 32]: LARA, Library of gamma and alpha emission, http://laraweb.free.fr/

[Ref 33]: Nucléide-Lara Bibliothèque des emissions alpha, X et gamma, Marie-Martine Bé, Vanessa Chisté, Christophe

Dulieu, CEA, Saclay , Fance (2008)

[Ref 34]: GEANT4, http://geant4.web.cern.ch/geant4

[Ref 35]: MCNPX™ user's manual, version 2.7.0, April 2011, Denise B Pelowitz, editor

[Ref 36]: Calculating Gamma-Ray Buildup Factors in Multilayer Shields, Nucl. Sci. & Eng. 147 (2004), pp. 419-424.

[Ref 37]: DOSIMEX, « Calculs de doses générées par les rayonnements ionisants », A. Vivier, G. Lopez, edp science, CEA, INSTN

[Ref 38]: ICRP PUBLICATION 74, "Conversion Coefficients for Use in Radiological Protection Against External Radiation," Adopted by the ICRP and ICRU in September 1995.

[Ref 39]: The JEF-2.2 Nuclear Data Library, NEA JEFF Report 17 (2000).

[Ref 40]: Démantèlement et assainissement des locaux constituant l'INB 18 –ULYSSE,CEA-SACLAY, DPE.IDF.2012.PRO.112.TECH (17/09/2012)

[Ref 41]: STMI http://www.areva.com/FR/activites-1244/stmi-societe-des-techniques-en-milieu-ionisant.html

[Ref 42]:Ulysse-INB 18 Dossier de demande de mise à l'arrêt définitive et de démantèlement

[Ref 43]: Activation analysis of graphite and concrete in the experimental reactor RUS M. Cometto, D. Ridikas, M. C. Aubert, F. Damoy, D. Ancius, CEA Saclay, DSM/DAPNIA F-91191 Gif-sur-Yvette, France 2004

[Ref 44]: NIST http://physics.nist.gov/PhysRefData/XrayMassCoef/tab4.html

[Ref 45]: C. Suteau, M. Chiron and G. Arnaud, Improvement of Mercure-6's General Formalism for Calculating Gamma-Ray Build-up Factors in Multilayer Shields, Nuclear Science and Engineering 147, 43-55 (2004)

[Ref 46]:DONNEES TECHNIQUES POUR LA FABRICATION DES CAISSONS METALLIQUES DE 5M3 AT 10 M3 DESTINES AU CONDITIONNEMENT DES DECHETS RADIOACTIFS DU CEA SACLAY, NT CEA/STED /CCAE/SP/531 (2011)

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2 INTRODUCTION

This report is about my internship at CEA in Saclay covering the period of 5 months from April to September 2014.The subject of internship was the assessment of the dose rate for the reactor block cutting during the future dismantling of ULYSSE training reactor. During this internship, a bibliography, computation and analysis tasks were performed in order to fulfill the objectives.

In the report are collected the most relevant analysis and discussion as followed:

- Firstly, in the introduction of the report, a brief description of the reactor and its aims can be found. It is followed by an overview of the operations that have been already finished by today in the preparation of dismantling and the ones planned. A brief review of the Nuclear Waste strategy in France which is important for understanding the starting point of dismantling operations that are to be performed on ULYSSE is also given.
- Later on, the results on material activation computed by CEA will be discussed. Softwares for dose rate computations will be presented and their results compared. This includes the analysis of different phenomena related to the exponentially activated concrete.
- The collective dose for works will be calculated. In this purpose, a simple tool is developed that is enabling to change important parameters of dismantling operation and see how they affect the total collective dose.
- Finally, an optimized storage and arrangement of the cut blocks depending on the activity magnitude will be proposed.

This study was lead at the INSTN with the help of DPAD, and more precisely the technical support of M. Jeanjacques. The DPAD, Departement Projet Assainissement Démantèlement, is in charge of all the management of dismantling operations (studies and works) in the CEA.

Figure 1: ULYSSE training reactor [Ref 3**]**

3 PRESENTATION OF THE ULYSSE TRAINING REACTOR [Ref 2] [Ref 3]

3.1 Reactor ULYSSE

ULYSSE reactor is an Argonaut type reactor with a nominal power of 100 kW that was especially designed for training courses. It was constructed between January and June 1961 and was started up in July 1961. From this date it was mainly used for training courses and experimental purposes until the decision to shut down the reactor, after more than 40 years of operation, was taken in 2003. After the restart of ISIS reactor [Ref 1] in 2006, ULYSSE reactor was definitively shut down in February 2007 and the training activities were transferred on ISIS reactor in March 2007.Note that 90% of energy was produced between 1961 and 1980, due to the importance of neutron studies performed during the three first decades of operation of the installation.

ULYSSE reactor is located within the French Atomic and Alternative Energy Commission (CEA [Ref **25**]) center at Saclay, it is inside the INSTN infrastructure.

3.2 Description of the reactor core

ULYSSE reactor is water moderated and cooled reactor, surrounded by graphite reflector. It is using MTR (Materials Testing Reactor) type fuel elements with high enriched uranium.

The main characteristics of ULYSSE are as follow:

- thermal power: 100 kW,
- moderator: water
- reflector: graphite
- maximum thermal neutron flux in the core: 1.4 $\times 10^{12}$ n/cm^2/s
- mean thermal neutron flux in the core: 5 $\times 10^{11}$ n/cm2/s
- maximum fast neutron flux in the core (E > 0.1 MeV): 1.4 $\times 10^{12}$ n/cm2/s
- dimension of the core (mm): diameter 900,height 600
- fuel: enriched uranium 90% ^{235}U,
- core inlet temperature: 20°C,
- core outlet temperature: 25°C,
- primary core flow rate: 16 m^3/h
- secondary flow rate: 10 m3/h

The core of the reactor is located in the center of a concrete block that ensures radiation shielding. The aluminium vessel which is surrounded by graphite external reflector contains the annular core. The Figure 2 illustrates the horizontal cross section of the core and its surroundings. The vessel has diameter of 90 cm and a height 150 cm.

The reactor core contains:

- 24 external fuel elements arranged around an external crown,
- 24 internal graphite elements arranged around an internal crown,
- a central graphite reflector (60 cm in diameter)
- and 23 graphite wedges between these components.

The graphite components placed in the core have an aluminium cladding. Each fuel element is sleeved by an aluminium box containing 11 fuel plates separated by a distance of about 4,7 mm where water, which is the main moderator flows. The reactor is operated with 6 control rods, in order to hold down the control of reactivity. Control rods are made of cadmium plates placed in aluminium wagons. The rods are moved vertically in the external graphite reflector close to the core as illustrated on Figure 2 and Figure 3.

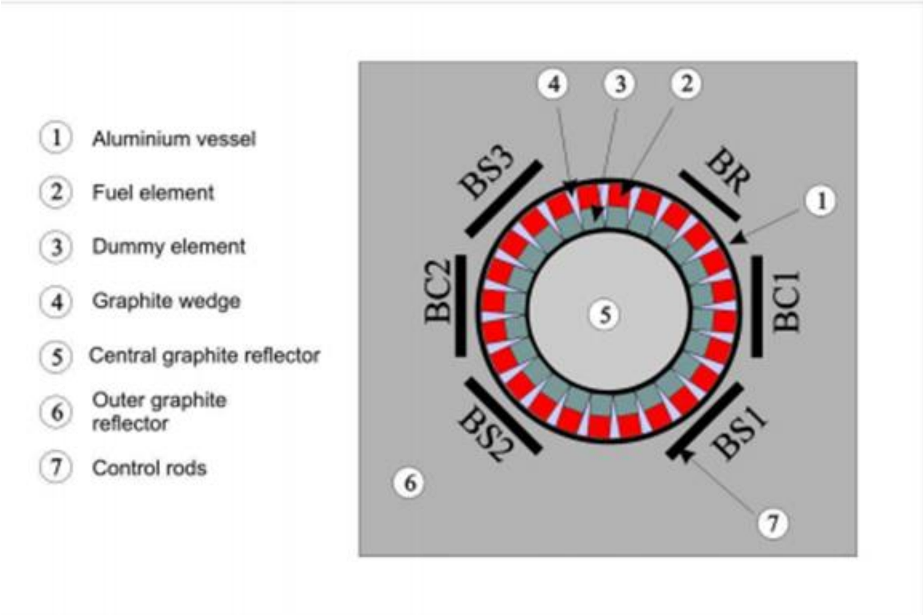


Figure 2: Schematic representation of the reactor core (horizontal cross section) [Ref 2]

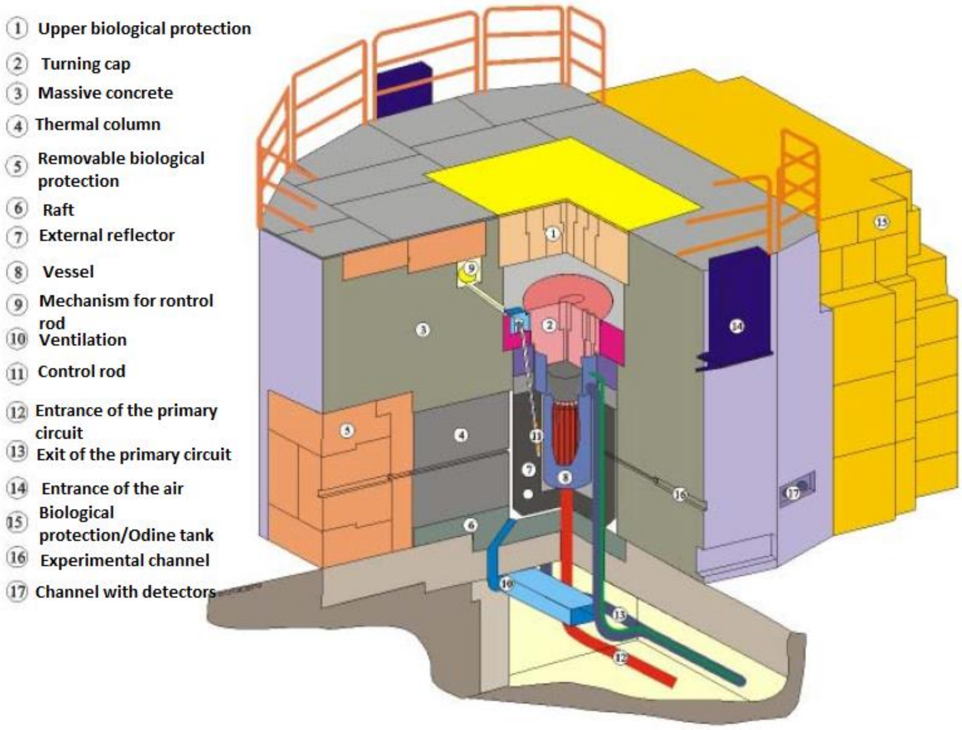


Figure 3: Schematic representation of ULYSSE reactor

3.3 Description of the reactor block

The reactor block is a parallelepiped, approximately 7.40 m long, 6.00 m wide and 4.10 m high. This block mainly consists of concrete surrounding the reactor vessel playing the role of mechanical support and biological protection with regard to the ionizing radiation emitted by the core of the reactor during its operation. This allowed the operating staff to work in the hall while reactor was in operation.

From the inside outwards the reactor block was composed of:

- Reactor vessel, containing light water serving as a moderator and coolant, the twenty four fuel elements with the inner reflector,
- Control rod and their mechanisms ,
- External reflector and graphite thermal columns,
- A compound of different concrete solid (conventional concrete, heavy concrete and concrete with boron, and heavy concrete with boron), and a set of movable concrete blocks housing a testing device (“Ondine“ water tank 7m^3 which was used for irradiation experiments under water),
- A set of caps and protective slabs, especially for unloading core.

The concrete block is pierced by a set of experimental channels (for placing samples and measuring devices near the core) and a channel containing the core (see Figure 4).

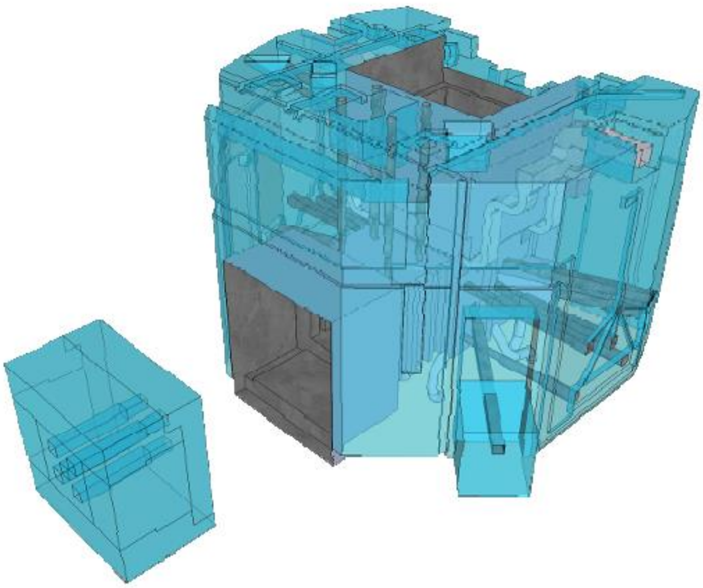


Figure 4 : 3D model of Ulysses reactor massive concrete block drilled with channels [Ref 3]

The fixed part of the concrete foundation rests on the ground and its essential role is to support the other components of the reactor (vessel, reflector, control rods…). At the same time, this part which is made out of heavy concrete reinforced with a steel frame ensures biological protection made out of heavy concrete reinforced with a steel frame.

The upper part of the massif has in its center a rectangular cavity in which is housed the outer reflector bearing. The floor under the reflector is borated concrete that contains about 2 % hydrated calcium borate. The boron is present in the concrete as a neutron absorber.

The reactor block forms a tunnel, which contains the thermal column made of graphite. Graphite slows down the neutrons in order to give irradiation channels with mainly thermal neutrons.

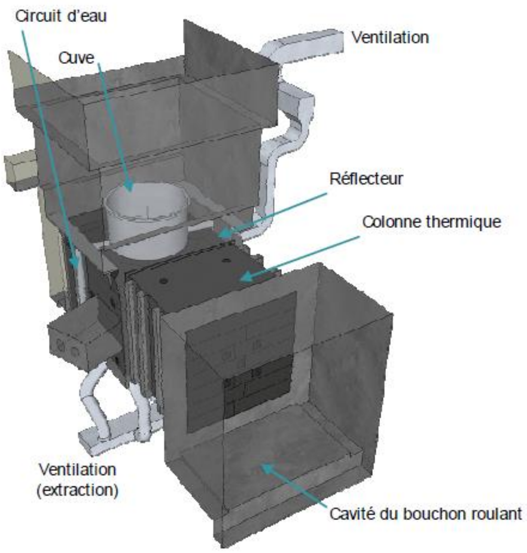


Figure 5: 3D representation of the graphite thermal column, reflector, core and ventilation system [Ref 3**]**

The western cavity is closed by this graphite thermal column and then in addition by a movable block.

Biological protection of the eastern cavity is ensured with movable concrete blocks. These mobile blocks may be removed in part or in whole, during dismantling process. Moving blocks are, depending on their position relative to the core, ordinary concrete, heavy concrete, or heavy concrete with boron. Heavy concert has density of 3.5 g/cm3 while conventional concrete 2.3 g/cm3.

Due to this consideration, the impact of the density on the dose rate will be studied.

4 DISMANTLING OPERATIONS AND COLLECTIVE DOSE

4.1 Introduction

The knowledge of the past and planned dismantling operations is a prerequisite for this internship. A bibliography task and comprehensive survey were conducted in order to have a better understanding of the dismantling operations already done and the one that are planned. In this section are reported the most relevant operations for the core removal and the near future dismantling operations.

Decommissioning is carried out under stringent regulatory regime in order to ensure that the safety, security and well-being of the workforce and the public are maintained and the protection of environment is not compromised in any way. For obtaining the decree for dismantling CEA had to submit following documentation to ministers:

- Initial state of the plant
- Main steps of dismantling
- Various Maps
- Safety analysis report
- Description of the plant premises
- Environmental impact study
- Operating rules for monitoring and maintenance
- Risk evaluation

Those documentations were sent in June 2009 and updated upon request of the Regulatory body in September 2011. Public Inquiry obtained in 2012 was in favor of dismantling. Currently, ULYSSE reactor is waiting to obtain the decree. In 2013, the primcontractor, Areva- STMI [Ref 41] was chosen for preforming the operations.

4.2 Operations finalized until 2012

After the final reactor shutdown, the fuel elements were unloaded from the core and placed in the storage pool in February 2007. The 24 fuel elements were later on evacuated to the La Hague plant of Areva NC in January 2008. Transportation was carried out by TN International with packaging TN-MTR after obtaining permission from the Nuclear Safety Authority (DEP letter ORLEANS - 0945-2007 22 August 2007).The experimental channels have been emptied and the neutron sources used for reactor startup have been removed from the reactor block. The other operations carried out between 2008 and 2012 included the draining of the water circuits, the removal of experimental devices and the evacuation of all neutrons sources and nuclear materials.

