4. Licensing ITER in France

4.1. Licensing procedure

4.1.1. ITER safety approach

The ITER safety approach adopted by the project draws upon experience in nuclear and non-nuclear industries. Objectives are established for protection of the public, the worker, and the environment. Quantitative criteria are set against which to judge the success in achieving the safety objectives. Hazards are identified that may challenge the objectives. Preventive and mitigative measures are implemented as needed to ensure that facility operation meets the objectives. Techniques of deterministic safety analysis supplemented by those of probabilistic assessment are used to determine and demonstrate the safety of the facility.

Following site selection, the host country regulations will apply which may lead to modification of the requirements or their detailed implementation.

The ITER project safety objectives, principles and criteria are documented in the Plant Design Specification [PDS01] and are summarised below.

- **Safety objectives**

  ITER safety objectives [PDS01] address the potential hazards in ITER from normal operation, accidents and waste:

  - ensure in normal operation that exposure to hazards within the premises is controlled, kept below prescribed limits, and minimised;
  - ensure in normal operation that exposure to hazards due to any discharge of hazardous material from the premises is controlled, kept below prescribed limits, and minimised;
  - prevent accidents with high confidence;
  - ensure that the consequences, if any, of more frequent events are minor and that the likelihood of accidents with higher consequences is low;
  - demonstrate that the consequences from internal accidents are bounded as a result of the favourable safety characteristics of fusion together with appropriate safety approaches so that there may be, according to IAEA guidelines [IAEA96], technical justification for not needing evacuation of the public (external hazards are site dependent, but are considered for a generic site);
  - reduce radioactive waste hazards and volumes.

  In addition, ITER is to be developed in such a way that it can be sited by any participant in the ITER EDA Agreement with minor design modifications.

- **Safety principles**

  The following safety principles are considered in the safety approach. These principles not only provide direction to guide the design, but also include on-going, independent review and assessment to ensure the design will meet the safety objectives.

  - *'As Low As Reasonably Achievable' (ALARA)*

    As a basic principle, exposures to hazards shall be kept as low as reasonably achievable, economic and social factors taken into account.

  - *Defence-in-Depth*

    All activities are subject to overlapping levels of safety provisions so that a failure at one level would be compensated by other provisions. Priority shall be given to preventing accidents. Protection measures shall be implemented in sub-systems as needed to prevent damage to confinement barriers. In addition, measures to mitigate the consequences of postulated accidents shall be provided, including successive or nested barriers for confinement of hazardous materials.

  - *Passive Safety*

    Passive safety shall be given special attention. It is based on natural laws, properties of materials, and internally stored energy. Passive features, in particular minimisation of hazardous inventories, help assure ultimate safety margins.

  - *Consideration of ITER Safety Characteristics*

    The safety approach shall be driven by a deployment of fusion's favourable safety characteristics to the maximum extent feasible. However, the experimental nature of the facility shall also be addressed. A robust, fault-tolerant safety envelope shall be provided to enable flexible experimental usage. In view of the limited operational experience with DT plasmas and hence some plasma physics uncertainties, experimental components will be conservatively designed considering the expected loads from plasma transients so as to reduce the demands on systems which are required for safety. A safety function will not be assigned to experimental components, but faults in these will be considered as expected events in the safety assessments. The experimental programs and related machine modifications and operations shall be developed to take advantage from preceding operations.

  - *Review and Assessment*

    Safety assessments shall be an integral part of the design process and results will be available to assist in the preparation of safety documentation for regulatory applications. These analyses shall comprise normal operation, all categories of accidents, and waste.

- **Safety and environmental criteria**

  Regulatory approval will be required before the construction of ITER, but before site selection, the design will follow international recommendations, in particular technology-independent ones. Guidelines for doses to the staff from radioactivity and releases to the environment are set for the design, construction, operation and decommissioning. These project guidelines follow the recommendations by the ICRP and the IAEA. Following site selection, host country regulations will apply.

  ITER complies with the ICRP recommendations regarding occupational exposure. The radiation protection practices are consistent with the IAEA and ICRP recommendations and make use of best practices. In particular, efforts are made such that exposures during operation, maintenance, modification and decommissioning are as low as reasonably achievable (ALARA).

