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Article

Radiological protection in ITER*

H. MAUBERT[1], T. PINNA[2], M.-T. PORFIRI[2], S. CIATTAGLIA[1]

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Abstract **ITER will be the first fusion reactor producing much more energy than that necessary to establish and heat the plasma. This nuclear energy is mainly exported from the plasma by neutrons and deposited in the plasma facing components, generating nuclear activation. One of the elements used in the fusion reaction is tritium, a radioactive isotope of hydrogen. Although some feedback may be used from in service fusion machines, such as JET in England, the ITER radiological protection issues need to be assessed as early as possible. The protection against radiological hazards is, as usual shielding, waiting for radioactive decay, limitation of airborne contaminants, reduction of exposure time, etc. This has been done in the present design and has followed a first step of ALARA studies. Radiation fields have been calculated and work effort estimated leading to the evaluation of a collective yearly dose of 356 p mSv, below the project dose target which is 500 p mSv. A second step of occupational radiological exposure studies should allow refining further the dose estimate as well as the ALARA optimization.**

Keywords: **nuclear fusion / activation / tritium / ALARA / radiological protection**

RÉSUMÉ **La radioprotection dans ITER.**

ITER sera le premier réacteur à fusion qui produira plus d'énergie que celle injectée pour créer et chauffer le plasma. Cette énergie nucléaire est transportée par des neutrons et déposée dans les composants situés face au plasma, ce qui génère de l'activation neutronique et donc des produits radioactifs. L'un des éléments utilisés dans la réaction de fusion est le tritium, un isotope radioactif de l'hydrogène. Bien qu'un retour d'expérience d'autres machines à fusion, notamment du JET, soit disponible, les enjeux de la radioprotection dans ITER doivent être traités aussi tôt que possible. La protection contre les dangers des radiations sont, comme toujours l'interposition d'écrans, l'attente de la décroissance radioactive, la limitation des contaminants dans l'atmosphère, la réduction du temps d'exposition, etc. Ces mesures ont été mises en œuvre dans la conception actuelle d'ITER et les études ALARA déjà réalisées. Les champs de radiation ont été calculés et les temps d'intervention en milieu radioactifs évalués, conduisant a une estimation de dose collective pour les travailleurs de 356 H mSv/an pour un objectif de dose de 500 H mSv/an. Une seconde étape concernant les études de radioprotection conduira à affiner les estimations de dose et surtout à les optimiser dans le cadre d'une approche ALARA.

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[1] ITER Organisation, Cadarache Joint Work site, 13108 Saint-Paul-lez-Durance, France.
[2] ENEA FUS/TEC, Via Enrico Fermi 45, 00044, Frascati, Rome, Italy.

1. Introduction

ITER will be the first fusion reactor producing much more energy than that necessary to establish and heat the plasma. This nuclear energy is mainly exported from the plasma by neutrons and deposited in the plasma facing components, generating nuclear activation. One of the elements used in the fusion reaction is tritium, a radioactive isotope of hydrogen. Although some feedback may be used from in service fusion machines, such as JET in England, the ITER radiological protection issues need to be assessed as early as possible. This paper is a summary of the present state of the occupational radiological exposure assessments and ALARA evaluations.

2. Short description of the ITER facility

The fusion reaction is realised in a plasma magnetically confined in a machine called Tokamak (Fig. 1). The Tokamak comprises a vacuum vessel (VV), surrounded by superconducting magnets. Inside the cryostat vacuum ($<10^{-4}$ Pa) is maintained (How, 2007). The cryostat is surrounded by a concrete bioshield providing a biological protection for workers.

Access for maintenance to the inside of the vacuum vessel is possible through ports which are on three levels, divertor, equatorial and upper level. These ports are normally shut by port plugs. The bioshield is surrounded by galleries.

There are also auxiliary nuclear buildings: the hot cells where the plasma facing components (blankets and divertor) inside the vacuum vessel are refurbished or discarded, and the tritium building for the processing and storage of the hydrogen isotopes 1, 2 and 3, and where the tritiated water is processed. The ventilation systems and the air detritiation systems are also located in the tritium building.

3. Sources of radiations

All the calculations and radiation field assessment are based on conditions in the 20th year of operation, at the end of the ITER experimental program. It includes two 10 years operation phases each of them modified by the addition of a 6-day “aggressive” campaign in which the machine should be used at the maximum available capacity prior to shutdown (Iida *et al.*, 2004).