4.3 Future dismantling operations

The dismantling operations will be carried out in three different phases. In the Table 1 are summarized the main steps for each batch, their duration and initial estimation of collective dose rate for operation. This estimation was done by Amec Spie, in a pre-study of the reactor dismantling. Based on the report dated in 2006 [Ref 31], **the total collective dose for dismantling operation was**

estimated to ~2.64mSv. In safety report [Ref 42], *it was decided to take into account a margin and consider a total collective dose for dismantling operation of 3mSv for a team of 10 people among* ***6 operators.***

BATCH	Phase	Actions	Duration	Collective Dose mSv
Non-Nuclear	A-0	Site preparation	1 month	0,05
	A-1	Dismantling of the chimney and galleries	2 months	0,02
	A-2	Dismantling of the horizontal pit	1 month	0,08
	A-3	Dismantling of the pool	2 months	0,03
	A-4	Dismantling of the vertical pits	1 month	0,02
	A-5	Layout of the waste storage area for LLW	1 month	/
Nuclear Part	B-1	Removal of Odine tank	2 months	0,19
	B-2	Dismantling of the core equipment	1 month	0,31
	B-3	Removing of the west side cap and graphite	2 months	0,6
	B-4	**Dismantling of the conventional concrete**	**4 months**	**0,53**
		Dismantling of the nuclear part of concrete and ground	**2 months**	
Final Assessment	C-1	Final Radiological Assessment	1 month	0
	C-2	Withdrawal Operation	1 month	0
	C-3	Final controls	2 months	0

Table 1: Future dismantling operations, their duration and associated dose rate

Observing the Table 1 we can see that the most critical tasks, concerning dose rate, are the removal of the graphite and concrete. Dose rate coming from graphite removal is slightly higher due to the operations that require contact and operation in a limited space. Workers will receive part of the dose coming from concrete since graphite is the first to be removed and placed in the central part of the concrete block which is the most activated.

The collective dose rate for the concrete operations includes the cutting, handling and storage of the concrete blocks. **In 2006, the dose for these operations was roughly estimated to 0.71 mSV that was decomposed as:**

- ~0.53mSV for the "dismantling of the concrete" including the cutting operation,
- ~0.18mSV for the handling and storage of waste blocks.

In the general studies of Amec Spie there are no details on the approach applied in order to obtain the final values for the collective dose (reported in Table 1).Their calculations were made with Microshield and taking into account the operations in contact with the reactor block, but there are no details about the methodology with which they proceed neither time ratios used. The maximum value for the activation that was taken into account for concrete was 3815 Bq/g. According to our estimation this value corresponds to the peak activation of graphite and it is significantly smaller compared to concrete activation taken as an input value for calculations performed (14 500 Bq/g). *Nevertheless, we will see that my final estimations give results close to those find by Amec Spie.*

The main aim of this study will be to assess more accurately the collective dose rate and to compare the computed results to this previous estimation.

According to STMI planning, total duration of project is expected to be 23 months.

5 BUILDING DISMANTLING METHODOLOGY

5.1 General Approach and Legislation [Ref 9]

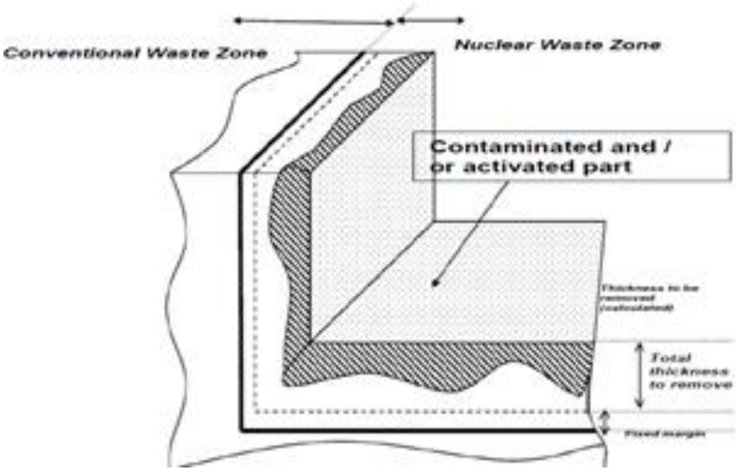
The regulatory basis for site clearance comes, at least in part, from two acts, a decree, and a policy. Specifically, they are:

- Act No. 2006-686 of 13 June 2006 on the transparency and security in the nuclear field—generally, this Act empowers the Nuclear Safety Authority [Autorité de Sûreté Nucléaire (ASN)] to regulate the operations of the nuclear industry;[Ref 10]
- Act No. 2006-739 of 28 June 2006 program on the sustainable management of materials and radioactive waste—this Act provides regulations for the management of radioactive wastes. Its definitions include:
  - ✓ A radioactive substance shall include any substance containing natural or artificial radionuclides, the activity or concentration of which warrants a radiation-protection control; and
  - ✓ Radioactive waste shall include any radioactive substance for which no further use is prescribed or considered[Ref 11]
- Decree No. 2007-1557 of November 2, 2007 relating to nuclear installations and control under the Nuclear Safety of transport of radioactive substances—this Decree addresses the timing and processes of decommissioning, among other things [Ref 12];
- ASN Policy for dismantling and decommissioning of nuclear facilities in France - April 2009—this Policy refines the requirements on timing and processes for decommissioning, among other things [Ref 13].

In general, the ASN guides appear to allow great flexibility of the approaches and processes that nuclear installations are required to address in a comprehensive list of required topics. Unlike some other national approaches, the French approach to clearance is woven into the entire operation and final-status, including financial, decommissioning and dismantlement, and handling of waste. The global approach is based on the concept of zoning for the waste—the separation of conventional materials from materials with associated radioactivity from the installation's operation. As a consequence, each installation may be different in the details on how to arrive at the final-status, but, at the same time, each is required to address the same topics.

5.2 ASN guide n°14- Acceptable Methods of Complete clean-up of Nuclear Facilities

The basic process for clearance of nuclear facilities in France is dependent upon the identification of waste zones. In contrast to the approach of several countries, this guide is focused on the radioactive waste and the non-radioactive waste. Radioactivity on facility surface and in depth of materials may be viewed as a waste to be removed. The parts of facility that fit this potential to contain radioactivity associated with the operation of the facility constitute the zone that serves as the first line of defense.



[Ref 14]

Figure 6: Illustration of the zone of the first and the second lines of defense. The wide line separates the zone of conventional waste from the zone of nuclear waste. The total thickness to be removed consists of the calculated thickness to be removed plus, the flat) rate, precautionary margin

This guide emphasizes that a comprehensive overview must be carried out to define the zone. This definition is not based on simple ad hoc measurements which are only a way of verifying zoning. Within the first line of defense, there can be four categories of potential and physical characteristics of radioactivity and their respective treatments (See Table2).

Category	Radioactivity	Treatment
Category 0	no surface radioactivity or activation	no treatment
Category 1	demonstrated or suspected radioactive contamination of radioactive dust or aerosol	treatment of a very thin surface
Category 2	areas with proven or suspected radioactive liquid contamination	treatment remove the thickness in the defined area with suspected or proven liquid radioactive contamination
Category 3	surface activated or contaminated with penetrating radioactivity	treatment case-by case

Table 2: Categories of potential physical characteristics of radioactivity and their respective treatments

5.3 Lines of defense for waste zoning

Waste zoning was preliminary established for reactor block structures by calculating the neutron activation of the components of the block. From the calculated neutron flux and estimation of induced activity in structures (mass spectrum and activity), the concrete block and various removable structures were categorized into nuclear or conventional zone.

In accordance with the recommendations of the Nuclear Safety Authority on full remediation methodologies acceptable, the approach adopted is based on the use of independent lines of successive defense

- The first line of defense is based on a thorough reflection on the state of the installation, and shall in particular seek to understand and to quantify the physical phenomena involved,
- The second line of defense is confirmation of the conventional character of reclaimed surfaces (with appropriate measures),
- The third and last line of defense is a radiological control of conventional waste.

In our case, the first line of defense is the physical modeling of the phenomenon of neutron activation of the structures, and the determination of the specific activities arising therefrom. Conservative assumptions (on the isotopic composition of materials, assumptions about the mixture of materials in the mesh calculation ...) are routinely taken in this modeling.

From this physical representation in three dimensions, as it is not possible to define intrinsically a limit beyond which the activation phenomenon disappears due to neutrons. A value acceptable for residual activity was defined and modeled in accordance with the recommendations of the nuclear Safety Authority. The optimization of this value reflects International recommendations of the International Atomic Energy Agency (IAEA) and the methodology defined by the CEA.

An additional margin of precaution, with a fixed thickness, is used to define the boundary between nuclear waste zones and zone conventional waste.

A maximum uncertainty for the residual activity must be assigned. This is due to the measuring devices used in the monitoring program (second line of defense). These detection limits are therefore consistent with the decision criteria are acceptable to verify the conventional nature of the concrete structures that are being dismantled (or cleaned).

The values used to categorize structures in nuclear or conventional zone are:

- 11 Bq/g for the conventional concrete (including contribution of tritium that is around 10 Bq/g),
- 2.2 Bq/g for the heavy concrete (including tritium around - 1.4 Bq/g)
  Surface contamination:
    - Less than 0.04 Bq/g for α,
    - Less than 0.4 Bq/g for β and γ.

Residual activity is:

- Less than 100 µSv / year

This methodology allows:

- Defining in which industry waste will be evacuated (nuclear or conventional)
- Defining in the reactor concrete block volume of the nuclear and conventional waste and their boundary

6 MASSIVE BLOCK INVENTORY

6.1 Introduction

Knowledge of both the structures to be dismantled and their environment is also a prerequisite for optimizing dismantling operations and for dose rate calculations. But in practice, the dismantling operations of nuclear installation are often difficult due to the lack of knowledge about the position, the identification and the radiological characteristics of the contamination. However, this knowledge appears essential to define the working conditions.

6.1 Brief description of activation computation: scheme and main results

Here is briefly presented the activation computation done by CEA in 2008[Ref 15].

The calculation of activation in the reactor materials is performed in two separate steps (in yellow), as illustrated in Figure 7. In the first step, spatial neutron flux distributions are determined in the whole reactor for a nominal power of 100 kW.

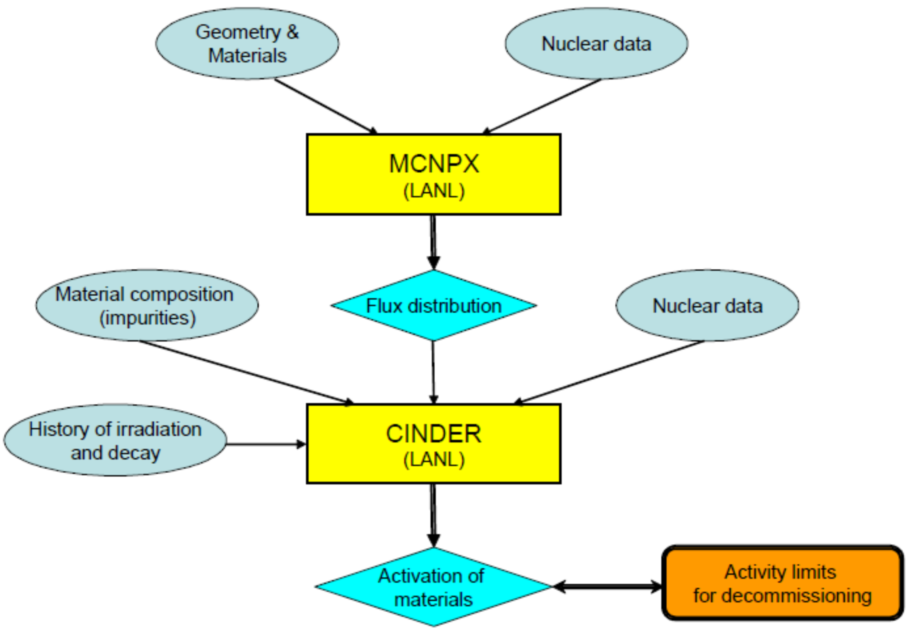


Figure 7: Scheme for neutron transport and activation calculations

In the second step the "static" neutron flux is combined with the history of irradiation and decay in order to obtain the activation of materials in the whole reactor. Neutronics calculation are performed with the Monte-Carlo high-energy transport code MCNP [Ref 5] and using ENDF-B/VI basic nuclear data library. Variance reduction techniques are used to improve the accuracy of neutronics flux estimates in the shielding structures. As shown in Figure 8, the reactor is accurately described using a complete three-dimensional geometry model. The continuous spatial neutron flux distributions resulting from a MCNP calculation are then condensed in a 63 energy-group structure to be used by the CINDER '90 activation code.

The activation of the materials is computed with the deterministic code CINDER '90[Ref 6]. CINDER '90 uses its own nuclear data library originating from different sources, mainly from ENDF, JEF and JENDL but also from theoretical models. Essential inputs for the code are the exact isotopic composition of the reactor materials and the history of irradiation and decay. For the latter, the annual averaged values of the neutron flux are used in the simulations. As a result of the activation calculation, the total activity and its isotopic composition are determined at appropriate time steps after the final shut-down of the reactor.

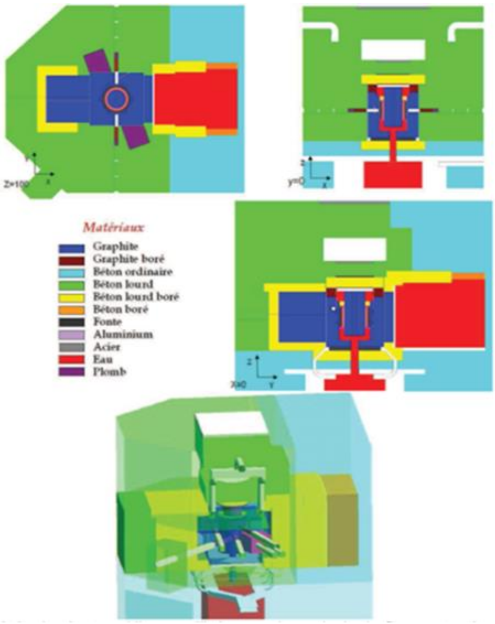


Figure 8: Geometry of the reactor considered for calculation on neutron flux with Code MCNPX

For the calculation fuel is considered new and the control rods fully inserted. The maximum flux in the reactor was $2.95 \times 10^{12} n/cm^2 s^{-1}$, and it is found in the center of inner graphite reflector.

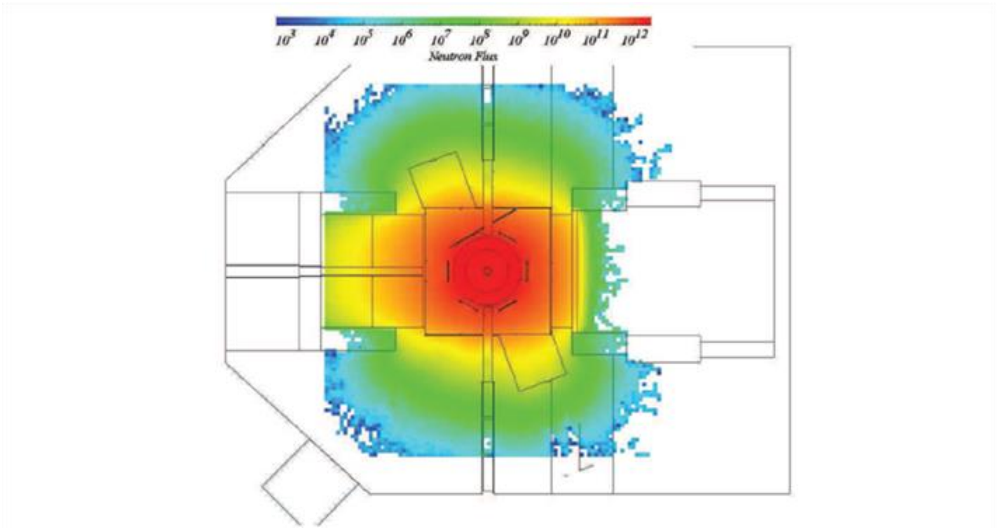


Figure 9: Neutron flux

Emerging from the reactor core, neutrons are gradually thermalized promoting activation and neutron capture in graphite. Once escaped from graphite, neutrons are strongly absorbed by boron in heavy concrete. The evaluation of the total flux at 100 kW (nominal power of the reactor) through the whole reactor block is shown in the Figure 9 for a height of 100 cm above the ground level. The drop of the neutron flux drop in the north and south direction can be observed due to the heavy concrete containing boron which is a strong neutron absorber. This drop is less sharp in the west east direction due to absence of boron or other neutron-absorbing elements (graphite column).

This strong spatial variation has an impact on the activation profiles and must be taken into account in the calculation of the activation.

Neutron flux decreases gradually with distance from the core, which contributes to the gradual decrease in the level of activation of the materials with the distance from the core.

The following figures from 10 to 12 are showing the activation profiles corresponding to the different axes X Y Z.