  The favourable characteristics of ITER can be further demonstrated if, even for hypothetical events that can be postulated, the calculated doses to the local population are below 50 mSv (early dose). This would be below the generic optimised intervention level for temporary evacuation developed by the IAEA [IAEA96] which is 50 mSv avertable dose within a period no more than 1 week.

  ITER safety is described in the Generic Site Safety Report, GSSR. The scope of GSSR is primarily radiological safety of ITER; however, some additional information on conventional hazardous materials and possible environmental issues are included to provide a more complete indication of hazards such as may be required to support localisation efforts.

The following chapters give an overview of the main safety issues that will be addressed in the framework of licensing ITER in France and are based:

- on the one hand, on the information contained in the ITER Generic Site Safety Report (GSSR), which reports the extensive safety studies carried out during the ITER Engineering Design Activities,
- on the other hand, French safety documentation referred at the end of each chapter.

Details of the safety studies made by the ITER team are normally referred to and explicitly recovered when helping for comprehension.

4.1.2. Licensing process in France

The main objective of the licensing process is to obtain authorisation to build and operate a nuclear installation, which is formalised by two decisions at government level: the "Décret d'Autorisation de Création – DAC" (required to start construction) and the "Décret d'Autorisation de Rejets et de Prélèvements d'Eau – DARPE" (required to start operation).

Within French regulations, the licensing procedure is clearly defined for any nuclear installation. The definition of a nuclear installation, called an "Installation Nucléaire de Base (INB)", is based on the inventory of radioactivity. ITER will be classified as INB due to, at least, the expected tritium inventory.

The licensing procedure is based on a continuous technical dialogue between the plant owner, the ITER Legal Entity (ILE) and the Safety Authority, the "Direction de Sûreté des Installations Nucléaires (DSIN)", assisted by technical experts.

The ILE has not yet been defined as formal ITER owner, but it will have to be established before the beginning of the public enquiry, i.e. the last step of the licensing process as described in the following chapter. Practically the ILE will exist from the first steps of ITER construction until the end of the operation. Consequently, CEA will initiate the licensing process prior to the establishment of the ILE.

Concerning waste management and decommissioning an agreement may be signed between CEA and ILE which will have to state clearly that CEA would take care of decommissioning and waste management provided that financial provision will be made by the parties during ITER operation.

There are three sequential steps prior to the signature of the "Décret d'Autorisation de Création" which are briefly described in the following chapter:

- The "Dossier d'Options de Sûreté" (DOS)
- The "Rapport Préliminaire de Sûreté" (RPRS)
- The "Enquête publique"

4.1.3. Roadmap for licensing

1. **The "Dossier d'Options de Sûreté" (DOS)**

   Duration: approximately 1 year.

   This document will be issued by the CEA and it will be assessed by the Safety Authority.

   The DOS is a document, prepared by the plant owner, which defines the major risks and the proposed means to avoid or to mitigate them. It describes the installation briefly, proposes general safety objectives and explains how they have planned to be implemented in the installation.

   The document is being prepared by the CEA, it will be checked by the International Team in order to assure the coherence with reference design and will be sent to the Safety Authority for assessment.

   The DOS is a concise technical document that may refer to more detailed supporting documents, which do not formally belong to the DOS itself. Although certain points are not mentioned explicitly in the DOS, in particular the supply, transport and control of tritium, some of their implications will nevertheless be addressed (e.g. accidental tritium releases).

   An advice on this report by the Safety Authority is expected in the middle of 2002.

2. **The "Rapport Préliminaire de Sûreté" (RPRS)**

   Duration: approximately 2 years.

The second step of the licensing procedure is the preparation and the assessment of the "Rapport Préliminaire de Sûreté, RPRS". The RPRS, or preliminary safety report, consists of a detailed description and a comprehensive safety analysis of the plant. The RPRS is examined by the technical services of the Safety Authority (the "Institut de Protection et de Sûreté Nucléaire" – IPSN). During this phase the Safety Authority usually asks advice to a "Groupe Permanent", i.e. an advisory board at national level.

The RPRS will be prepared under EFDA auspices but will have to be submitted by the owner, the ILE.

In parallel, ITER project may be submitted to a "Débat Public" (**Public Debate**), as it is stated in a recent French law (loi Barnier, 95/101). Any new large project or installation may be submitted to a public debate during its elaboration phase. The objective of the debate is to launch an overall country discussion on the project socio-economic or environmental advantages and drawback. It should focus on the **objectives and main characteristics of the projects** and must be organised **during their elaboration phase**. An independent commission is in charge of surveying this process, which cannot last more than six months. CEA is exploring how and when it may be initiated.