The radiation sources in ITER are:

- the primary neutronic field resulting from the fusion reaction $^{2}H + ^{3}H \rightarrow ^{4}He + n$ occurring in the vacuum vessel, however, since access to the Tokamak

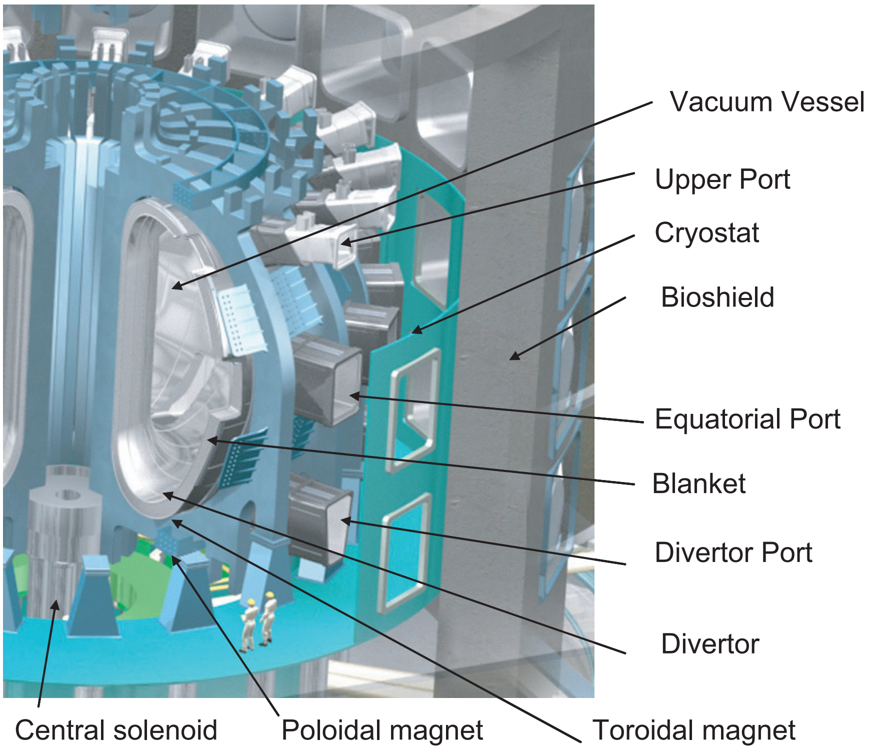


Figure 1 – ***3-D view of the Tokamak, cryostat and bioshield.***
Vue 3-D du Tokamak, du cryostat et de la protection biologique.

building is restricted during operations, the exposure of workers to neutrons is impossible;

- the gamma radiation emitted by activated products, including plasma facing components, vacuum vessel structures, activated corrosion products generated in the cooling loops and activation of the inner wall of cooling water pipes, potentially leading to external irradiation, and loose contamination from activated dust generated in the vacuum vessel, potentially leading to inhalation of radioactive materials. To determine the risk of exposure, the main radionuclides are the isotopes of cobalt-58 and -60, manganese-54 and -56, iron-59, tungsten-187 and other isotopes of tungsten, tantale and rhenium, chromium-51, niobium-94;
- tritium used as fuel for the fusion reaction. The inventory of tritium is approximately 4 kg distributed equally in the long term storage, the tritium building, the Tokamak and the hot cells and radwaste facility;
- wastes, still containing tritium and gamma emitters.

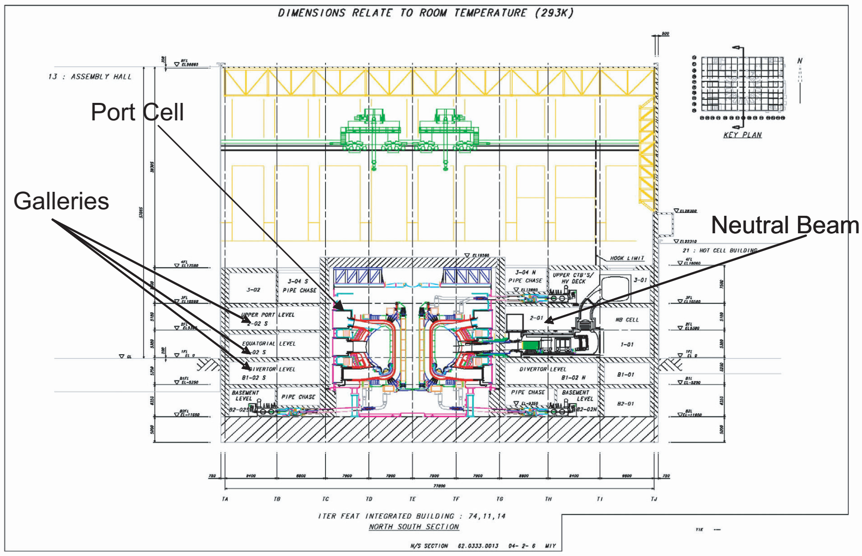


Figure 2 – ***ITER Tokamak Complex Full Vertical Cross Section Including Crane Housing.***
Coupe du complexe Tokamak d'ITER incluant les ponts roulants.