- Those figures could neglect potential hot spots that might be found in the reactor block since it contains pipes, experiment channels, control rode guide channels. All mentioned elements have been neglected in this study for simplification reasons.
- In addition, only the activity of the main materials and structures are shown in this figure, which means that, for example, steel plates or all of embedded iron (most active material) in a solid block are not necessarily reflected in following figures

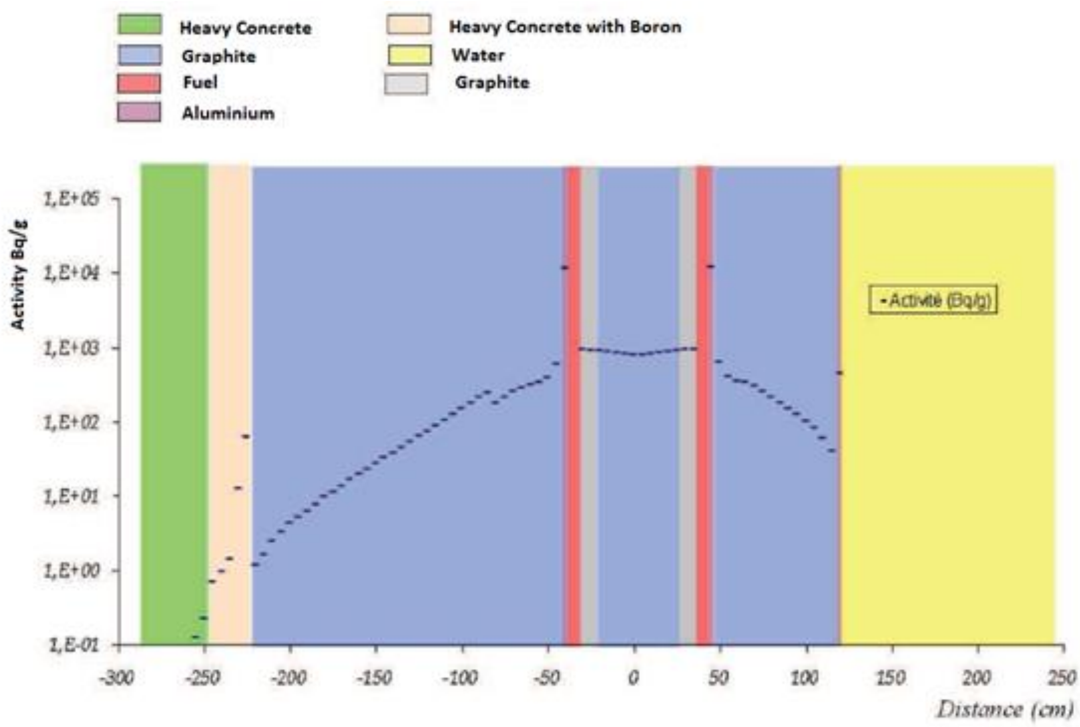


Figure 10: Activation profile of the main material along direction West-East

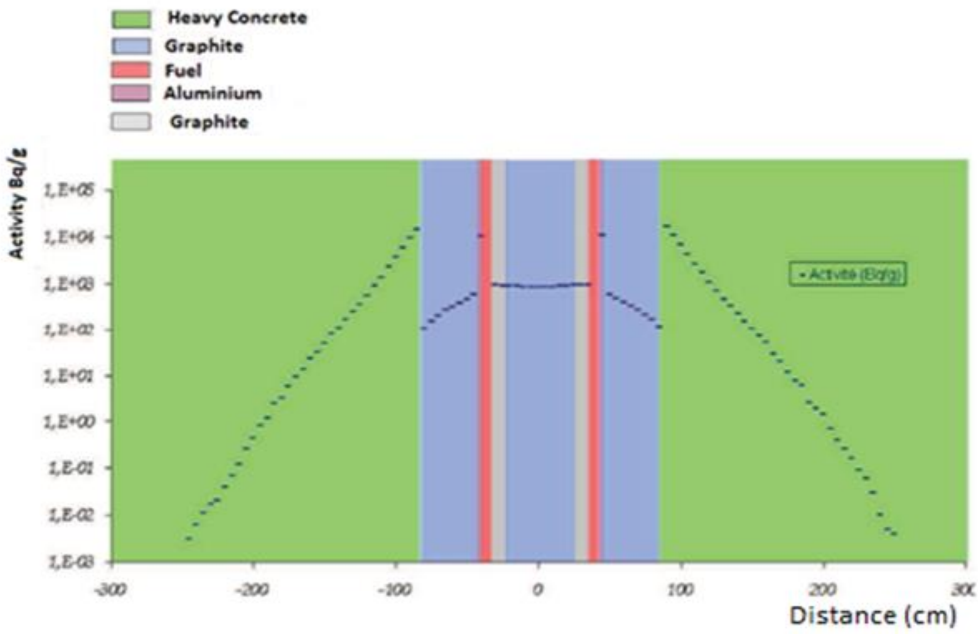


Figure 11: Activation profile of the main material along direction North-South

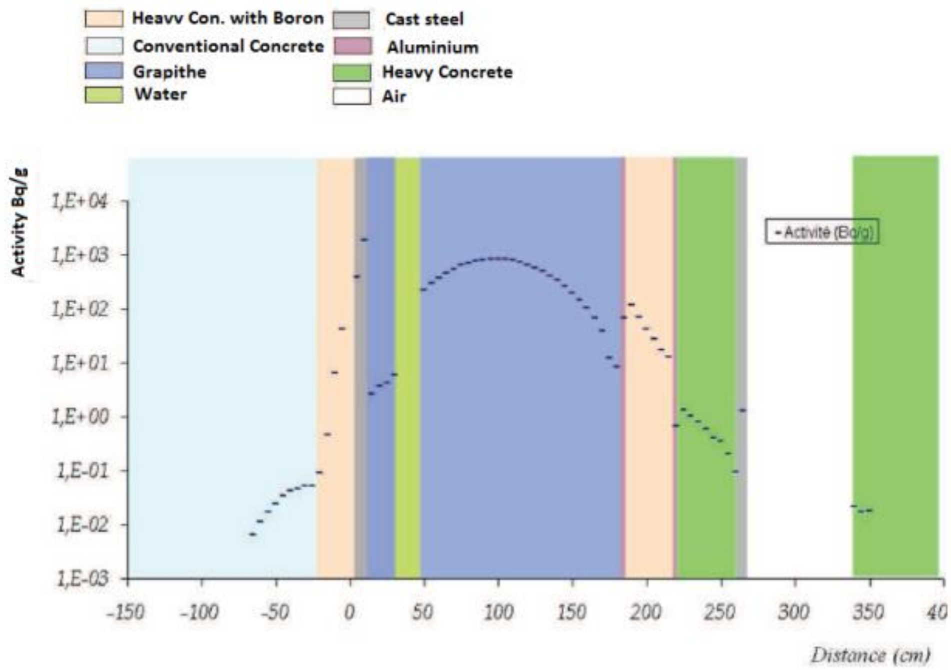


Figure 12: Activation profile of the main material along the height of reactor

6.2 Concrete composition

The long-term activation of structures in a nuclear reactor is mainly due to the interactions between thermal neutrons and impurities in the reactor materials. For this reason, the knowledge of the isotopic composition of all materials present in the system is crucial for activation estimates. Table 3 gives the concrete composition in ppm (particles per million). The potential reactions that could include activation of material are listed in Table 4. Both tab les come from the study performed by CEA [Ref 15], which was conducted to define the waste zoning and to make a preliminary estimation for the waste management of ULYSSE reactor.

It is important to mention that activation of materials also depends on parameters related to the source, fluence, energy spectrum, cross section and time of irradiation time.

Chemical composition of different concrete in ULYSSE reactor				
	Conventional	Heavy	Borated	Heavy-Borated
H	3500	1018	3500	1018
Li	20	12	20	12
Be	0,95	0,95	0,95	0,95
N				
B	16	5,85	20016	20005,85
F		47100		47100
C	37170	16948	37170	16948
O	528800	280308	514102	260138,8
Na	920	445	920	445
Mg	2775	724	2775	724
Al	14555	4340	14555	4340
Si	217600	65395	211552	62089045
P	262	87	262	87
S	3400	84865	3400	84865
Cl	100	30	100	30
Ar	3,06	3,06	3,06	3,06
K	5150	3160	5150	3160
Ca	174000	99800	174000	99800
Sc	3,3	2,44	3,3	2,44
Ti	960	180	960	180
V	39	39	39	39
Cr	39,2	6,62	39,2	6,62
Mn	252	1007	252	1007
Fe	9640	1850	9640	1850
Co	6,82	1,13	6,82	1,13
Ni	26	10	26	10
Cu	30,4	11,5	30,4	11,5
Zn	77	84	77	84
Ga	3,67	3,67	3,67	3,67
Ge	1,24	1,24	1,24	1,24
As	9	5,26	9	5,26
Se	1,05	0,07	1,05	0,07
Br	2,93	0,95	2,93	0,95
Rb	25,9	7,37	25,9	7,37
Sr	328	8730	328	8730
Y	9,45	9,45	9,45	9,45
Zr	54,7	54,7	54,7	54,7
Nb	2,97	2,97	2,97	2,97
Mo	1,45	1,45	1,45	1,45
Ru	2,41	2,41	2,41	2,41
Chemical composition of different concrete in ULYSSE reactor				
	Conventional	Heavy	Borated	Heavy-Borated
Rh	65,7	65,7	65,7	65,7
Pd	16,5	16,5	16,5	16,5
Ag	0,22	7,88	0,22	7,88
Cd	0,6	0,6	0,6	0,6
In	0,03	0,03	0,03	0,03
Sn	1,3	1,3	1,3	1,3
Sb	9,5	98,4	9,5	98,4
Te	1,2	1,2	1,2	1,2
I	4	4	4	4
Cs	2,07	1,05	2,07	1,05
Ba	132	297500	132	297500
La	9,3	9,42	9,3	9,42
Ce	16,2	16,4	16,2	16,4
Pr	2,16	2,16	2,16	2,16
Nd	8,45	11,1	8,45	11,1
Sm	1,88	2,38	1,88	2,38
Eu	0,36	1,8	0,36	1,8
Gd	1,72	1,72	1,72	1,72
Tb	0,26	0,61	0,26	0,61
Dy	1,38	1,38	1,38	1,38
Ho	0,3	0,3	0,3	0,3
Er	0,8	0,8	0,8	0,8
Tm	0,12	0,12	0,12	0,12
Yb	0,74	1,62	0,74	1,62
Lu	0,1	0,19	0,1	0,19
Hf	1,1	1,1	1,1	1,1
Ta	0,29	0,15	0,29	0,15
W	2,35	0,78	2,35	0,78
Re		0,21		0,21
Os	0,33	0,33	0,33	0,33
Ir				
Pt	2,15	2,15	2,15	2,15
Au				
Hg	0,12	0,12	0,12	0,12
Pb	44,4	44,4	44,4	44,4
Bi	0,2	0,2	0,2	0,2
Th	2,4	2,4	2,4	2,4
U	1,14	1,14	1,14	1,14

Table 3: Concrete composition in ppm [Ref 15]

Isotope	Period	Potential reaction	Potential Origin
^{3}**H**	12.3y	$Li^{6}(n, \alpha)$	concrete
^{14}**C**	5 700y	$C^{13}(n,\gamma)C^{14}$, $N^{14}(n,p)C^{14}$	graphite
^{41}**Ca**	$1^{E}5$ y	$Ca^{40}(n,\gamma)Ca^{41}$	concrete
^{54}**Mn**	312 d	$Fe^{54}(n, p)$	iron, steel, stainless steel ,concrete, copper ,aluminum
^{55}**Fe**	2.73 y	$Fe^{54}(n,\gamma)$, $Ni^{58}(n, \alpha)$	iron, steel, stainless steel ,concrete, copper ,aluminum
^{60}**Co**	5.27 y	$Cu^{63}(n, \alpha)$,$Co^{59}(n,\gamma)$,$Ni^{60}(n,\gamma)$	copper, concrete, stainless steel
^{63}**Ni**	100 y	$Ni^{62}(n,\gamma)$	stainless steel
^{87}**Rb**	5E10 y		concrete with boron
^{93}**Nb**	16.1 y	$Nb^{93}(n,n)Nb^{93m}$	graphite
^{94}**Nb**	20 000 y	$Nb^{93}(n,\gamma)Nb^{94}$	graphite
^{108}**Ag**	2.4 min	$Ag^{107}(n,\gamma)$ Ag^{108} Ag^{110m}	lead
^{110m}**Ag**	254 d	$Ag^{109}(n,\gamma)$ Ag^{110} Ag^{110m}	lead
^{133}**Ba**	10.5 y	$Ba^{132}(n,\gamma)$	heavy concrete
^{134}**Cs**	2.06 y	$Cs^{133}(n,\gamma)$	concrete
^{151}**Sm**	93 y	$Sm^{150}(n,\gamma)$	conventional concrete and with boron
^{152}**Eu**	13.53 y	$Eu^{151}(n,\gamma)$	concrete, graphite
^{154}**Eu**	8.59 y	$Eu^{153}(n,\gamma)$	concrete

Table 4: Isotopes responsible for potential activation of material of reactor [Ref 16]

6.3 Concrete physical inventory

Physical inventory assessment was the starting point of my work experience placement. The inventory assessment and the identification of initial state of the facilities is a very important step for dismantling. Estimation of the weight will affect the way of decommissioning, waste production and therefore the cost the project. Proper radiological assessment will give the optimum number of workers for performing different tasks.

A comprehensive and accurate identification of elements to dismantle has been made in all areas affected by the decommissioning by CEA and STMI. Using their documentation composed of drawings, technical notes, safety report and historical study, it was possible to assess the volume and mass of the concrete block (See tables 5 and 6).

Density g/cm3	
Heavy Concrete	3,5
Conventional Concrete	2,3

Table 5: Density of different concrete

	Reactor block	
	Volume (m3)	Mase (t)
Conventional (CNV) :	46,625	163,1875
Very Low Activated (TFA) :	21,514	75,299
Low Activated (FA) :	14,282	49,987
Total	82,419	288,4735

Table 6: Concrete mass and volume estimation

Based on activation computation performed by CEA previously presented, it was possible to classify waste on conventional, very low level waste and low level waste using the zoning principle.

6.4 Radiological spectrum

The computation of the activation also gives the radiological spectrum. As it was mentioned before, Ulysses reactor is constructed from four types of concrete: conventional concrete, heavy concrete, heavy concrete with boron and finally conventional concrete with boron. Heavy concrete has a density significantly higher when compared to conventional concrete. This was taken into account in this study due to the limited capacity of the crane: 6 tones.

The massive part of the reactor block will be cut in the dismantling process is made of heavy concrete or heavy concrete with boron. This is also why we take into account their isotopic composition in computation of dose rate for worke

rs.

In the following table we can see that by the end of 2008 **the two main isotopes contributors were ^{3}H and ^{152}Eu.**

	Heavy concrete Heavy concrete with boron	Conventional concrete Concrete with Boron	Mix of all the different concrete types
	End of 2008	End of 2008	End of 2008
^{3}H	66.83%	95.46%	80.43%
^{152}Eu	17.8%	3.1%	10.51%
^{133}Ba	14.98%	-	7.56%
^{154}Eu	0.79%	-	0.47%
^{41}Ca	0.46%	0.69%	0.58%
^{60}Co	0.14%	0.75%	0.45%

Table 7: Radiological spectrum of the concrete at the end of 2008[Ref 16]

Since tritium is a low energy beta emitter, it is not dangerous externally (its beta particles are unable to penetrate the skin), but it is a radiation hazard when it is inhaled, ingested via food or water, or absorbed through the skin. The most important isotopes dose rate computation are 60Co, 152Eu, 154Eu and 133Ba which build up over time from (n,γ) reactions under neutron exposure from trace amounts of stable Europium, Cobalt, and Cesium that are normally present in concrete. For simplification reasons neglect the contribution of that 41Ca due to its long half-life and very low probability of emitting gammas.

The peak value of the activation in concrete is 14 500 Bq/g. As for the contribution of *gamma emitters is concerned, the maximum activity is reported in Table 8 and will be used hereafter in our dose rate calculation*.

Maximum activity per cm^3 of gamma emitters in concrete (2008)				
	^{60}Co	^{152}Eu	^{154}Eu	^{133}Ba
Percentage of the total activity	0.14	17.8	0.79	14.98
Activity Bq/cm^3	71	9043	401	7620

Table 8: Contribution of gamma emitters to the total activity of concrete

In the table below are listed gammas with their energies and their yield taken form LARA 2000 library [Ref 32] [Ref 33]. It is important to mention that, for some isotopes, like Europium (Eu), the values are compressed and that is the reason why we don't see some gammas with very low yield on the list. In Janis [Ref 23], Nuclear Data Information System, the complete list of all gammas can be found.