In parallel to the preparation of the RPRS, the various administrative procedures required to obtain the licenses (DAC and DARPE) should be started. The documents required to do so will refer extensively to documents prepared in support of the RPRS (design justification study, accident analysis study, environmental impact study, etc.). It will be necessary to clarify which parts of these documents only require the re-writing of existing reference documents and which parts require substantial new work (e.g. the Seismic, Fire and Waste studies).

3. The "Enquête Publique"

Duration: approximately 2 years.

After the first positive advice on the RPRS by the Safety Authority, the Public Enquiry Procedure can be launched. It is a consultation process limited to the local communities (located within a radius of 10-15 km from the proposed plant site) on the external effects resulting from the construction and operation of the plant.

The construction can only start after a positive opinion from the Public Enquiry Commissioner. It is, nevertheless, possible to anticipate a positive decision and to start site preparation work and the procurement of components that do not have any safety function. In the case of ITER, it should be possible to order the magnet superconducting cable, which is on the construction schedule critical path.

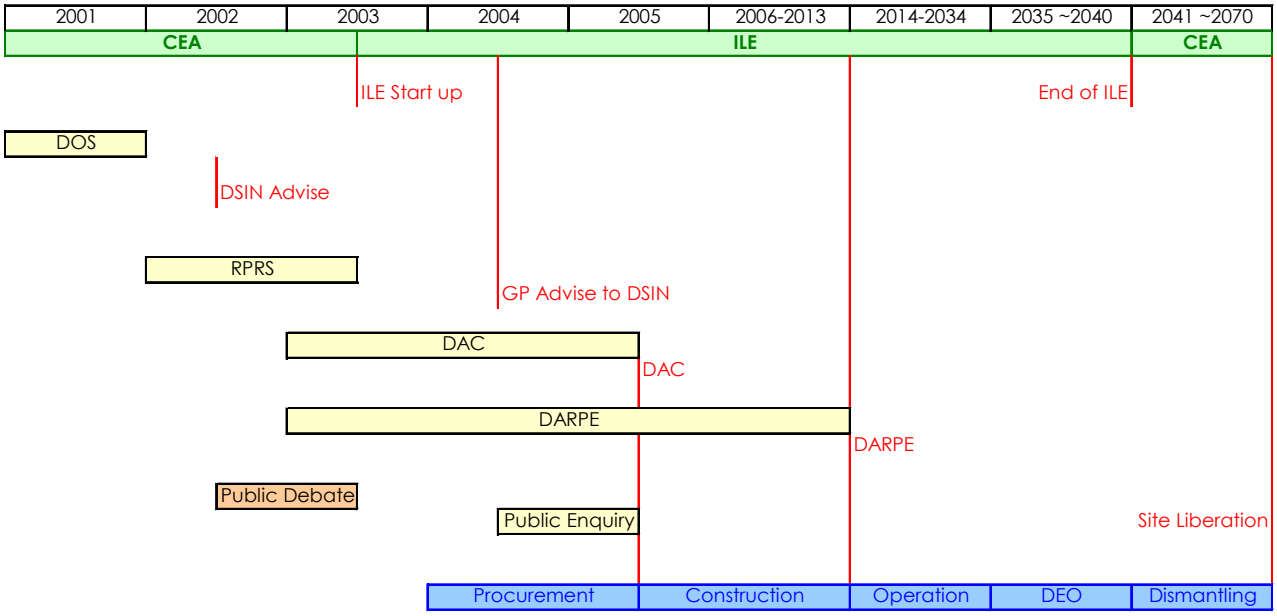
Time Schedule

The overall time schedule for ITER preparation and construction could be as follows:

- The legal implementation, which has started officially mid of November 2000 in Europe, with the decision taken by the European Council of research Ministers to delegate the European Commission to negotiate the "ITER Legal Entity" with the partners.

- In the meantime, administrative procedures will start to prepare all legal documentation, in view of the official approval, which should allow the beginning of the construction phase.
- From the technical point of view, some anticipation can be envisaged, in particular for the construction of the components, which are on the initial path, as the superconducting coils.
- The excavation can also be anticipated to the official "décret".