4. Location of the radiological hazards

The purpose of this subchapter is to identify the areas with radiological activity where a human intervention is necessary (ITER, 2006). The logic is to start from the VV where the fusion reaction occurs and where the higher dose rates are expected, and to enlarge the scope of the investigation both on a geographical and a process point of view.

The equipments situated inside the cryostat are designed to last the lifetime of ITER without maintenance. However as repairs may be necessary, the radiological conditions inside the cryostat will be monitored and a specific ALARA study of the intervention will be made.

Access of the remote handling devices to the vacuum vessel will require removing the port plugs. Although this is made remotely, human intervention is necessary to clear the port plug cells from all the connections to the plugs like cooling lines, wave guides, or other. Therefore one of the first concerns for radiological protection assessment is to evaluate the dose rates in the different port cells, and for each type of maintenance work to determine the duration of the work effort. This includes the following operations:

- replacement of plasma facing components, blankets modules or limiters and divertor cassettes. Some blanket modules are test blankets designed for experimental testing such as tritium breeding,
- maintenance of vacuum pumping systems, heating systems (Ion Cyclotron, Electron Cyclotron, Radio Frequency, …), diagnostics equipments.

The function of the ITER plasma diagnostic system is to provide accurate measurements of plasma behavior and performance. They are situated around the vacuum vessel in the equatorial and upper ports. At the lower level six "diagnostic" cassettes accommodate waveguides and optical diagnostics. The diagnostic sensors gather information about the plasma through primary windows, closing the vacuum vessel. For operators, the maintenance conditions of diagnostics are equivalent to the conditions for port cells clearing operations.

The NB H&CD (Neutral Beam Heating and Current Drive) system is designed to help in accessing the H-mode and heating plasma at $Q > 10$, provide steady state current drive capability, modify current density and q profile, provide plasma rotation provide power to sustain the density during shutdown and allow for controlled transition from H to L-mode at the end of burn. These equipments are located in a room called neutral beam cell. The maintenance operations on highly radioactive parts are done by remote handling but some human interventions may be necessary.

The cooling loops of the blankets and of the divertor will carry activated water and corrosion products. The source terms that are likely to affect the collective dose for the operations performed around TCWS (Tokamak Cooling Water System) components are mainly the Activated Corrosion Products (ACP) on the inner surface of the cooling pipes, the activation of the pipes themselves by the delayed neutrons of ^{17}N produced in the coolant and the airborne tritium in working premises. Other source terms like ^{16}N in the coolant or prompt radiation from plasma burning were not considered because of the extremely short half life of involved radionuclides.

The equipment necessary to the operation of the cooling loops will be installed in a room called TCWS vault. Human access to this vault will be necessary for visual inspection, maintenance of pumps, pressurizers, heat exchangers, filter and valves, leak tightness testing, cooling water sampling…

During the transportation of activated equipment by remote handling no human intervention is allowed in the concerned areas. The remote handling equipment will be maintained by remote handling inside the hot cells.

In the tritium plant, the tritium containing vessels, ducts, pipes, etc. should be under double envelope. A ventilation system will provide a renewal of the atmosphere, and the concentration of tritium in air should stay under detection level (~10^5 Bq/m^3).

The function of the hot cells systems is to process and repair components, tools, and equipment which have become activated by neutron exposure and/or contaminated with tritium or activated dust. Components which enter the system for repair may be diverted to the hot cell waste processing system. The hot cell waste processing and storage system provides up to 6 months storage of radioactive waste for an interim period prior to hand-over to the host country. Shielding and ventilation are similar to other equivalent facilities in the nuclear industry; they are the major means of personnel protection in the hot cells.

There are other locations where some radioactive elements will be present, like the low level radwaste building, the cryoplants and the cooling towers. These are not mentioned in this preliminary review as their operation is expected to yield very low doses for workers.