Spectre gamma			
Radionucléide	*Isotope*	*E(keV)*	*I (%)*
Co	60	347,14 keV	*0,008 %*
Co	60	826,1 keV	*0,008 %*
Co	60	1173,23 keV	*99,85 %*
Co	60	1332,49 keV	*99,983 %*
Co	60	2158,57 keV	*0,001 %*
Eu	152	63,39 keV	*103,081 %*
Eu	152	328,07 keV	*38,348 %*
Eu	**152**	**455,27 keV**	**3,924 %**
Eu	152	649,53 keV	*3,278 %*
Eu	152	800,74 keV	*17,985 %*
Eu	152	963,44 keV	*16,193 %*
Eu	152	1100,18 keV	*25,862 %*
Eu	152	1259,68 keV	*3,397 %*
Eu	152	1409,1 keV	*21,322 %*
Eu	152	1531,43 keV	*0,292 %*
Eu	152	1647,02 keV	*0,006 %*
Eu	152	1769,09 keV	*0,009 %*
Eu	154	93,13 keV	*66,087 %*
Eu	154	249,69 keV	*7,12 %*
Eu	154	452,39 keV	*1,259 %*
Eu	154	614,43 keV	*8,784 %*
Eu	154	731,26 keV	*25,383 %*
Eu	154	874,65 keV	*14,615 %*
Eu	154	1001,9 keV	*28,842 %*
Eu	154	1271,13 keV	*36,75 %*
Eu	154	1490,7 keV	*0,832 %*
Eu	154	1596,66 keV	*1,823 %*
Ba	133	43,21 keV	*160,93 %*
Ba	133	340,82 keV	*97,514 %*

Table 9: Gammas energies and yield for the main isotopic contributors to the dose rate

6.5 Time impact on the Radiological spectrum

Due to the fact that the activation assessments were performed in 2008 it appears interesting to estimate the remaining radioactivity in 2014 and to see how much time has contributed to the decrease in radioactivity. Considering the half-life of isotopes of gamma emitters and their activity in 2008, Figure 13 illustrates the impact of the decay and the remaining activity in 2014 for gamma emitting isotopes. Value used for computing is maximum activity found in Bq /g. **The remaining activity in 2014 represents 70% of the total activity in 2008. The data used for the dose rate calculation will be the one from the year 2008.**

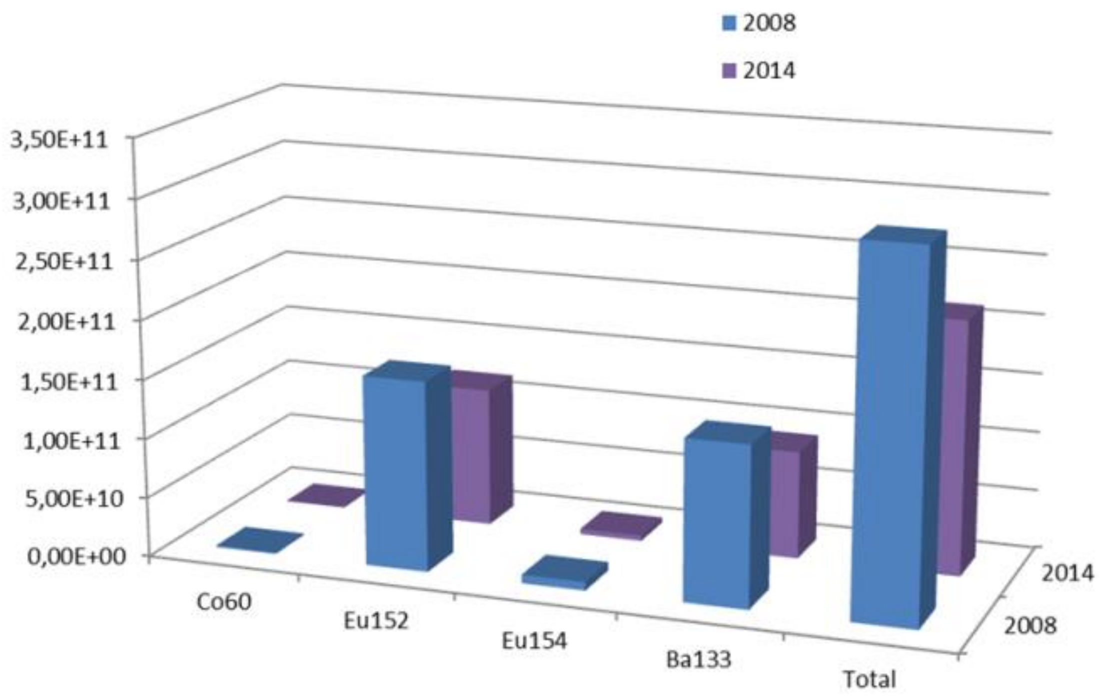


Figure 13: Activity of gamma emitting isotopes calculated at the end of 2008 and 2014

7 RADIATION ENVIRONMENT ASSESSMENT AND SOFTWARE PRESENTATION

7.1 Introduction [Ref 22]

Radiation environment assessment is mandatory in design, operation, maintenance and decommissioning of nuclear facility for proving compliance with ALARA principles. Nowadays mathematical modeling methods and corresponding software are widely used for this purpose. All the software used for radiation fields modeling falls into two main groups.

To the first belongs **software based on Monte Carlo methods**, such as MCNP [Ref 35], TRIPOLI [Ref 27], Geant [Ref 34], etc. This leads to perfect accuracy of the calculated values even for complex models accounting of radiation transport effects . At the same time Monte Carlo calculations require considerable amount of computing time. Computation burden increases rapidly for complex geometries, multiple sources and thick shielding.

The other group of **software use point-kernel method for doze calculations**. In this study such codes will be used –Microshield [Ref 20], Mercure [Ref 17], Mercurad [Ref 18], DosiMex[Ref 37], Narmer[Ref 21]. Due to macroscopic approach to radiation transport this methods lacks consistency, but it is significantly less computationally intensive than Monte Carlo method. The main problem point kernel method encounters is taking into account scattered radiations which is usually implemented through semi-empirical approximation. Additional "build-up" factor must be introduced as a multiplier to the attenuated doze. Determination of the appropriate buildup factor can be rather complex as it depends upon the energy, the thickness and type of material. Uncertainties in determining build-up factor essentially limit the accuracy of point-kernel method.

7.2 Point-source kernel method

Point-kernel method is a macroscopic approach used for gamma radiation exposure rate calculations.

For suitable geometry, the point source approximation is often employed for isotropic emission from a source, which emits particles per unit time.

Within this approach gamma radiation propagation is assumed to be beam-like. Effects of radiation interaction in matter are described using macroscopic linear attenuation factors. Consistent scattered radiation accounting cannot be achieved within this macroscopic approach. So, common practice is to use semi-empirical relations, such as Berger formula, Taylor formula, etc.

According to the main idea of point-kernel method radiation source volume is cut up into elementary cells (point kernels). Each point kernel gives contribution to the doze rate at the detecting point for radiation energy *E* and it is computed using following equation:

$$A(\vec{r}, \vec{r}', E) = C(E)B(t, E)\frac{\exp(-t(E))}{4\pi(\vec{r} - \vec{r}')^2},$$

Where

- *C* (*E*) is gamma flux density to doze rate conversion factor,
- *B* (*E*) build-up factor,
- t (E) path length along line $\vec{r} - \vec{r}'$ measured in mean free paths (mfp)

$$t = \sum_{i=1}^{n} \mu_i(E) X_i,$$

Where

- i -is index of the space region,
- n - number of regions,
- $\mu_\iota(E)$ -linear attenuation factor for i-th region and
- *Xi* - length of the section of $\vec{r} - \vec{r}'$ line in i-th space region.

Geometry used for point-kernel calculations is shown on the Fig.14.

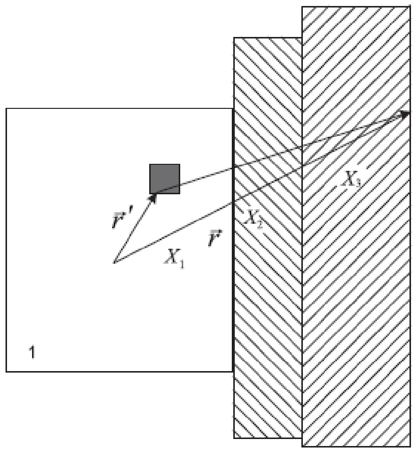


Figure 14: Geometry of point kernel method

The linear attenuation coefficient (μ) for photons is analog to the macroscopic cross section (Σ) for neutrons, and hence, the μ for a mixture of components (1 and 2) in equal proportion is: $\mu_{mix} = \mu_1 + \mu_2$.

The mass attenuation coefficient is μ/ρ where ρ is the material density. For a mixture based on weight fractions, ωi, it results $(\mu / \rho)_{mix} = \Sigma \omega i \, (\mu / \rho)_i$ (2)

The mean free path, which is the mean distance that a photon crosses between interactions, is mfp = 1/μ. Factors μ_ι describe radiation attenuation along the given path. From the microscopic theory it follows that attenuation coefficients μ_ι depend on matter composition and on photon energy.

The mass attenuation coefficients of ordinary and barite concrete are presented in the Figures 15 and 16[Ref **44**].

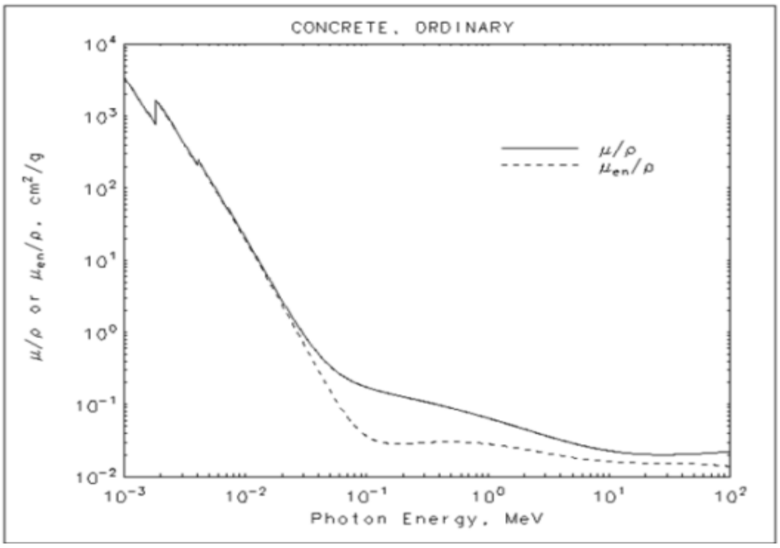


Figure 15: Mass attenuation (µ/ρ) factor for concrete

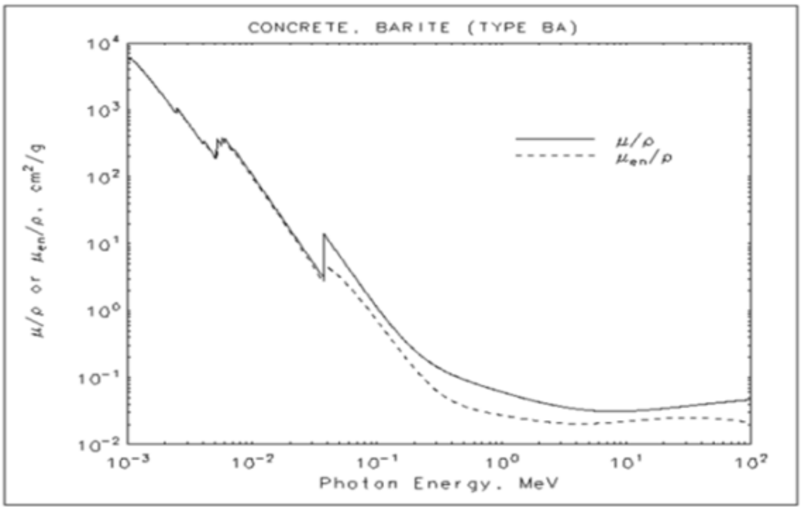


Figure 16: Mass attenuation (µ/ρ) factor for concrete barite

The dose buildup factor is a dimensionless quantity that represents the ratio of total dose at the dose point to primary photon dose at the same point. Noted as B, buildup factor account for

- ✓ photoelectric effect,
- ✓ pair production,
- ✓ scattering (elastic ,inelastic),
- ✓ secondary gamma emission on absorption of the original gamma ray (Bremsstrahlung),
- ✓ forward scattering by the shield.

It depends on:

- ✓ material,
- ✓ γ-ray energy,
- ✓ geometry.

B is generally determined from tables, or empirical formulas. There are a number of algebraic expressions that have been used to represent build up factor. Among the most popular is an expression referred to as Taylor's form of the buildup factor,that we applied in DosiMex and which is expressed as:

$$B = Ae^{-\alpha_1 \mu x} + (1 - A)e^{-\alpha_2 \mu x}$$

where α_1, α_2, A are parameters that depend on material composition and on radiation energy E.

Another analytical forms used for the buildup factor can be applied as Berger's form:

$$\mathrm{B}(\mu x, \mathrm{E}) = 1 + \mathrm{a}\ \mu x\ \mathrm{e}^{\ \mathrm{b}\mu x}$$

,where a and b are constants for a given energy and shield material,

Point-source kernels are assumed to be independent. So total doze rate at detection point is obtained by the integration of :

$$A(\vec{r}) = \sum_{k=1}^{N} \iiint_V d\vec{r}' A(\vec{r}, \vec{r}', E_k)$$

over the source volume and summation over the energies Ek of radiation spectrum. For simple geometries integration can be carried out analytically.

Spatial integration presents a problem for detection points located too close to source. This is a common drawback of point-source kernel methods due to integral divergence at such points. One of the possible solutions is to exclude from integration kernel points that fall within some predefined region near the detection point. This procedure provides stabilization of integration procedure but leads to some loss of accuracy. At the same time for most of the cases concerning radiation protection this problem does not arise at all. The matter is that detection points are usually separated from the source by shielding layers and no divergence takes place.

7.2.1 Mercure 6.3

The MERCURE calculation code (version 6.3) simulates the photons transport from 1 keV to 10 MeV in three dimensional geometries between volume sources and calculation points. It is based on the integration of attenuation point kernels in straight line with accumulation factors.

The total microscopic cross sections of the library operated by Mercury6.3 are taken from the library JEF2.2. Dose conversion data reflects the standard from ICRP 74[Ref 39]and [Ref 38].

Build-up factors are computed by a method developed at CEA / SERMA[Ref 45]. Build-up factors are calculated from transport calculations performed using the discrete ordinates method (SN), with general anisotropic scattering. The energy groups, the angular, and space mesh meet the following features, respectively: 218 energy groups between 1 keV and 10 MeV and for transport calculation Legendre expansion of the scattering cross section and angular quadrature are assumed.

7.2.2 Mercurad

MERCURAD is a 3D simulation software for dose-rate calculation, enabling 3D dynamic simulation of structures (sources, shielding, collimators, etc.). MERCURAD is based on the powerful 6.0 version of the MERCURE code developed and validated. It allows multi-source, multivolume and multi-point calculations, much faster than a Monte-Carlo based method like MCNP for a given accuracy. The new MERCURAD software makes extensive use of a powerful graphical user interface, thus allowing very complex items to be easily defined and presented. Computing method for gamma-ray transport is straight line attenuation mode, with integration of point kernels in 3D geometry, while for build-up algorithms it uses new CEA method with multi-layer formula developed using neural networks (up to 50 mfp).Numerical integration of the sources is done by the Monte Carlo method.

7.2.3 Microshield

Microshield is a comprehensive photon/gamma ray shielding and dose assessment program. It is widely used for designing shields, estimating source strength from radiation measurements, minimizing exposure to people, and teaching shielding principles. Microshield is useful to health physicists, waste managers, and design engineers, and radiological engineers, among others. One of the primary advantages of this software is that it only requires a basic knowledge of radiation and shielding principles.

Library data (radionuclides, attenuation, buildup, and dose conversion) reflect standard data from ICRP 38 and 107 as well as ANSI/ANS standards (ANSI/ANS-6.4.3 1991) and RSICC publications.

7.2.4 Narmer

Narmer is a code that simulates the transmission of gammas using method of attenuation in straight line with accumulation factors. The geometry, which is in three dimensions, is the same as for the Monte Carlo transport code Tripoli-4. The method used in NARMER is the point-kernel integration method with build-up factors. It is a macroscopic approach for gamma dose rate calculation, taking into account the interaction with matter.

The Narmer 1.4 code allows the calculation of dose equivalent rates and temperature rise. The scattered flux is taken into account through build-up factors. Build-up factors are computed by a method developed at CEA / SERMA. The build-up factors for mono-layer are evaluated by a SN method and are tabulated in function of energy and relaxation length.

Spatially distributed sources are processed by integrating the kernel point by Monte Carlo method. Cross sections are available in multigroup library (195 energy groups) and point-wise library. The build-up is only available in the multi-group.

7.2.5 DOSIMEX-G 1.4

The software is dedicated to engineers or students who have not necessary an extensive experience in code applications.

DOSIMEX-G 1.4 is a computer tool for dose rate calculation coming from gamma and X radiation with an easy-use interface that allows constructing models using simple functions and geometries. The deterministic method is based on the straight line calculation with attenuation using build-up factors correction based on X COM library (NIST). Code is written in VBA and used in Excel. Different type of sources could be used with this tool: volume sources (sphere, cylinder and cube), surfaces or linear sources. Many options have been implemented; it is possible to take into account a gradient of activity (NAA) in the parallelepiped geometry. DOSIMEX-G uses the fluence-equivalent dose coefficients for calculating standardized operational quantities (ICRU 57 and ICRP 74) and the build-up factor of ANS / ANSI (ANSI/ANS-6.4.3 1991 as Microshield). DOSIMEX-G is user-friendly tool. The results were validated by comparison to a large number of benchmark with Mercurad and Microshield.

7.3 TRIPOLI CODE

TRIPOLI-4 solves the Boltzmann equation for neutrons and photons, with the Monte Carlo method, in 3D geometry. The code uses ENDF format continuous energy cross-sections, from various international evaluations including JEFF-3.1.1, ENDF/B-VII.0, JENDL4 and FENDL2.1. Its official nuclear data library for applications, named CEAV5.1.1, is mainly based on the European evaluation JEFF-3.1.1 and is the only one distributed in this package. TRIPOLI-4 solves fixed source as well as eigenvalue problems. It has advanced variance reduction methods to address deep penetration issues. With its large V&V data base, TRIPOLI-4 is used as a reference code for industrial purposes (fission/fusion), as well as a R&D and teaching tool, for radiation protection and shielding, core physics (without depletion in this package), nuclear criticality-safety and nuclear instrumentation.

The fluence-equivalent dose coefficients for calculating standardized operational quantities comes from ICRP 74 [Ref 39]and [Ref 38] and the build-up factor by a method developed at CEA / SERMA[Ref 45].

Energy ranges:

- ✓ Neutron : 10E-11 to 20 MeV
- ✓ Photon : 1 keV to 100 MeV

TRIPOLI-4 running time is comparable with other state of the art Monte Carlo codes. It is heavily case-dependent.

7.4 Operational quantities

The body related dose quantities (equivalent dose and effective dose) are not directly measurable and, therefore, cannot be used directly in radiation protection monitoring. For that reason operational quantities have always been applied for the assessment of effective dose or mean doses in tissues or organs. Operational quantities are aimed at providing a conservative estimate or upper limit for the value of the protection quantities related to an exposure, or potential exposure of persons under most irradiation conditions. They are often used in practical regulations or guidance instead of the protection quantities. These quantities complement the system of quantities generally applied in radiological protection. As shown in the Figure 17, different types of quantities are used for internal and external exposure situations.