The regulatory process is described in the Table 4.1.



Acronym	*French meaning*	*English Translation or Meaning*
ASN	Autorité de Sûreté Nucléaire	French Safety Nuclear Authority
DAC	Demande d'Autorisation de Création	Request of Authorisation of Creation
DAC	Décret d'Autorisation de Création	Decree of Authorisation of Creation
DARPE	Demande d'Autorisation de Rejet et de Prélèvement d'Eau	Request of Authorisation for Water Release and Extraction
DARPE	Décret d'Autorisation de Rejet et de Prélèvement d'Eau	Decree of Authorisation for Water Release and Extraction
DEO	Mise à l'Arrêt Définitif (MAD)	Definitive End of Operation
DIN	Division des Installations Nucléaires	In charge of Nuclear Installation Control
DOS	Dossier d'Options de Sûreté	Description of Safety Objectives (not compulsory)
DSIN	Direction de la Sûreté des Installations Nucléaires	ASN Legal Head
GP	Groupe Permanent	Permanent Group (in charge of Request Examination)
ILE	Entité Juridique ITER	ITER Legal Entity (status to be defined)
IPSN	Institut de Protection et de Sûreté Nucléaire	ASN technical Adviser
RPRS	Rapport Préliminaire de Sûreté	Initial Safety Report (compulsory, to be borne by ILE)

Table 4.1: Overall Licensing Process, from Today until Site Liberation

4.2. Key elements for the "Dossier d'Options de Sûreté"

4.2.1. DOS main elements

The main objective of the "Dossier d'Options de Sûreté", DOS, is to present the terms of reference of safety approach when applied to ITER in the site of Cadarache.

The ITER Generic Site Safety Rapport, GSSR, has been taken as the base document in order to specify the safety functions and the envisaged way to implement them. This approach will give the assurance of controlling hazards related to Cadarache site conditions in accordance with French regulations.

The safety approach in the DOS is the following:

- Definition of the safety objectives,
- Identification of consequent safety functions,
- Definition of the technical safety options which will maintain the safety functions.

In particular, objectives on effluents and liquid releases are defined for normal, incidental and accidental cases. In addition a preliminary evaluation of the impact of these releases on the populations has been made and it will be verified that they are under the project limits. Besides, it must be shown that no cliff effect will evolve from an off-dimension situation.

This report will also present a preliminary strategy for tritium transport, waste management and effluent releases produced during machine operation as well as the foreseen steps for decommissioning and dismantling of the installation at the end of life, and the issues regarding different waste generated during dismantling.

The following chapters give an overview of the main safety issues that will be addressed in the DOS.

4.2.2. Preliminary safety analysis

The radiological source terms that could potentially present an occupational or public hazard, and the energy sources that might be available to drive an accident sequence are largely independent of site characteristics. These are reviewed first (section 4.2.2.1). Then the occupational safety analysis in GSSR is assessed against the likely requirements for French licensing (section 4.2.2.2), followed by a review of analyses of potential accident hazards (section 4.2.2.3). These are all based on internally initiated events, whereas external events are more site-specific, so the particular characteristics of Cadarache are outlined in reference to regulatory requirements (section 4.2.2.4).

4.2.2.1. Radiological and energy source terms

Radioactive sources, chemically toxic or reactive substances, and stored energy sources to be used or handled in ITER are given in Volume III of the GSSR. They include:

- Radioactive materials (tritium, neutron activated products in structure and in-vessel dust, and neutron activated corrosion products in coolant water);
- Non-nuclear hazardous materials – chemically toxic (e.g. beryllium) or reactive (e.g. hydrogen);
- Stored energy sources (plasma, magnets, activation decay heat, coolant thermal energy, chemical energy).

Each of these has been quantified in the ITER design and are summarised bellow:

- **Tritium**

After a phase of H-H and D-D operation, ITER will be operated with D-T fuelling. Tritium will be supplied from off-site sources. Tritium transport, inventory tracking and accountancy have to be done following French regulation. A container to transport 25 g of tritium has been internationally qualified. Alternatively a container with a larger capacity could be designed and qualified on the basis of an existing cask. A specific accountability has to be done in France for non-proliferation means. The total site inventory is less than 3 kg. A particular French regulation concerning the storage of tritium does not exist. A periodic inventory assessment will be done, in order to minimise machine uncertainties and to respect administrative limits of tritium inventory in the torus. For the tritium processing, detailed studies of the uncertainties in the measurement are underway. Tritium is located inside the vacuum vessel, in the fuel cycle processes, in the long storage zone inside the tritium building and in the hot cell and waste controlled buildings.