5. Assessment of the radiation fields and of the potential contaminations

During the operation of the fusion reactor, the dose rates (Iida *et al.*, 2004) inside the vacuum vessel are about 10^8 to 10^9 Gy/h, and about 10 Gy/h inside the cryostat behind the VV external wall. 10^6 seconds (12 days) after shutdown, the dose rates decrease to 630 Gy/h inside the VV. The dose rates at the contact of plasma facing component, blankets and divertor cassettes is in the vicinity of 10^3 Gy/h 30 days after shutdown and a few hundreds of Gy/h one year after shutdown.

The consequences of these dose rates are that:

- any direct human intervention in the VV and on the plasma facing components is impossible. For maintenance purposes a system of remote handling is necessary, for intervention in the VV, transport of Plasma Facing Components from the VV to the hot cells and work inside the hot cells;
- shielding is necessary to allow admissible dose rates in the buildings where human intervention will be allowed; shielding is also necessary to avoid nuclear heating in the superconducting magnets, but this is out of the scope of this paper.

5.1. ALARA guidelines

The standards and guidelines for radiological safety developed for ITER are based upon international standards and recommendations from IAEA (2002) and ICRP.

TABLE I
Project guidelines for doses from Occupational Exposure.
Valeurs guide du projet pour l'exposition aux rayonnements ionisants.

Project guidelines	
Annual individual worker dose	10 mSv/y
Individual dose for any single shift	0.5 mSv/shift
Collective annual worker dose	500 mSv/y
Maximum individual worker dose after an incident	10 mSv
Interim ALARA thresholds (trigger for 2nd iteration)	
dose rates	100 μSv/hr
collective dose to operate a system for a year	> 5 % of total collective dose
collective dose for a task performed less than annually	> 5 % of total collective dose

An ALARA assessment process (ITER, 2006) has been established for operations and work yielding the higher doses. This iterative ALARA process provides an opportunity for changes in the design and operational approach; it has been consistently used in the design as an instrument to optimize shielding and maintenance procedures. This process has led to reduced project guidelines and reductions in the anticipated Occupational Radiological Exposure for high risk systems.

The steps in the ALARA process for ITER are as follows:

1. the starting point is to identify the hazards and systems in the plant posing the greatest risks;
2. the second step is to identify ALARA assessment guidelines to help determine where effort and resources should be focused to reduce occupational exposure and risk;
3. in the third step guidelines are applied to individual. Those exceeding guidelines require analysis to improve occupational exposure and risks.

As a starting point, shielding has been calculated to allow working conditions in accordance with the frequency of access. In areas where access should be frequent, like the galleries, shielding has been designed for a residual dose rate of 10 μSv/h with all port cell doors closed. In the areas where access is occasional and space constraints very high, such as the port cells, the shielding has been designed for a residual dose rate of 100 μSv/h. In addition for the operation on a complex system, a collective dose of 30 p mSv has been considered as a trigger for a more refined ALARA study and a possible design improvement. It is recognized that is arbitrary by a factor of two to three.

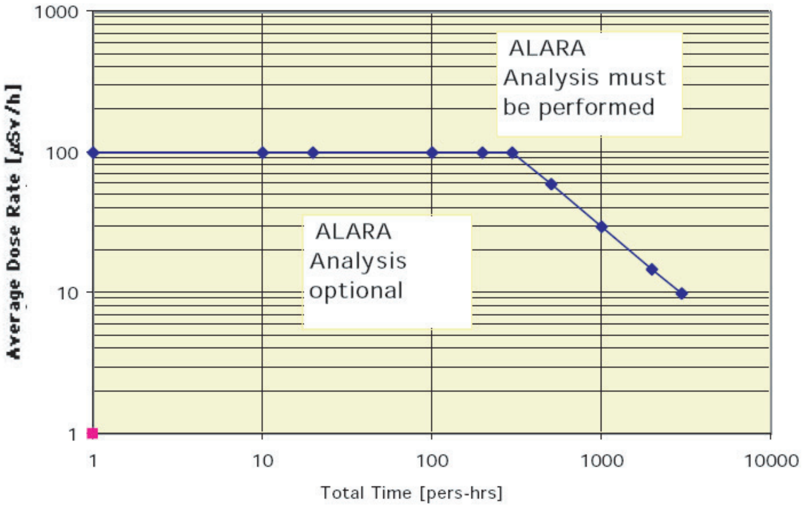


Figure 3 – ***Summary of ALARA guidelines in ITER.***
Résumé des valeurs guide pour la démarche ALARA dans ITER.

The whole approach is summarized by Figure 3, with triggering limits on the dose rates of 100 µSv/h, (the horizontal line) and 30 p mSv (the decreasing straight line).