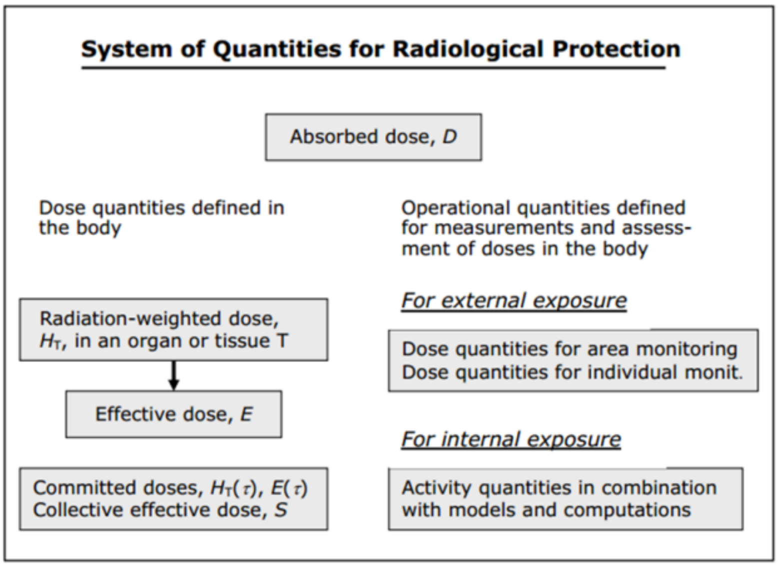


Figure 17: System of Quantities for Radiological Protection

For radiation monitoring in cases of external exposure (area or individual monitoring) operational dose equivalent quantities are defined. Operational quantities are used for monitoring external exposures because

- protection quantities are not directly measurable,
- for area monitoring point quantities are needed, effective dose is not appropriate in area monitoring, because in a non-isotropic radiation field its value depends on the orientation of the human body in that field,
- instruments for radiation monitoring need to be calibrated in terms of a measurable quantity for which calibration standards exist.

Due to the different tasks in radiological protection, including area monitoring for controlling the radiation in work places and for defining controlled or restricted areas and individual monitoring for the control and limitation of individual exposures, different operational dose quantities have been defined. While measurements with an area monitor are mostly performed free in air, personal dosimeters are usually worn at the body. As a consequence, in a given situation, the radiation field "seen" by an area monitor free in air differs from that "seen" by an personal dosimeter worn on a body where the radiation field is strongly influenced by the backscatter and absorption of radiation in the body. The use of different operational dose quantities allows for such phenomena. For the different tasks of monitoring of external exposure the following quantities are defined:

Task	Operational quantities for	
	area monitoring	individual monitoring
Control of effective dose	ambient dose equivalent $H^*(10)$	personal dose equivalent $H_p(10)$
Control of skin dose	directional dose equivalent $H'(0.07, \Omega)$	personal dose equivalent $H_p(0.07)$

Figure 18: Operational quantities

With respect to the application of the operational quantities the ICRU (1993) has stated that H*(10) and Hp(10) are designed for monitoring strongly penetrating radiation, e. g. photons (above about 12 keV) and neutrons, while Hp(0.07) is applied for monitoring weakly penetrating radiation, e. g. α- and β-particles. Furthermore, Hp(0.07) is also used for monitoring the doses to the extremities from all ionizing radiation.

8 DOSE RATE BENCHMARKING: RESULTS AND DISCUSSION

8.1 Introduction

In the previous chapters were dealing with the presentation of the ULYSSE reactor, the activation spectrum and the theoretical background necessary for the dose rate assessment.

The main difficulty for computation of the dose rate comes from the fact that the concrete is not homogenously activated. Thus **activation shows a sharp exponential decrease when going from the core to the external part of the reactor block.** The activation profile in concrete can be seen in Figure 19.

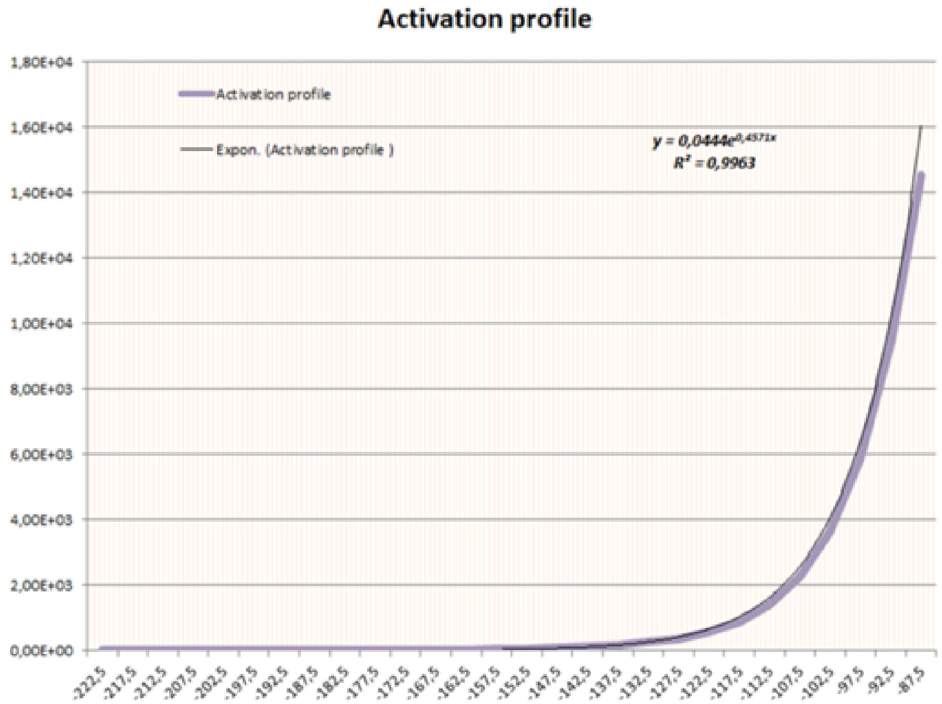


Figure 19: Activation Profile in concrete

This is expected since the inner layer of the concrete has been exposed to the highest neutron flux that was then attenuated gradually.

Most codes for computing the dose rate consider homogeneous sources. Computationally they are simpler than the ones with the possibility to consider gradient activation.

In this chapter dose rate will be calculated considering two different models for the concrete block:

- ✓ Simplified -considering homogeneous source and maximum activity 14 500 Bq/g
- ✓ After implementation of gradient activation in the code

Mercure code is used as a reference code in dose rate calculation.

8.2 Assumptions and common input data

After obtaining the concrete activation profile, considering the geometry of the reactor block and the working conditions during dismantling, some assumptions are taken to simplify the problem.

Here is the list of the assumptions that were taken for the calculation:

- First of all we consider for the calculation only the nuclear part of the concrete block and not the conventional part which is not activated
- The Layer 1 (as defined in Figure 20 to 22) is the most irradiated part of the concrete, classified as Low Level Waste (FA)
- Layer 2 and 3 (as defined in Figure 21 to 22) are classified as Very Low Level Waste (TFA) and Layer 2 is between Layer 1 and 3.
- Each Layer of concrete has side that is less-activated and side that is more-activated
- Activation profiles found in CEA documents [Ref 15] are used to perform the calculation
- In the Chapter 8 calculations are made for the Layer 1, and later on, in Chapter 9 will be presented the results for Layer 2 and 3.
- Since we are interested in the dosed rate received by the worker performing dismantling, we consider simplified model, with just one half of the reactor. We consider that the worker is standing in front of the wall (Figure 23).
- Dose rate is computed at four different points. Two of them in contact with the concrete and the other two at one meter distance from the concrete block.
- For computing the mass and volume of the blocks, density of heavy concrete is taken it to account
- Computations are performed for both, conventional (2,3 g/cm3) and heavy concrete (3,5 g/cm3)
- The data taken into account for activation was from the year 2008
- Only gamma emitters are taken into account
- Some of gamma emitters are neglected because of the low yield of radiation or long half-life (41Ca long half-life, 56Fe 0% yield)
- Graphite that constitutes the external reflector is not taken in to account because in every scenario it would be first removed before starting the cutting process

Below, the figures 20 to 22 define the geometry of the reactor block and the different Layer 1 (FA waste), 2 and 3 (both TFA waste).

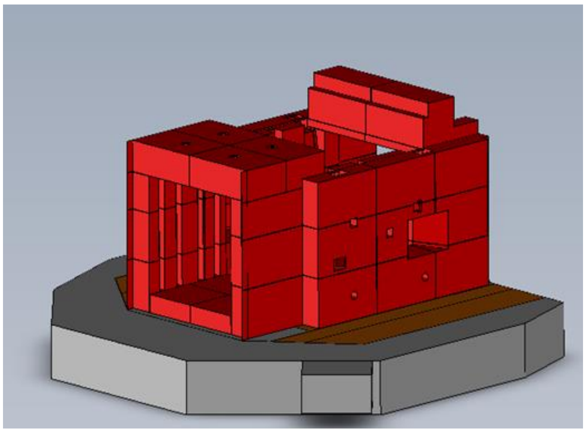


Figure 20: Layer 1

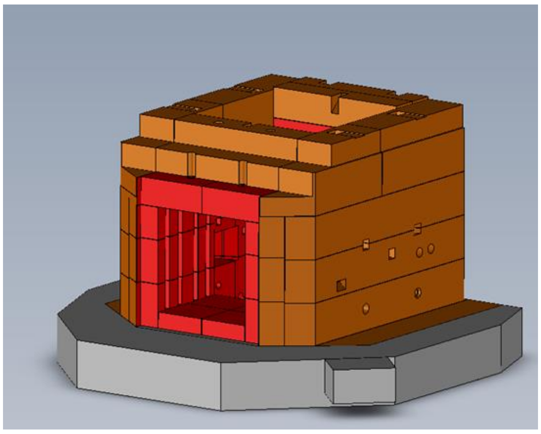


Figure 21: Layer 1 and 2

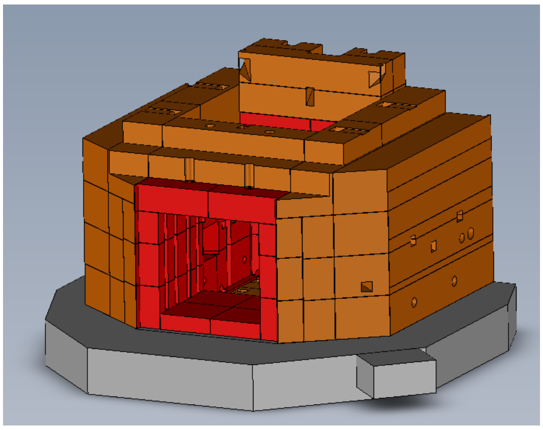


Figure 22: Layer 1, 2 and 3

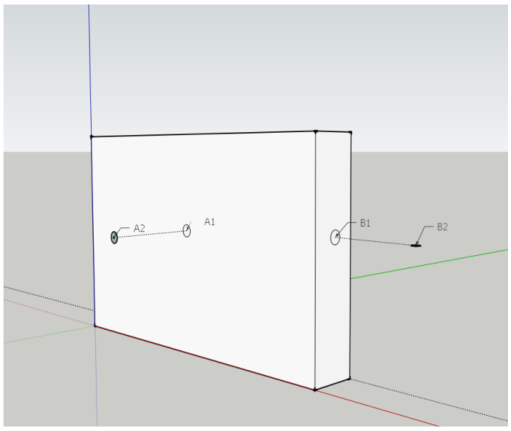


Figure 23: Points of interest for dose rate calculation

8.3 Calculation with homogeneous source

In Table 10 are reported the results computed with Mercure for ambient dose equivalent in case of two different densities. First computation is made with concrete density of 3.5 g/cm^3 that corresponds to heavy concrete, and the second with density 2.3 g/cm^3 for conventional concrete. An homogeneous source is assumed, taking into account the maximum value for each isotope. To remind, the geometry considered is the one corresponding to Layer1. Computation point A1 is at 1 cm distance from the concrete block, while A2 at 1 m away from the front side of the wall. Point B1 is at 1 cm distance from the side of the wall and point B2 at one meter distance from the side (see Figure 23). The dose rate for heavy concrete is taken as a reference (ref) and the last line of Table 10 indicates the ratio between the dose rate for conventional and for heavy concrete.

Dose Rate H*(10) µSv/h					
Density g/cm3	homogeneous source	A1	A2	B1	B2
3,5	Mercure	1207	575	1185	154
		ref	ref	ref	ref
2,3	Mercure	1803	860	1826	193
	Δ	49,38%	49,57%	54,09%	25,32%

Table 10: Dose rate computed with Mercure at different points and with different density

As expected, significantly higher value for dose rate in the case of conventional concrete can be noticed. Moreover, the dose rate and density are roughly inversely proportional quantities. That justifies the use of heavy concrete as a shield in experimental reactor since it reduces significantly the dose rate. **Comparing the values at one centimeter and one meter we can see the difference of around 48%, which reminds at one of the basic principles of radioprotection-distance**. For point sources of gamma radiation, the photon flux is inversely proportional to the squared distance from the source. This leads to a well-known radiation shielding rule that doubling the distance from the source decreases the radiation by a factor of four.

Continuing analysis with homogenous source, other four codes previously presented are used for dose rate calculation. Tables 11 and 12 summarize the results and give the discrepancy taking into account Mercure as the reference code. Calculations are made for heavy and conventional concrete.

		Dose Rate H*(10) µSv/h			
Density (g/cm3)	CODE	A1	A2	B1	B2
3,5	Mercure	1207	575	1185	154
		ref	ref	ref	ref
	MicroShield	1131	541	809	145
	Δ	**-6,34%**	**-5,83%**	**-31,71%**	**-5,73%**
	Narmer	845	420	824	107
	Δ	**-29,99%**	**-26,96%**	**-30,46%**	**-30,52%**
	Mercurad	938	467	892	118
	Δ	**-22,29%**	**-18,78%**	**-24,73%**	**-23,38%**
	DosiMex	1124	559	1092	142
	Δ	**-6,88%**	**-2,78%**	**-7,85%**	**-7,79%**

Table 11: Dose rate benchmark for heavy concrete

		Dose Rate		H*(10) µSv/h	
Density (g/cm3)	CODE	A1	A2	B1	B2
2.3	Mercure	1803	860	1826	193
		ref	ref	ref	ref
	MicroShield	1729	813	1392	208
	∆	**-4,10%**	**-5,49%**	**-23,75%**	**7,90%**
	Mercurad	1238	627	1194	158
	∆	**-31,34%**	**-27,09%**	**-34,61%**	**-18,13%**
	DosiMex	1720	841	1640	210
	∆	**-4,60%**	**-2,21%**	**-10,19%**	**8,81%**

Table 12: Dose rate benchmark for conventional concrete [1]

The following preliminary conclusions can be made:

- Compared to Mercure, Mercurad and Narmer are showing the biggest discrepancy (typically between 20 and 30 %), although Mercurad is based on the 6.0 version of the MERCURE code. For further analysis it would be necessary to perform more detailed investigation and calculation with different libraries. Discrepancies may arise from the consistency of data library.
- Mercure gives higher value than any other code.
- DosiMex gives the closest values to Mercure. The discrepancy is typically lower than 8%
- The maximum value of the discrepancy is lower than 35%, which is satisfactory for our study.
- The activities used for the calculation were those from the year 2008. Since a decrease of 30 % in total activity due to the natural decay occurred until 2014, we expect to have a significant margin in the calculated dose rates (over estimation by about 30%).
- Homogeneous source with maximum radioactivity is considered, while we know that the activation inside of the concrete block is showing a sharp exponential profile. This means that at least for one side of concrete block there is an over estimation in dose rate.

8.4 Calculation with gradient source

Since assumption of homogeneous source is the "worst case scenario" -knowing that in practice there is an exponential activation of the concrete, there was a need for gradient option in the computation. This is expected to give more accurate estimation of the dose.

Table 13 gives the results of ambient dose equivalent obtained with Mercure code for homogeneous and gradient source, Table 14 gives the same type of results obtained with the DosiMex tool.

[1] * The calculations were performed by another engineer in CEA/DEN. We thank Ms Salmon and Mr Jaboulay (CEA) for kindly providing the results.

Dose Rate	H*(10) µSv/h	
Mercure	more activated side	less activated side
Gradient source	926	14
	ref	ref
Homogeneous source	1207	1207
Δ	30,31%	8742,49%

Table 13: Dose rate obtained with Mercure considering gradient and homogeneous sources

With both codes, we can notice an overestimation in the inner-more irradiated part of the block that is around 30%. As far as the outer (less irradiated side of the block) value is concerned, there is a strong overestimation by almost two orders of magnitude (factor of about 90). **This means that considering an homogeneous source represents a strong overestimation of the dose rate. From an operational point of view, this would lead to an increase in the time and the cost of the global decommissioning operation.**

Dose Rate	H*(10) µSv/h	
Dosimex	more activated side	less activated side
Gradient source	905	12
	ref	ref
Homogeneous source	1124	1124
Δ	24,20%	9266,67%

Table 14 : Dose rate obtained with DosiMex considering gradient and homogeneous sources

In Table 15, the comparison between the two options shows that the homogenous source is most penalizing even with DosiMex.

Dose Rate	H*(10) µSv/h	
DosiMex/ Mercure	more activated side	less activated side
Gradient source	926	14
Δ	-2,30%	-12,09%
Homogeneous source	1207	1207
Δ	-6,88%	-6,88%

Table 15: Comparison of calculation made with Mercure and DosiMex

Observing the tables we notice that the dose rate calculated on the more irradiated side after implementation of gradient is relatively close to the value obtained taking as the activity the maximum activity of the concrete.

Finally, in the Table 15, the comparison between Mercure and DosiMex are satisfying-the discrepancy ranges from -2,3% to 12%. Moreover, there is an underestimation of the results computed by DosiMex. We can see that the maximum value for dose rate is computed at the inner side of the concrete block (more irradiated), and it is around 0.9 mSv, depending on the computation method. In the other hand, very low values are noted for the outer part of the concrete bloc(Less irradiated). Sharp exponential drop of the radioactivity along the concrete block results in huge differences between the dose rates computed on each side. For instance, at one centimeter of the outer part of the first layer we have 12µSv/h, while at the inner part and same distance we compute 905µSv/h (with DosiMex). For further analysis and in order to confront our reference deterministic results, Monte-Carlo calculations will be performed in order to estimate the accuracy of the deterministic calculation.

8.5 TRIPOLI 4 Calculation

8.5.1 Results of Monte-Carlo code

In the Table 16 are reported the results obtained with TRIPOLI-4 code. Contribution of each isotope, statistical error and total value of dose rate can be seen in the table. Each calculation was performed with 4 000 batches of 50 000 neutrons with a statistical error (σ%) lower than 1.7%.

H*(10)	TRIPOLI4 (JEFF3, ICPR74)	Point A1 (σ %)		Point A2 (σ %)		Point B1 (σ %)		Point B2 (σ %)	
Béton normal densité 3,5 g/cm3	**TOT**	**1282 µSv/h**	1,45 %	**646 µSv/h**	0,08 %	**1244 µSv/h**	1,48 %	**165 µSv/h**	0,21 %
	^{60}Co	8 µSv/h	1,44 %	4 µSv/h	0,08 %	8 µSv/h	1,39 %	1 µSv/h	0,21 %
	^{152}Eu	1045 µSv/h	1,35 %	522 µSv/h	0,08 %	1008 µSv/h	1,42 %	133 µSv/h	0,19 %
	^{154}Eu	52 µSv/h	1,35 %	26 µSv/h	0,07 %	51 µSv/h	1,26 %	7 µSv/h	0,19 %
	^{133}Ba	**177 µSv/h**	1,67 %	**93 µSv/h**	0,10 %	**177 µSv/h**	1,85 %	**25 µSv/h**	0,25 %

Table 16: Dose Rate and statistical error obtained with TRIPOLI at 4 different points

8.5.2 Comparison with Deterministic Codes

Since Mercure code was taken as a reference for deterministic dose rate calculation, next step in validation of results is comparison with the results obtained using Monte Carlo TRIPOLI code. In the Table 17 are reported discrepancies in percentage for each isotope and for total value of dose rate.

H*(10)	Mercure / Tripoli4	Point A1	Point A2	Point B1	Point B2	mean
Béton normal densité 3,5 g/cm3	TOT	-5,88%	-10,95%	-4,78%	-6,64%	-7,06%
	^{60}Co	106,95%	100,25%	114,58%	95,10%	104,22%
	^{152}Eu	-13,27%	-16,71%	-11,26%	-12,75%	-13,50%
	^{154}Eu	-15,83%	-19,85%	-14,42%	-19,20%	-17,32%
	^{133}Ba	35,90%	19,46%	30,29%	27,85%	28,37%

Table 17: Discrepancy for each isotope separately and total discrepancy

One notice a general underestimation for ^{152}Eu and ^{154}Eu from -11% to – 20% and an overestimation for ^{133}Ba and ^{60}Co from +20% to 115%; it is the biggest in case of Cobalt. **In the other hand, the total dose rate is underestimated and discrepancy for the total dose rate ranges from -5 to -11 % which is quite satisfying.**

In the Table 18 are reported discrepancies between Tripoli and other Deterministic codes used for previous dose rate calculation with homogeneous source representation. **One notices a general overestimation for all the nuclides from +5% to +36%. It results a mean discrepancy of 7% and 13% for Mercure and DosiMex respectively.**

For further analysis it would be necessary to perform more detailed investigation and calculation with different libraries. Discrepancies may arise from the consistency of data library.

H*(10)	Deterministic code / MC code Tripoli4	Point A1	Point A2	Point B1	Point B2	mean
Béton normal densité 3,5 g/cm3	DOSIMEX-G	12,35%	13,43%	12,25%	13,92%	12,99%
	MicroShield	11,84%	16,15%	34,98%	11,99%	18,74%
	Mercurad	26,85%	27,68%	28,32%	28,47%	27,83%
	Mercure	5,88%	10,95%	4,78%	6,64%	7,06%
	Narmer	34,11%	34,96%	33,79%	35,14%	34,50%

Table 18: Discrepancy of Deterministic codes versus TRIPOLI code

9 DOSE RATE ASSESSMENT WITH DOSIMEX: RESULTS AND DISCUSSION

9.1 Introduction

In the previous chapter consistency of results obtained with DosiMex was shown. This allows the use of this application to proceed with the calculations and analyses at different cutting stages. DosiMex has an easy-use interface that allows constructing some model using simple functions and geometries. DosiMex was used to calculate the dose rate at different cutting stages.

In this chapter dose rate will be calculated considering three different models for the concrete block activation:

- ✓ Homogeneous sources- taking as an activity input:
    - Average activity in the block
    - Maximum activity in the block
- ✓ Gradient source

9.2 Dose rate calculation

The ambient dose equivalent and kerma at each cutting stage of the nuclear part of concrete have been are computed. Table 19 gives the results of the calculation.. Values are obtained considering a gradient source modeling.

			Dose Rate H*(10) µSv/h	
			More activated side	Less activated side
Layer 1	**Distance**	**1 m**	7	406
		1 cm	12	905
Layer 1,2		**1 m**	0,403	447
		1 cm	0,663	913
Layer 1,2,3		**1m**	0	442
		1 cm	0,0018	905

Table 19: Dose Rate computed with a gradient source modeling for each stage of reactor block cutting

Looking at the calculation results, the following preliminary conclusions can be made:

- **In the central hole of massive concrete block, after removing of vessel and graphite, dose rate is constant ~ 0.9 mSv/h**. The dose rate will remain the same, even after the removal of two outer layers of very low level waste. This is explained by the fact that the main contributor to dose is the Layer1-part of the concrete that will be removed last. Since workers won't operate form this position ,special attention to this value should be given in waste management operations and accidental conditions
- **At less activated side of Layer 1 the dose rate is 12 µSv/h at contact**.

- Layers 2 and 3 behave as a shielding, protecting workers form the most irradiated part. The dose rate at contact exhibits very low values, 0.6 µSv/h for Layer 2 and 0.002 µSv/h for Layer 3.

In addition, calculations with homogeneous source are performed, taking as an input average activity and maximum activity. Their values are shown in Table 20:

Activity Bq/g		
	mean	max
Layer 1	7 378	14 500
Layer 1,2	7 256	14 500
Layer 1,2,3	7 250	14 500

Table 20: Average and maximum activity for three different phases of cutting

The results of this calculation are reported in Table 21.

			Dose Rate H*(10) µSv/h	
			Mean Activity	Maximum Activity
Layer 1	**Distance**	**1 m**	85	168
		1 cm	570	1124
Layer 1,2		**1 m**	133	266
		1 cm	560	1121
Layer 1,2,3		**1m**	202	406
		1 cm	558	1114

Table 21: Dose rate computed with homogeneous source

Observing the tables it is obvious how misleading can be considering the average value of the activity as an input for dose rate calculation. There is a significant underestimation of the dose rate coming from the more activated side of the bloc, and on the other hand a significant overestimation of the dose rate coming from the less activated side of the block.

In such a delicate situation where there is different activation of block sides, dose received by worker strongly depends on the position of the worker to the block. We know that the dismantling operation will proceed from the outer part towards the inner part of the reactor block where the core of the reactor was. For this reason, dose rate coming from the less activated side of the block is taken for the computation of the collective dose for workers.

However, the dose rate coming from the more irradiated part of concrete has to be taken into account in the waste management process. This will be developed in chapter 10. **Considering maximum activity and an homogeneous source we can roughly approximate the value obtained with gradient source at the more activated side of the block.**

9.3 Dose rate as a function of cutting depth

From the point of view of the dose to the worker during the dismantling the main contribution to the dose will come from the Layer 1. Table 22 gives the characteristic of the wall model used in calculation for obtaining dose rate. It appears interesting to see how dose rate changes with the cutting depth. Activation data is taken from 2008 and afterwards computed for 2014 (New data taking into account the natural decay).

Table 23 gives the calculated values for dose rate, while Figure 24 illustrates it. For the calculation, a gradient activation was assumed.

block size cm			volume	mass
length	width	height	cm3	kg
414	55	244	5.56E+06	1.94E+04

Table 22: Characteristic of the wall model for Layer 1

Dose rate µSv/h		
Distance form core	2014	2008
10	312	452
20	154	214
30	65	91
40	27	37
55	7	10

Table 23: Dose rate at different cutting depth

From Table 23, we can see that 10 cm of concrete have significant influence on the drop in the dose rate. This sharp drop follows an exponential profile as shown in Figure 24.

Table 23 and Figure 24 also confirm that both dose rate and activity follow the same trend between 2008 and 2014: a decrease by 30 % due to the radioactive decay.

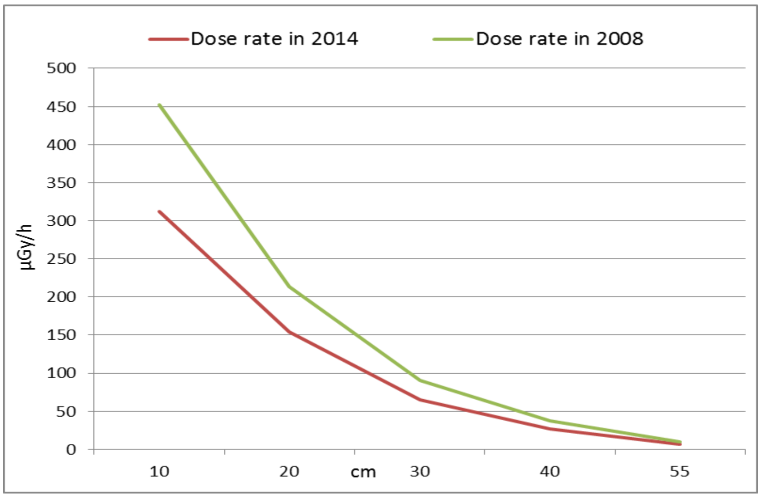


Figure 24: Dose rate at outer part of the concrete block as a function of the cutting depth

9.4 Isotope contribution and intensive property of dose rate

Next step would be determining the main isotope responsible for the dose in the air. To see which of those elements contributes the most to the dose rate, we choose the case with 3 layers and maximum activity and compared the values obtained. Dose rate is calculated on contact.

	Dose rate µSv/h	Percentage %
^{60}Co	15.63	1.3
^{152}Eu	873.23	76.7
^{154}Eu	42.35	3.7
^{133}Ba	208.74	18.3

Table 24: Dose rate computed for each isotope separately

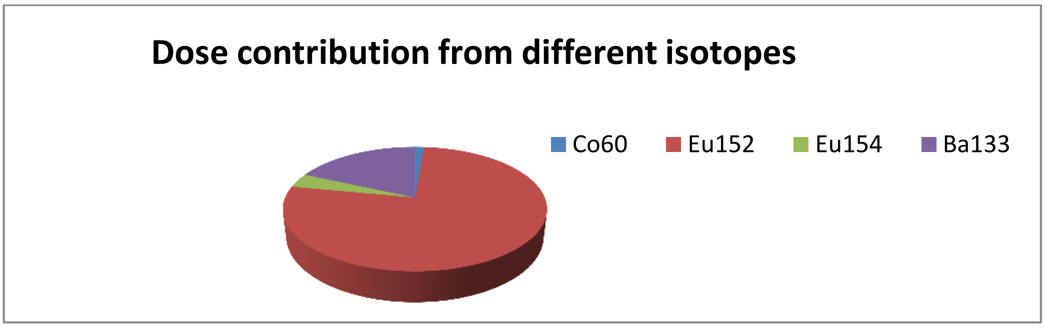


Figure 25: Dose rate contribution from different isotopes

Obviously the greatest contributor is 152Eu, followed by 133Ba in significantly smaller amount.

10 Waste inventory and management

10.1 Introduction

According to the CEA estimation, about 90% of the waste produced in the dismantling operation is concrete. Figure 27 shows the proportions of the categories of waste.

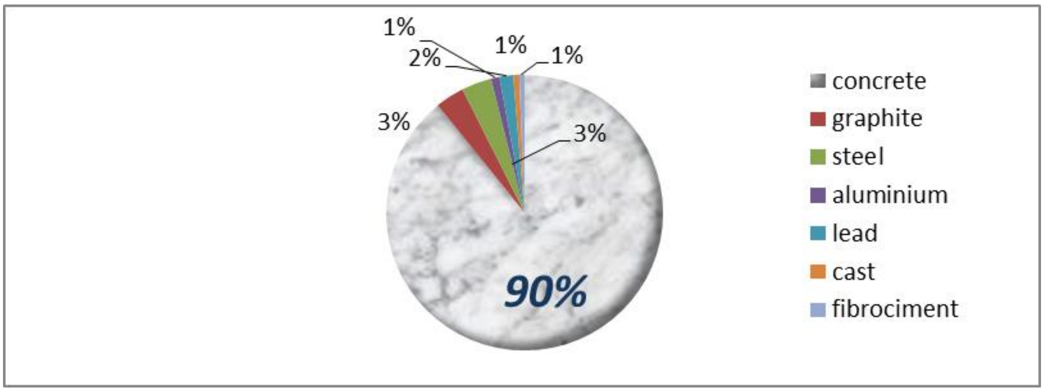


Figure 26: Estimation of primary waste coming from dismantling operations

Big part of the concrete waste is the one coming from reactor massive block decommissioning operation. In this chapter the waste storage planed by STMI (presented in their technical offer [Ref 40]) will be presented. Afterwards, the results of calculation for dose rate coming from waste casks will be shown.

Following the radiologic characterization, the waste categorization of concrete has been carried out. To remind, we refer on Table 6 previously introduced:

	Reactor block	
	Volume (m3)	Mase (t)
Conventional (CNV) :	46,625	163,1875
Very Low Activated (TFA) :	21,514	75,299
Low Activated (FA) :	14,282	49,987
Total	82,419	288,4735

10.2 Waste storage

In the first phase of the dismantling operation in phase A-0 of Table 1, an area of 20 m3 will be established outside of the building to serve as a site for the disposal of conventional waste. In the rector hall during dismantling of the nuclear part of concrete block (Phase B), buffer zones for VLLW and LLW disposal will be made.

Conventional waste

The storage area of conventional waste will be protected from the weather by a vinyl tarp. All the outgoing conventional waste will be radiologically monitored before leaving the reactor hall. Waste will be carried out by a forklift, and then deposited on the storage area of conventional waste.

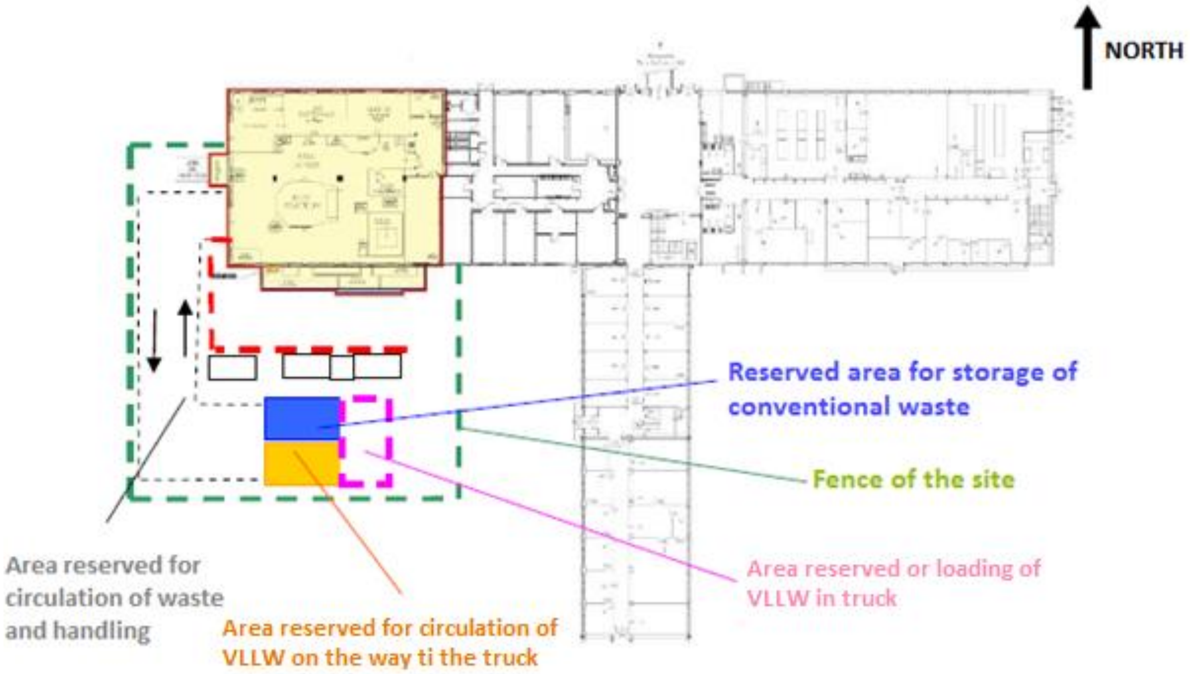


Figure 27: Map of the site for conventional waste and VLL waste circulation

VLL Waste

All waste generated during the VLLW dismantling of the installation will be monitored radiologically and before leaving the reactor hall. They will then be supplied by a forklift to the truck. Interim storage won't be created outside of the building for VLLW. They will only circulate through that area in the loading on trucks.

LL Waste

All LLW produced during decommissioning will be discharged in 5m3 containers that will be located in the buffer zone. Considering the limitation of crane and waste volume 9 containers will be needed. The concrete blocks will be radiologically checked before leaving the airlock. Then they will be brought using the crane into the container.

When a container of LLW will be fulfilled, it will be moved using the crane from the buffer zone for spectrometry analysis. After spectrometric measurements, a radiological control of the box will be carried out. The container will be then brought outside the building to be taken in charge by the CEA services in accordance with an agreement signed between STMI and CEA. No temporary waste storage area will be created for LLW and CEA will take over the containers as soon as the containers are ready.