Therefore, in the first two steps the ALARA process has been mainly considered as a system of dose constraints. In the next and third step of occupational exposure assessment, a formal ALARA study including an optimisation procedure should be performed for the operations yielding more than 5% of the collective dose as summarised in Table II.

5.2. Shielding and time lag

The main protections against neutron and gamma radiations are provided by the shielding of the Tokamak and the bioshield. During operations the gallery area and the cryostat are restricted as a reason of radiation fields and magnetic fields. After shutdown the gallery may be accessed, as well as the port areas. Since many of the activated materials have a short half life, an efficient way to reduce the worker doses is to wait. The optimal duration of this time between shutdown and the start of hands on operation is estimated to be 10^6 s (12 days).

The diagrams below (Fig. 4) show the evolution of the dose rates as a function of the distance from the torus axis both in operation and 10^6 s after shutdown.

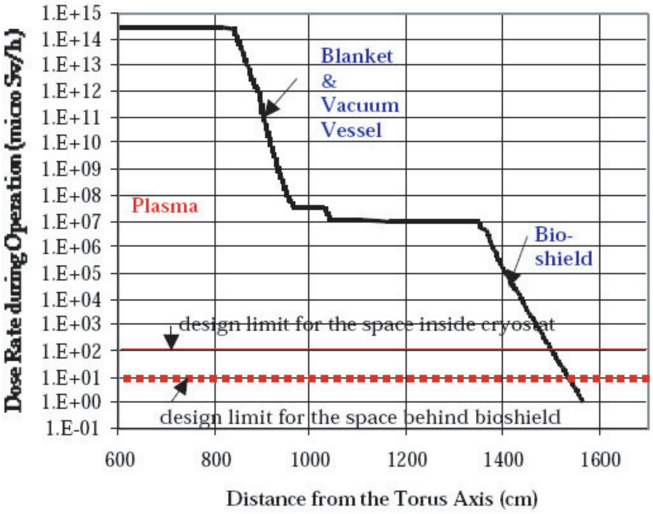
TABLE II
ITER Occupational Radiological Exposure assessment in 2006.
Estimation 2006 des doses collectives prévisionnelles dans ITER.

System and/or operation	Total (p-mSv)	Percentage of objective
Diagnostics	84.5	16.90
Cooling Water System	54.9	10.98
Remote Handling (Rh) Equipment	40	8.00
Hot Cell Processing And Waste Treatment	40	8.00
Electron Cyclotron Heating And Current Drive (EC H&Cd) System	38.7	7.74
Test Blankets	22.4	4.48
Blanket System	18.9	3.78
NBI	16.5	3.30
Vacuum Pumping & Leak Detection Systems	11.5	2.30
Ion Cyclotron Heating And Current Drive (Ic H&Cd) System	10	2.00
LH&CD	6.3	1.26
Divertor	6.1	1.22
Operational waste treatment	4.9	0.98
Tritium Plant	2.1	0.42
Machine Inspection and leak chase	tbd	
Fuelling and Wall conditioning	tbd	
Cryostat	tbd	
Cryoplant	tbd	
Coil power supply & distribution	tbd	
Facilities maintenance (liquid, gas, sampling, …)	tbd	
Total	**356.8**	**71.4**
Margin to the objective of 500 mSv	**143.2**	**28.6**

Tbd: To be determined.

The curve is obtained with the code ANISN, a 1-D modelisation, and using the FENDL2 library (Iida *et al.*, 2004).

The shielding is mainly composed of two parts; blankets and shields that are between the two shells of the vacuum vessel, and which function is mainly to reduce nuclear heating of the superconducting magnets, and activation in the cryostat, and the bioshield which function is to protect the workers in the galleries.



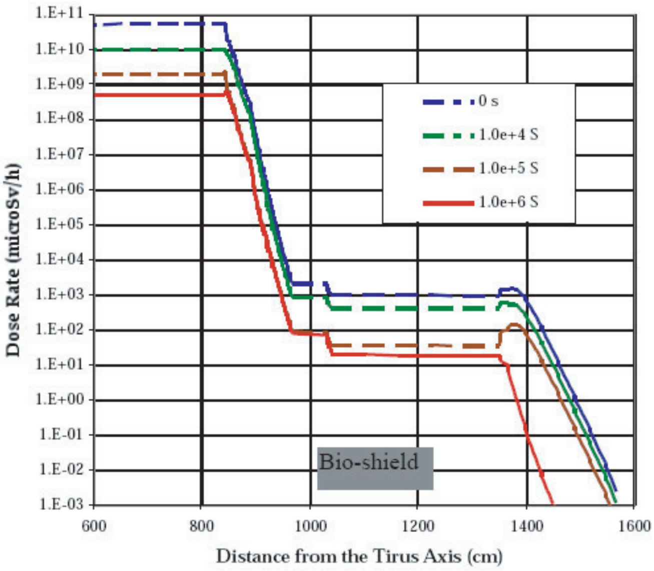


Figure 4 – ***Dose rates from the torus axis in operation and after shutdown.***
Débits de dose en fonction de la distance depuis l'axe du tore en opérations et après arrêt.