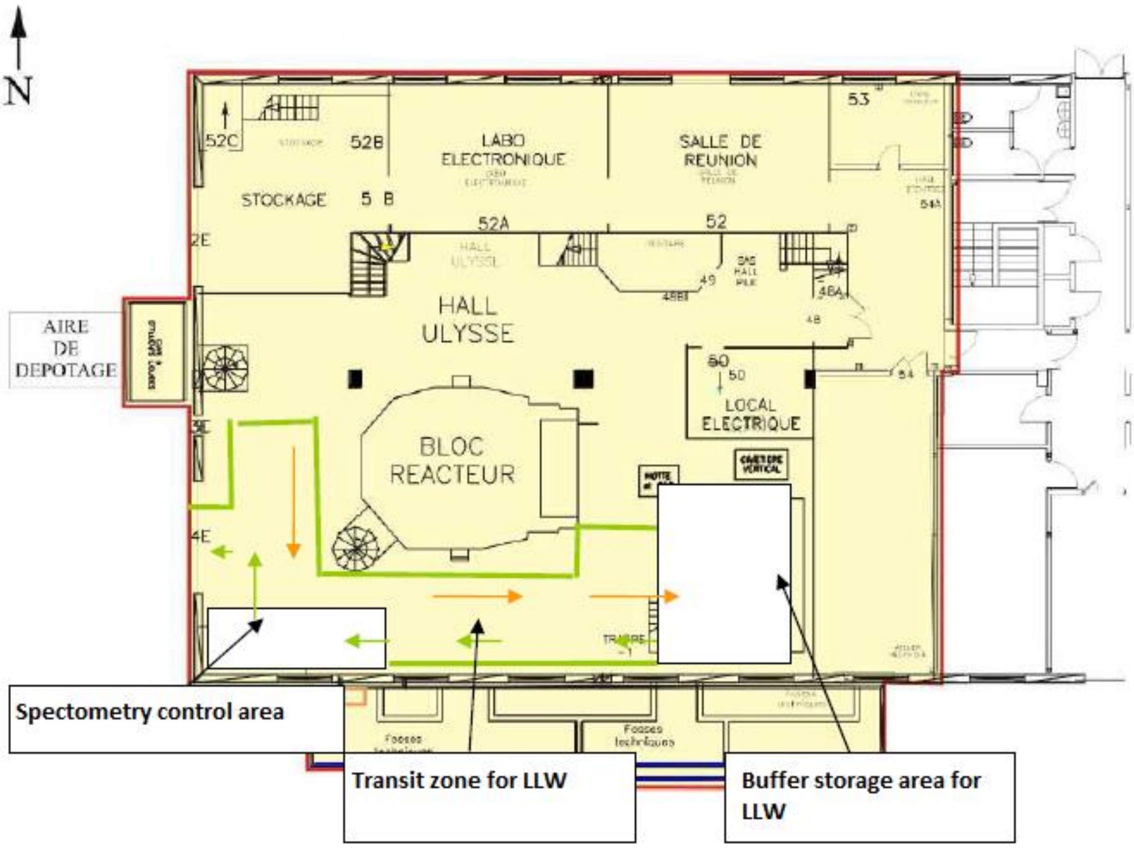


Figure 28: Buffer storage area for LLW and transit zones of LLW

10.3 Importance of LL waste block arrangement

The filling of the containers with LLW was studied in order to be better optimized. The boxes will be filled following the next principles:

- Optimization of the useful volume of the container
- Consideration of the crane maximum working load is 6t
- Dose rate reduction with proper arrangement of the blocks

The crane maximum working load should never be overpassed, even for the most filled packages.

Taking into account the container geometry from documentation issued by CEA [Ref 46] and using DosiMex, dose rate was calculated on the contact with the full container placed in the buffer zone. In the Figure 30 we can see the LLW container of 5 m^3 and its walls. The hole between interior and exterior steel will be filled with concrete at the storage place.

Gradient source was considered for the calculations of dose rate on contact with the waste container. Simple model of homogeneously filled container with the LLW was assumed. This consideration will give the most penalizing results possible.

For simulation of container wall we took:

- ✓ 0.4 cm of steel
- ✓ 10 cm of the air (the hole will be afterwards filled with concrete)
- ✓ 0.4 cm steel

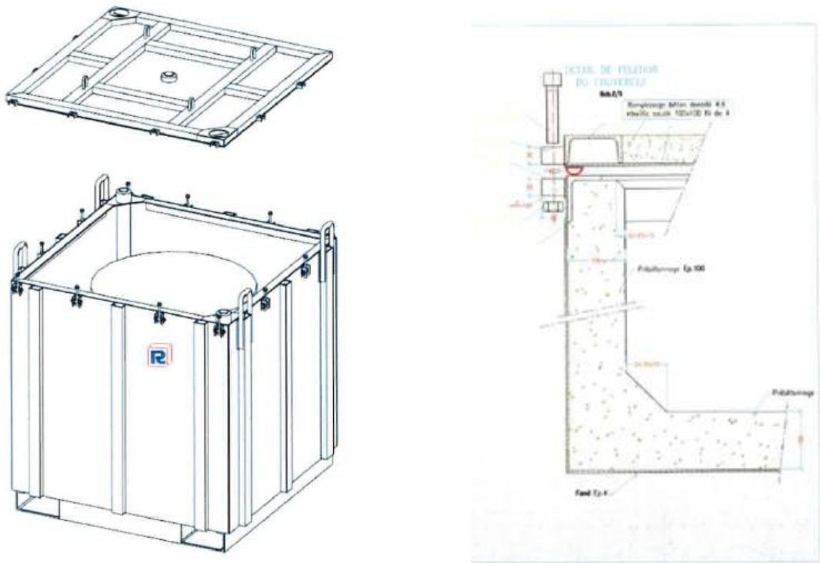


Figure 29:5 m^3LLW container and its wall

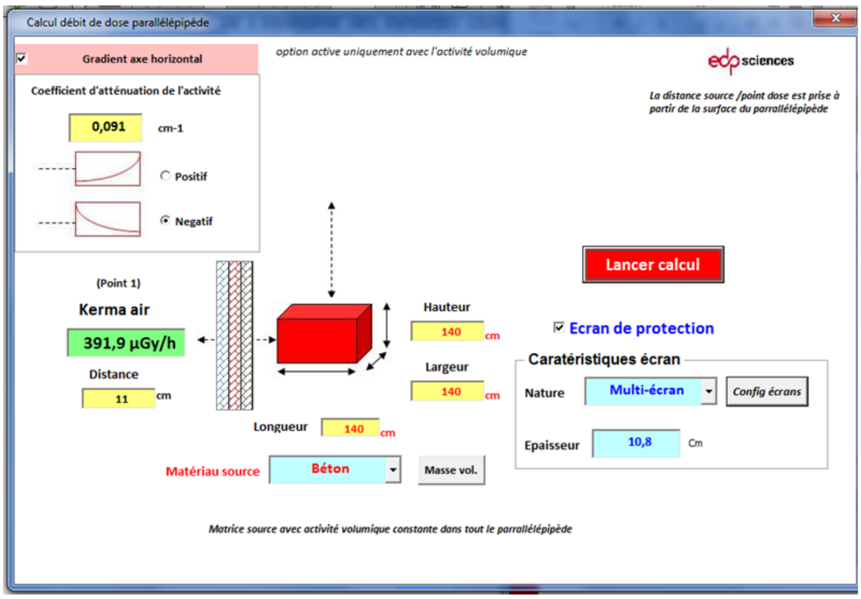


Figure 30: Simulation of dose on contact with container using DosiMex

Subsequently, two other calculations were performed. First one, taking the same case just after pouring concrete inside of the container (concrete as shielding instead of air).The second was done considering the less activated part of concrete in contact with container. Dose rate calculations are listed below:

- › More activated part of the block in contact with the container → 462 µSv/h

- More activated part of the block in contact with the container after pouring concrete $\rightarrow$ 40 µSv/h

- Less activated part of the block in contact with the container $\rightarrow$ 1,6 µSv/h

Analyzing these results we can see that 10cm of concrete reduces the dose rate on contact by factor 10. Also, putting the less irradiated side of the block in contact with container changes the dose rate dramatically. Having this in mind for decommissioning operation, the following rules are proposed:

- Color painting/marking of blocks immediately after cutting for distinguishing the sides with higher activation (See figure 31)

- Paint the block to fix any contamination

- Arranging the blocks in such a way that the less irradiated part is in the outer part of the container

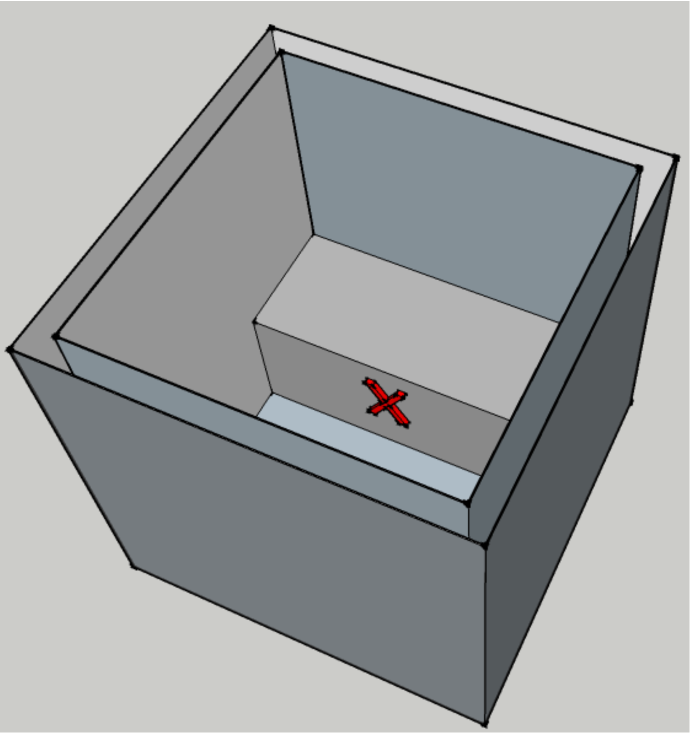


Figure 31: Optimization of block arrangement

11 THE ELABORATION SCENARI

11.1 Tools used

There are several possible tools that could be used for concrete cutting:

- Jackhammer
- Cable saw
- Circular saw

One of the important factors that had to be taken into account is minimization of noise in the classrooms of the INSTN that are in the same building as the reactor. Jackhammer is identified as the noisiest one but on the other hand reduces the time of the operation. Compared to the other methods, jackhammer produces a lot of dust. This is a big disadvantage since part of concrete is activated. Also there will be bigger release of tritium if jackhammer was used. For this reason, jackhammer was excluded as a potential tool for performing the operation of dismantling reactor block, except for the foundation bellow the reactor block.

Table 25 gives comparison of other two tools that can be used to cut the concrete block: the cable saw and the circular saw.

Cable saw		Circular Saw	
Advantages	Disadvantages	Advantages	Disadvantages
No limitation in thickness	-Needs a lot of sampling -Significant time for installation and setting of equipment -The total time of cutting (the ratio is about 0.15 m^2 / h) -Change of the cable geometry during cutting -Effluent management is complicated (splashing) -Increase in safety requirements - Guard needed during cutting process - safe distance - No coactivity in the airlock during cutting -Risk of the cable rapture	-Doesn't needs lot of sampling -Time for installation and setting of equipment reduced -Effluent capture effluent is easier -The sizing of blocks corresponds to the characteristics of the disc saw -More comfort for operators because the implementation is easier	Changing of disk - A first pass with a disc of 800 mm - A second pass with a disk of 1200 mm - A final pass with a disk of 1600 mm -The maximum depth of cut is 730 mm

Table 25 Comparison of tools used for cutting

STMI choose circular saw for cutting of the massive concrete block since it was the most suitable according to the requirements.

11.2 Collective dose computation

As a part of my internship a tool for determining collective dose for workers was developed. It is taking in to account all the daily operations during dismantling of the reactor block. It gives the possibility to manipulate important parameters such as:

- Number of workers for each operation
- Time of the operation
- Distance

Based on this it is possible to perform parametrical analysis and to choose the optimum scenario, the one that finds compromise between time spent and the dose received by workers. **All this was done using ratios provided by Mr Jeanjacques.**

Considering that 2m^3 have been removed during the day and properly handled, 41 working days is needed in order to perform the cutting of the block and waste arrangement at the same time. As a fist assumption, 6 workers are considered for performing the operations. Two of them are responsible for tools installation and removal, other two for handling system installation and removal. Finally one worker will be necessary for cutting operation and one for block handling.

Shielding implementation hasn't been considered as a radiation protection method in this case since it is too expensive for such a low dose rate.

Increasing the distance during the operation has shown to be the best practice since adding a half meter distance to the activated parts reduces significantly the total dose .For this reason, training of the workers is mandatory so that they can understand the radiation protection principles and their importance.

Table 26 contains the basic hypotheses for the number of workers, the distance and the time of the operations. It gives the ambient dose equivalent for each operation. **Considering this, the collective dose for 41 days of the operation is 567 man-µSv, as reported in Table 27.**

					Dose rate			
					Conventional	Nuclear Waste		
	Task	Time of the operation	Number of Workers	Distance		VLLW (Layer 1,2,3)	VLLW (Layer 1,2)	LLW (Layer 1)
		hours		cm	nSv/h	nSv/h	nSv/h	µSv/h
1	Tools instalation	1.5	2	50	0	1.46	533	9.1
2	Cutting	1.2	1	200	0	0	226	3.61
3	Tools removing	0.8	2	50	0	1.46	533	9.1
4	Handling System instalation	0.8	2	20	0	1.69	618	11
5	Block Handling	0.8	1	200	0	0	226	3.61
6	Handling System Removing	0.5	2	20	0	1.69	618	11
		Total 7 working hours						
				Volume m3	**47**	**21**		**14**
				Days	**23.5**	**10.5**		**7**

Table 26: Operational tasks and dose rate associated to each one

	Dose rate				
	VLLW(Layer 1,2,3)	VLLW(Layer 1,2)	LLW (layer 1)		
	nSv/h	nSv/h	µSv/h		
Total dose per layer/day	11	4511	78	**82**	µSv/day
Collective dose	58	23681	544	**567**	man-µSv

Table 27: Results for the collective dose rate in µSvm with the basic assumptions

Finally, observing the results from the Table 27 we can see that the biggest contributor to collective dose is the Layer 1, which is classified as a LLW. Its share presents 95% of total collective dose. This needs to be taken into account during the dismantling operation. Specific measures should be taken in order to reduce as much as possible the number of days devoted to cut and to manage the wastes at this state of the operation.

11.3 Sensitivity analysis for collective dose computation

There are general guidelines for controlling the exposure to ionizing radiation:

- minimizing exposure time,
- increasing the number of workers,
- maximizing distance from the radiation source,
- Shielding from the radiation source.

Knowledge of the status and history of the nuclear facility is essential for successful decommissioning planning, decommissioning strategy selection and execution from both safety and technical points of view. It is desirable to ensure that measures are taken during the entire operational phase to document the physical inventory of equipment, inventory of hazardous material and the radiological inventory. Ideally the knowledge of the operational staff is utilized during the decommissioning phase. Sometimes, though, experience has shown that during dismantling operation there are delays that impact the cost and time of the project. Also collective dose will be change in such a situation.

Tables from 28 to 35 are gathering values of collective dose when we are changing one or more parameters: distance, number of days and number of workers.

added distance	-20	**0**	20	50	100
collective dose (man-µSv)	652	**567**	494	400	283

Table 28: Sensitivity of collective dose to the distance

The case presented in Table 26 has been chosen as the reference. From Table 28 an important impact of working distance can be noticed. *By adding only 20 cm of distance, the value of collective dose decreases by 73 µSv/h.*

days for layer1	3	5	**7**	9	11
collective dose (man-µSv)	257	412	**567**	723	878

Table 29: Sensitivity of collective dose to days devoted to dismantle and store LLW bloc

Having in mind that in the Layer 1 the LLW part contributes the most to the collective dose, it is expected that an increase in the number of days devoted to the cutting activities and removal of this layer would significantly increase the total collective dose. *Adding 4 days of operation to this task, makes the value of collective dose reaching 878 man-µSv .* Decreasing the number of days would be possible by adding an additional cutting tool. In practice this is not feasible because of the limited space and cost

number o workers added for cutting	**0**	1	2	3
collective dose (man-µSv)	**567**	620	673	726

Table 30: Sensitivity of collective dose to number of workers performing cutting

number of workers added for handling	**0**	1	2	3
collective dose (man-µSv)	**567**	671	776	880

Table 31: Sensitivity of collective dose to number of workers performing handling

number of workers added for tools	**0**	1	2	3
collective dose (man-µSv)	**567**	695	843	991

Table 32: Sensitivity of collective dose to number of workers installing are removing tools

number of workers added for all sequences	**0**	1	2	3
collective dose (man-µSv)	**567**	878	1187	1498

Table 33: Sensitivity of collective dose to added number of workers for all the tasks

From the Tables 27 to 30 we can see the expected increase in collective dose with the increase of the number of workers.

number of workers added for handling&installation	**0**	1	2	2	2	2	2
days for layer1	**7**	7	7	5	4	3	2
added distance	0	0	0	0	0	0	0
collective dose (man-µSv)	**567**	825	1082	786	638	489	341

Table 34: Sensitivity of collective dose to number of workers and time for LLW management

number of workers added for handling&installation	**0**	1	2	2	2	2	2
days for layer1	**7**	7	7	5	4	3	2
added distance	**50**	50	50	50	50	50	50
collective dose (man-µSv)	**401**	582	764	555	450	345	242

Table 35: Sensitivity of collective dose to nmb of workers, time for LLW management and distance

Tables 34 and 35 are examining impact of the contribution of several parameters. **General conclusion is that the optimum scenario would be similar to the first one, since in practice is**

impossible to finalize the works on Layer 1 in less than 7 days without implementing an additional tool. This operation, same as shielding is not of interest at such dose rate level and in such a limited space.