TABLE III

Dose rates in the ports areas for the worker protection calculations.
Débits de dose dans les ports pour les calculs de radioprotection.

Port and system	Dose rates at 2 weeks after shutdown (μSv/h)	Reference
Upper ports	5	(Natalizio and Porfiri, 2005)
Electron Cyclotron Heating ports	100	(ITER, 2004a)
Equatorial ports	100-300	(ITER, 2004a)
Ion Cyclotron and Radio Frequency	50	(ITER, 2004a)
Neutral Beam cell at mezzanine level	10	(ITER, 2004a)
Electron Cyclotron Heating	30	(ITER, 2004a)
Lower Hybrid Heating & Current Drive	10	(ITER, 2004a)
Remote Handling ports	20	(Iida *et al.*, 2004)
Divertor ports	60-230	(Iida *et al.*, 2004)
Pumping ports	40	(Iida *et al.*, 2004)
Remote handling ports	60	(Iida *et al.*, 2004)

During operation access to the galleries areas, and the cryostat is restricted, therefore it is of no immediate concern for radiological protection. 10^6 s after shutdown, corresponding to the decay of ^{56}Mn, the dose rates out of the bioshield, which is 2 m thick and made of borated concrete, except in the Neutral Beam cell, are below 10 μSv/h. The bulk shield calculations yield dose rates below 0.1 μSv/h but higher dose rates are anticipated due to neutron streaming through some penetrations and ACP in cooling pipes. The dose rate between cryostat and bioshield is consistently less than 100 μSv/h.

In order to reduce radiation risk exposure, it was decide to use for ITER stainless steel with a low cobalt contents, less than 0.05%. The dominant metal in the cryostat is stainless steel, and therefore this choice also reduces the specific activity of wastes at dismantling.

The shielding around the Tokamak is not uniform. There are singular areas such as the port cells and the NB cell. In the port cells, the dose rates may vary greatly according to the exact location, whether it is at the primary closure plates where they may reach hundreds of μSv/h, at the cryostat level, a little less than 100 μSv/h, or around the ports, a few tens of μSv/h. In the Neutral Beam cell, the dose rates range from 100 μSv/h to 900 μSv/h around the port extension. However, the dose rates at the exact locations where human intervention is needed are lower that the maximum values in the port areas.

The ACP calculations have been performed with the PACTITER code (ITER, 2004b). Although working activities on the Tokamak port cells are not foreseen before 12 days after shutdown, work in the TCWS could take place 5 days after plasma shut down. For the main components, pumps, pressurizer, heat exchangers, valves, pipes, filters, etc. the dose rates have been calculated at a 30 cm and 1 m distance. 5 days after plasma shutdown, the maximum found is 60 µSv/h at a 30 cm distance from the heat exchangers.

5.3. Tritium control in the atmosphere

Tritium is present in the ITER plant due to two reasons: it is provided as fuel and it is produced by nuclear reactions in water and other materials. Originally tritium is therefore located inside the components of the fuel cycle system and of the cooling systems. The containment of tritium is designed to reduce as much as possible the contamination of the building surfaces and atmosphere. However, the phenomena of diffusion and absorbtion/outgassing from materials make it possible finding tritium everywhere beyond the primary barrier and also in the cooling system vaults. The ventilation facility relies mainly on two systems. The first is called HVAC (Heating, Ventilation and Air Conditioning); it is a once through system providing air renewal, adequate temperature and moisture and subatmospheric pressure. Another system, ADS/VDS (Air Detritiation System / Vent Detritiation System) provides depression in the event of HVAC failure and detritiation to any and all compartments with contamination above the action level, 10^8 Bq/m^3, defined to reduce the releases to the environment before shifting to detritiation systems. The containment of tritium and the ventilation systems should keep the airborne tritium level below the monitoring devices detection level. If in one room the tritium concentration in air exceeds one VDO (Derived Operational Value, equivalent to 25 µSv/h) the personnel will evacuate or wear protective equipment. All tritium is conservatively considered in the HTO form. As a result of these precautions, the doses resulting from tritium exposure should be very limited (<1% of objective). This is confirmed by the JET feedback.

5.4. Remote handling

The components inside the vacuum vessel should be removed and transported to and from the hot cells with the help of remote handling. The plasma facing components may be massive, 10 t for one divertor cassette, and the dose rate at 1 m distance may reach 155 Gy/h 12 days after shutdown (ITER, 2004c). The components are transported inside contamination tight casks. However as a reason of available space and admissible weight, the casks will not include a radiation shield. In order to reduce the possibility of worker exposure, the casks will be

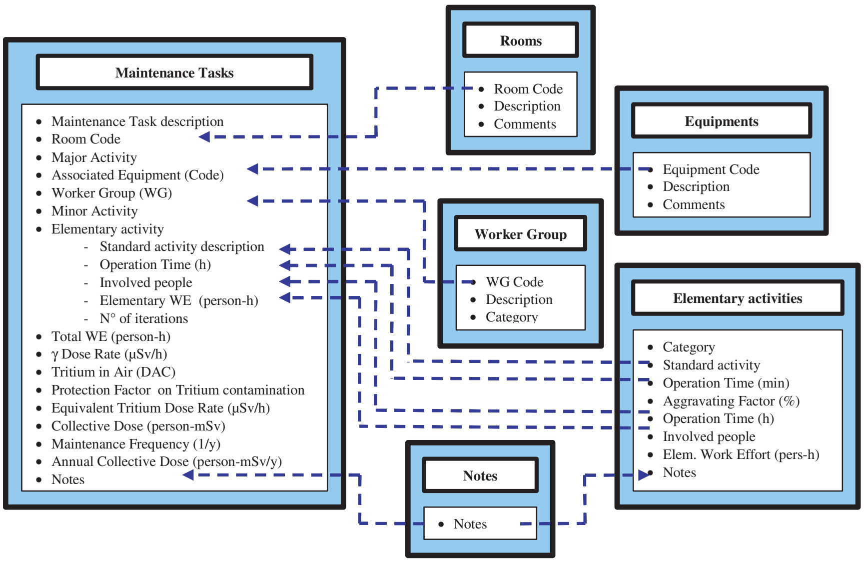


Figure 5 – ***Software tool for occupational radiological exposure assessment.***
Outil informatisé pour les calculs d'exposition radiologique.

transported at night. This procedure seemed to be the optimum balance between the cost of increasing the building size and the reduction of already minute doses. Rescue vehicle and robots will be foreseen also to repair RH equipments which could fail during RH operations and recover incidental conditions.

6. Estimation of collective doses

The method to estimate the collective doses is to establish a list of maintenance tasks and to evaluate the time necessary to complete these tasks. In order to simplify the work and to build a database for future operational radiological exposure evaluation a tool has been developed (Pinna, 2005). An analysis of the work to be performed is made according to following main elements (Fig. 5):

- the rooms and locations where the work is performed, along with their attributes such as dose rates, tritium concentration, protection factors, etc.,
- elementary tasks such as bolting, cutting, welding, etc. with the associated time duration,
- the equipments concerned like blanket module, divertor, fuelling system, neutral beam maintenance, remote handling maintenance, waste processing, etc.

TABLE IV

Evaluation of the collective doses for the replacement of a primary window at upper level (Natalizio and Porfiri, 2005).

Évaluation de la dose collective pour le remplacement d'une fenêtre primaire au niveau supérieur (Natalizio et Porfiri, 2005).

Step	Work effort (p h)	Dose (p µSv)
Clear port cell	170.45	852
Remove bioshield blocks	70.25	7001
Clear port interspace	107.1	10710
Remove and install window	103.5	10350
Reinstall port cell components	125.5	12550
Reinstall bioshield blocks	65.85	6205
Reinstall port cell components	218.45	1092
Total	**861.1**	**48761**

The diagnostic systems generate the greatest contribution to collective doses. As an example of doses evaluation the estimation for the replacement of a primary window is given below. Table IV shows the different elemental activities, the estimated time to perform each of them and the collective dose. The dose rates are indicated in Table III.

As the window replacement frequency was assumed to be once every 4 years, the final contribution of this operation to the annual collective dose is 12.2 p mSv.

Finally, the results of exposure assessment (Massaut, 2006) distributed according to the main systems are shown in Table II and Figure 6. The total dose is estimated to be 357 p mSv, below the 500 p mSv objective. The main contribution comes from the activated components and corrosion products; the contribution from tritium is minor.