12 THE SAFETY ANALYSIS

The CEA plans that the overall radiation dose received by workers, within about two years of work, won't exceed 2man-mSv. This estimation is based on the initial state of radiological property of facility and the experience gained from previous decommissioning, in particular the experimental reactor RUS (Strasbourg). This level is very low and as a reference, the maximum dose received by a worker in the nuclear sector over a year is 20mSv.

Although this may seem optimistic estimate, the important is the follow-up measures and protection adopted by the client. These monitoring measures (use of personal dosimeters agents, detection device for abnormal release of radioactive elements) and protection (definition of zoning conventional and nuclear, subsequent conduct of conventional and nuclear sites, containment of nuclear construction, treatment of effluents, training of officers and taking into account the human factor in organization of work ...) comply with the rules of the art of the nuclear business. They are properly summarized in the impact assessment and developed in the study of control risks.

12.1 Nuclear Risk

Initially, all the non-active blocks that are under conventional waste will be cut and removed. They do not cause exposure so there is no radiological risk related to this phase of operation.

Gradually the operation will proceed toward the center of the reactor block. The workers will be first subjected to external exposure (activation of concrete block and elements constituting the reactor) and also, to the risk of internal exposure coming from resuspension of tritium and aerosols.

12.1.1 Internal exposure-Tritium and Aerosols

Risk coming from tritium

Contribution of tritium in total activity of concrete block is around 67%, so its impact is important to analyze. It decays into helium-3 by beta decay and has a half-life of 12.3 years. Beta particles from tritium can penetrate only about 6.0 mm of air, and they are incapable of passing through the dead outermost layer of human skin.

The only danger regarding tritium is the internal contamination. Several hypotheses are made in order to estimate the worker exposure from tritium.

-sawing 2 cm thickness of concrete block causes transfer of 4 GBq activity in a volume of 10 m^3 of water, where 50 liters per day are evaporating(experience of Strasbourg reported those values).

If we transpose this observation to ULYSSE reactor, the total tritium release is estimated to be 0.8GBq 20MBq /d (Aj) or 0.83MBq /h (Ah).

Calculation of the exposure of one worker present is airlock:

Time of the presence (Tp)	7	h
Duration of the work (Dc)	40	days
Volume of the air lock (Vs)	100	m3
Breathing rate of a worker(Dr)	1,2	m3/h
DPUI(HTO)	1,80E-11	Sv/Bq

Table 36: Parameters used for tritium exposure estimation

The airlock is ventilated and the average volume activation in ^{3}H remains constant or:

$$A_v = \frac{A_v}{V_s} = 8300\ Bq/m^3$$

E_j is the daily exposure of the operator $E_j = T_p \times DPUI \times A_v \times D_r = 1.25\mu Sv$

E_t is the total exposure du

ring the project $E_t = D_c \times E_j = 50\mu Sv$

Total exposure of an operator without respiratory protection is estimated at 50 µSv for completing all the work on massive concrete block.

Calculation of exposure that would undergo a person considered public, located near the ventilation duck exit at the rooftop.

Flow extraction of air lock (Q)	3000	m3/h
Time of presence (Tp)	24	h

Table 37: Parameters used to asses public exposure

$$A_v = \frac{A_j}{(Q \times T_p)}$$

This means that the person will receive

$E_p = A_v \times D_r \times T_p \times DPUI = 0.15\mu Sv$,or 6 µSv if person stays present during the all project of cutting and dealing with the waste at the roof top close to the chimney.

At the Saclay Center two different methods are used for the survey of the release or the atmospheric contamination with tritium which is in liquid form:

- Differential chamber of 10 liters connected to control panel for gases (except tritium),

- "Bubbler" for tritium

Aerosols

In the normal site conditions the amount of aerosol resuspension is negligible while during the dismantling operation there can be significant amount of the aerosols present in the working area.

Regarding the monitoring of discharges, the most suitable solution is to use a monitoring device (PIAFF) equipped with a fixed filter which is collected weekly in order to be measured. This monitoring device is located after the barriers of high efficiency filters. In order to respect zoning method according to radiological property, it is necessary to set up atmospheric sampling in the airlock.

12.1.2 External Exposure

The risk of external exposure is related to the evacuation and storage of nuclear part of concrete blocks. Also concrete contains various metal parts like reinforcement and screws that can be locally very activated.

Prevention measures:

- Optimization of the time of operation
- Keeping distance from nuclear waste if there is not specific task that need to be carried out
- Storage of Low Level Waste enclosed by biological protection
- Use of passive and active dosimeters

In the Table 38 are summarized potential nuclear risks at different operation stage.

Phase	Exposure	Level and source	Protection measures	Measuring means
Conventional waste	Internal	Non concerned	Unnecessary	Unnecessary
	External	Negligible	Unnecessary	Portable dosimeters Photoluminescent dosimeter
Very Low Level Waste TFA	Internal	Aerosols and tritium	Airlock	PIAFF filters Bubbler Smear control Direct measurement Airlock measurement
	External	Negligible , tens of µSv	Distance	Portable dosimeters Photoluminescent dosimeter
Low Level Waste FA	Internal	Aerosols and tritium	Airlock	PIAFF filters Bubbler Smear control Direct measurement Airlock measurement
	External	Several µSv	Optimization of time Distance	Portable dosimeters Photoluminescent dosimeter Cartography
Waste casks in disposed in hall	Internal	Non concerned	Unnecessary	Unnecessary
	External	Several µSv to 400 µSv depending on the block arrangement	Biological protection Distance	Portable dosimeters Photoluminescent dosimeter

Table 38: Nuclear risks at different operation stage

12.1.3 Release of radioactive material

The risk of spreading radioactive material in this phase comes from nuclear part of concrete that needs to be cut and stored. During the operation it is possible to find contamination in different airlocks, on workers outfit and tools used for cutting. Also, nuclear waste packages present potential danger if their confinement is not maintained.

Prevention of spreading radioactive material is based on use of static confinement between personnel and material, or between environment and material.

During the entire dismantling operation the reactor hall acts as a containment barrier. Airlock also presents a static confinement that can be complemented with additional dynamic confinement that is another ventilation system.

Prevention measures

- Aspiration of the dust ,that will be produced during the cutting process
- Using cutting process that generates as little of dust as possible
- Use of static and dynamic barriers
- Marking the path that is used for transport of waste packages
- Protection of waste package during loading to avoid external contamination
- Regular cleaning and decontamination of workplace
- Visual inspection of integrity of confinement before any operation within them
- Regular monitoring of filters and ventilation
- Periodic checks of the surface contamination
- Mobile radiation and aerosols detectors
- Control of the proper workers dressing and the way they remove protective cloths, and regular inspection after they leave the airlock

12.2 Non-nuclear risks

Non-nuclear risks that are not relevant for the operation of dismantling of concrete block are:

- Explosion
- Overpressure
- Internal flooding
- Chemical
- Asbestos
- Asphyxia

Other, more relevant risks are analyzed in the following chapters.

12.2.1 Fire

The risk of fire is related to the simultaneous presence of combustible materials, oxidizing (air) and activation energy (electricity).

The internal source of fire risk is mainly due to

- The presence of electrical components
- The oil in the hydraulic equipment
- The fuel present in the vehicle used for handling

Prevention and limitation of the consequences are based on:

- Limitation of combustible material present in the hall
- Turning the devices (appliances) off when they are not used
- The network of automatic fire detectors with alarms reported to the central station security
- Visual inspection after performing work
- Training the stuff to use extinguishers and how to behave in case of an incident
- Fire extinguishers located at the facility in accordance with local fire risk, including agent
- Telephones, call buttons that are connected to central station of security

12.2.2 Handling

The risk of handling is related to the operation of handling heavy load, whose fall or collision with an obstacle could jeopardize the integrity of a containment barrier. Waste conditioning and disposal present a potential source for this kind of accident, as well as use of crane, hoists during the removal of concrete blocks.

The most penalizing consequences of handling incident are:

- The deterioration of equipment involved in risk management
- The fall and damage of a waste cask, which could lead to internal and external exposure of workers

Prevention of risk in handling is obtained following provisions as

- Periodic inspections and preventive maintenance of handling equipment
- Organize training sessions to train employees in proper manual handling techniques. The training should demonstrate how to avoid risks arising from manual handling and how to use mechanical aids, team carrying and personal protective equipment such as gloves, footwear
- Test prior starting work on site
- Marking the paths of waste
- Mark the weight of the load
- Determination of a lifting height limit
- Strict limitation of moving crane or any other equipment over waste packages if not justified
- Frequently used objects should be within easy reach to avoid bad postures
- Remove obstructions from the working area and carrying route. The route should be sufficiently wide and clear of obstacles so that the carrying activity is performed in an upright

posture without the need for body bending or twisting. Obstacles should also be removed to avoid falling

12.2.3 Electrical

An electrical risk is a risk to a person of death, shock or other injury caused directly or indirectly by electricity. The main hazards associated with these risks are:

- Contact with exposed live parts causing electric shock and
- faults which could cause fires
- fire or explosion where electricity could be the source of ignition in a potentially flammable or explosive atmosphere (for example in a spray paint booth)
- spreading of radioactivity and destruction of containment barrier

Preventive measures

- Use of electrical appliances according to legislation
- Connections of electrical appliances made by qualified and authorized personnel
- The application of the regulations on the protection of workers against the risk of electric shock
- Protection of power panels by differential circuit-breaker with a suitable calibration.
- Ensuring good electrical recording (check that no voltage) when working on electrical equipment.
- Respect safety distances
- Wearing appropriate personal protective equipment: gloves, mask, shoes, carpets, insulation stools
- Regulatory periodic inspections of electrical equipment and supplies.
- The cable insulation based on risk of local or area (the presence of water, dust, chemicals ...).

12.2.4 Work at height

The risk is related mostly to the installation of equipment for cutting where scaffolding will be used.

Prevention of injury is assured by

- Operators training
- Wearing appropriate equipment
- Good maintenance of the working zones (marking up, open area, limiting access)
- Use of tools that are controlled previously
- Presence of other operators that can alert and carry first air, before the intervention of medical service

12.2.5 *Injuries and falls*

Injuries and falls are the most common cause of injury at work. In order to prevent them, measures should be taken, such as:

- Plan pedestrian and vehicle routes to avoid contaminated areas
- Make sure lighting is sufficient and that slopes or steps are clearly visible.
- Keep walkways and work areas clear of obstructions.
- Consider how work is organized and managed, avoid rushing, overlapping the tasks, trailing cables
- Workers should tell their employer about any work situation that they think is dangerous, or if they notice that something has gone wrong with their health and safety arrangements
- Closing and marking pits, cavities or holes

12.2.6 *Loss of electrical power*

Risks related to the loss of power at the site of dismantling concern mainly the loss of ventilation, the availability of bridge maintenance and monitoring installation (radiation and fire).

Preventive measures adopted are as follows:

- Electrical equipment and installations comply with standards and regulation and are subject to regulatory controls..
- The sensitive organs of the plant (ventilation, handling bridge, monitoring devices gas emissions) have an uninterruptible power supply (generator, batteries) or circuit specific electric.
- A means of connection can help the entire electrical distribution installation using a mobile generator provided by the Centre.

In addition, in case of loss of power supply, operations are stopped, and the modes operating procedures and defining, for each step, what to do and actions to achieve to the construction and installation in a safe condition.

12.2.7 *Loss of ventilation*

Ventilation systems play a key role in limiting contamination in nuclear facilities. Each zone should be ventilated so that there is a pressure gradient between adjacent zones, the aim being to ensure that any movement of airborne radioactive contamination is from the zone with the lowest to that with the highest contamination.

Ventilation system should be monitored and instrumented to detect potential for leakage and to ensure the appropriate volume flow rates and depressions are obtained and maintained. Filters can be susceptible to damage and defect post testing and certification by the manufacturer. Operators should perform appropriate pre-installation inspection of filters to confirm no damage or defect is present and the filter type is appropriate for the installation.

12.2.8 Human and organizational factors

The risk arises from the accidental or deliberate non-compliance of instructions by the operator. The non -compliance instructions by the operators may be due to fatigue, lack of information or training.

Prevention of risk is obtained following the provisions:

- Supervision of stuff and monitoring workflow
- The adaptation of workplace
- Training of workers and information stakeholders about the potential risk
- The use of appropriate tools
- Ensure that there is an understanding of how tasks are carried out
- Introduce job rotation and job enlargement to avoid monotonous tasks and to make variations of posture possible. Altering postures is important so that the same muscles do not become too stressed. Job rotation can also help to increase work motivation
- Provide written procedures that have ‘place markers’ or spaces to tick off each step
- Strictly enforce rules about interrupting staff on critical tasks.
- Avoid performing carrying tasks under adverse environmental conditions since this will impose an extra effort on the worker, increasing the risk of injury. Provide sufficient visibility for the task (such as sufficient light and no reflections) so the worker can perform the work safely. Poor lighting increases the risk at falling or running into something. Working in a cold environment changes the precision of movements, while working in a hot environment can lead to excessive tiredness. Optimal conditions are:
  - temperature of 19-21°C
  - humidity of 35-50%
  - lighting of + 200 Lux

13 CONCLUSION

This document describes the dose rate calculation for cutting operations on ULYSSE training reactor that will be performed in near future. Peculiarity of ULYSSE concrete block is that the activation shows a sharp exponential decrease when going from the core to the external part of the reactor block.

Dose rate calculations were performed using different deterministic codes that use point-kernel method: Mercure, Mercurad, MicroShield, Narmer and DosiMex. **Mercure was taken as a reference deterministic code. Discrepancy between deterministic codes was lower than 31% for heavy concrete block calculations**.

A comparison between Tripoli MC code and other Deterministic codes was performed with homogeneous source representation. **A mean discrepancy of 7% and 13% was obtained for Mercure and DosiMex respectively. However, we have noticed important discrepancy for ^{60}Cobalt between Tripoli and deterministic calculations.** Discrepancies may arise from the consistency of data library**.** For further analysis it would be necessary to perform more detailed investigation and calculation with different libraries.

Calculations firstly assume simple models with homogeneous source. Then, more realistic representation of the source was considered with a gradient source. The values and discrepancies obtained with Mercure between the gradient and the homogeneous source are reported in the table below. **In the more activated side of the block (inner part of the block), overestimation is 30% while in the less activated side (outer part of the block) factor of the overestimation is around 87**.

Dose Rate	H*(10) µSv/h	
Mercure	more activated side	less activated side
Gradient source	926	14
	ref	ref
Homogeneous source	1207	1207
Δ	30,31%	8742,49%

Taking into account gradient source has shown great importance for the final dose assessment since homogeneous source turned out to give big overestimations. Consequently, it results in increase of operation time and cost.

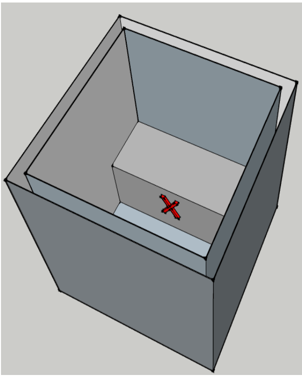
Collective dose for the concrete operation was estimated also and the results turned to be close to the values obtained by Amec Spie, despite the different method and input values used. To remind, **collective dose if the works were performed in 2008 would be 0.57 man-mSv while today in 2014 collective dose is 0.4 man-mSv.** Value obtained by Amec Spie for 2008 is 0.71 man-mSv. In the following tables different operation sequences can be seen with the basic assumptions for calculations concerning time, distance and number of workers. **As expected, the most important contribution to the collective dose comes from LLW (layer 1 inner concrete block) whose share**

accounts for 95%. Moreover, parametric calculations showed impact of distance, time and number of workers on the final value of collective dose.

	Task	Time of the operation	Number of Workers	Distance	Dose rate: Conventional	Dose rate: Nuclear Waste: VLLW (Layer 1,2,3)	Dose rate: Nuclear Waste: VLLW (Layer 1,2)	Dose rate: Nuclear Waste: LLW (Layer 1)
		hours		cm	nSv/h	nSv/h	nSv/h	µSv/h
1	Tools instalation	1.5	2	50	0	1.46	533	9.1
2	Cutting	1.2	1	200	0	0	226	3.61
3	Tools removing	0.8	2	50	0	1.46	533	9.1
4	Handling System instalation	0.8	2	20	0	1.69	618	11
5	Block Handling	0.8	1	200	0	0	226	3.61
6	Handling System Removing	0.5	2	20	0	1.69	618	11
		Total 7 working hours						
				Volume m3	**47**	**21**		**14**
				Days	**23.5**	**10.5**		**7**

	Dose rate: VLLW(Layer 1,2,3)	Dose rate: VLLW(Layer 1,2)	Dose rate: LLW (layer 1)		
	nSv/h	nSv/h	µSv/h		
Total dose per layer/day	11	4511	78	**82**	µSv/day
Collective dose	58	23681	544	**567**	man-µSv

Arrangement of concrete blocks was analyzed also. **Optimized waste arrangement that reduces the dose rate requires that the less activated side of the block gets in contact with the container.** More activated block sides should be placed face to face in the center of the waste container as illustrated on the figure below. This would reduce the dose rate by factor of 400. Arranging the more activated side of the LLW concrete block in contact with container would lead to dose of 462 µSv/h at 1 cm distance from the container.



Finally, for prospects, if we want to improve the assessment of dose rate, further calculations with statistical- Monte Carlo codes would be performed with different libraries. Also, measurements at the contact with the most irradiated side of the concrete block would be very useful for comparison. At this phase it is not feasible to perform such measurements since the graphite hasn't been removed yet.