These estimations show some consistency with the feedback from JET (Natalizio *et al.*, 2003) shown in Figure 7.

The groups of workers collecting the highest doses in JET are the groups of Maintenance and Repair (MAC) followed by Torus installation, and diagnostic. The dose is due mainly to activated components and to activities inside and around the torus.

The active cooling systems in JET are mainly for the neutral beam and radiofrequency system, so they are very limited compared to ITER. Therefore, the doses linked to the cooling systems in JET and ITER cannot be compared.

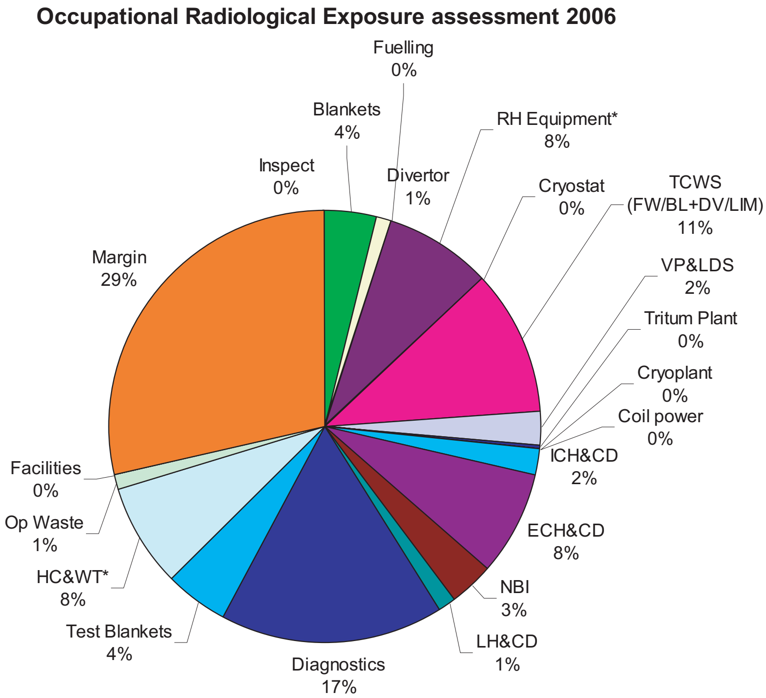


Figure 6 – ***2006 Total Occupational Exposure Assessement for ITER distributed by system.***
Dose collective totale pour l'exploitation d'ITER repartee par système ; évaluation 2006.

7. Future work and conclusion

For the future work, an arbitrary guideline has been chosen to run a formal ALARA study on the operation generating more than 5% of the collective doses. The preliminary calculations show that the operation of diagnostics, cooling water system, maintenance of the remote handling equipment, hot cells and radwaste treatment, and heating and current drive systems are concerned.

In the present state of radiological exposure assessment work effort is less reliable than any other parameter as the design is not finalized. In the future it will be necessary to consider the port areas from a worker perspective in order to minimize the necessary work effort.

This new evaluation will need a refined nuclear analysis to improve the knowledge of the radiation fields. It will also be necessary to study again the necessary hands-on work efforts.

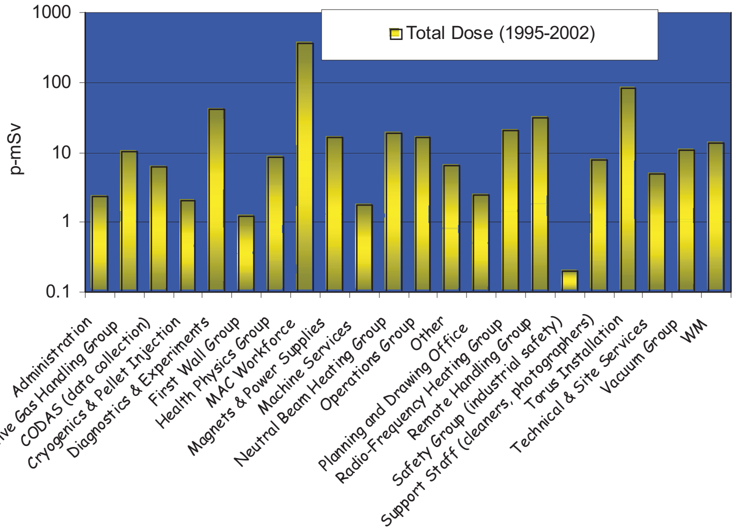


Figure 7 – ***Summary of occupational doses in JET. Years 1995 to 2002.***
Doses collectives par système au JET ; années 1995 à 2002.

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