- **Dust**

Dust will be generated during plasma pulses inside the vacuum vessel by erosion of plasma facing components. Consequently those materials, beryllium, carbon and tungsten will accumulate inside the vacuum vessel. Analyses and extrapolation from experiments suggest that between 100 and 200 kg of dust are expected after 500 plasma pulses. The dust inventory must be limited with regard to chemical reactivity (Be, C, W) and activation hazards (W). Methods for extracting them have been developed for previous devices (JET) and will be investigated for ITER. The potential of dust mobility and radioactive inventory have been detailed in GSSR vol. III and have been taken into account for impact calculations in normal and abnormal operations following ITER guidelines.

- **Activated Corrosion Products (ACP)**

ACPs are located in the different coolant loops. As they impact on safety, an assessment methodology has been set-up, which is based on code development and associated experiments. As for dust radioactive resulting inventory is given in GSSR vol. III and has been used for calculation of impact on the public.

- **Conventional (non-nuclear) Hazardous Materials**

The most important chemical sources of energy described in GSSR and of concern from the French regulation point of view are:

- Hydrogen present in the different systems connected with the tritium plant.
- Exothermic reactions between beryllium or tungsten with air or steam inside the vacuum chamber at high temperature.

  These reactions can also drive to hydrogen production, from which a potential risk of explosion in case of infringement of the conditions of flammability / explosion in the air during the incident can follow. Concerning the explosions of dusts, the maximal mass of dust admitted inside the vacuum chamber (free volume of 1350 m^3) has been defined to protect itself.

- Ozone production inside the cryostat in case of an air leak with a potential of release of chemical energy from a concentration 4 % molar.

- **Stored Energy Sources**

  The stored energy source terms are related to the installation activity: fusion power, plasma, magnetic, decay heat, chemical and thermal coolant energy.

In conclusion, it should be pointed that main radiological and energy sources are well described in GSSR taking into account some uncertainties, due to the design conception phase of the project and the fact that ITER is an experimental device that will itself answer to a lot of today estimations and extrapolation. Nevertheless the solid basis of the project allows making a clear assessment of the needs required for licensing ITER in Cadarache.

4.2.2.2. Occupational safety

This chapter is focused on a regulatory perspective. Consequently the French Regulatory Regime, and more specifically the CEA regulatory regime at Cadarache have been taken into consideration in the following and are commented and compared with ITER safety approach. Then an ITER radiation protection assessment is presented. Finally conclusions are presented.

Cadarache Regulatory Regime

State regulations and CEA guidelines regulate French nuclear installations.

State regulations specify that in France worker radiation protection is based on three fundamental principles:

- **Dose justification**: requires that each nuclear facility or device has sufficient benefit to society to justify the radiation exposure that will arise from its use.
- **Dose optimisation**: requires that the radiation exposure resulting from the use of a nuclear facility or device be optimised such that the number of persons exposed and the level of exposure will be maintained As Low As Reasonably Achievable (ALARA), social and economic factors taken into consideration.
- **Dose limitation**: requires that regulatory limits for worker doses be established to ensure a consistent level of protection and to eliminate the risk of overexposure.

The requirement for worker radiation protection is enshrined in the French Regulations and applies to all French nuclear facilities [1], [2]. These are the existing regulations based on the recommendations of ICRP-26 [3] and conforming to Euratom Directive 80/836 [4]. The Regulations contain the dose limits and the fundamental requirements to maintain doses as low as reasonably achievable (ALARA), social and economic factors taken into consideration. The current regulations are being updated [5] to comply with ICRP-60 [6] and Euratom Directive 96/29 [7]. They will be issued in the near future.

The radiation zoning that is applied in nuclear facilities is given in the Table 4.2[13]:

[13] Where SPR stands for "Service de Protection contre les Rayonnements", or the "Radiation Protection Division".
DAC: Derived Air concentration

	BLUE	GREEN	YELLOW	ORANGE	RED
Time Spent		Permanent (8 h/day)	Controlled/ monitored	Controlled/ monitored	Controlled/ monitored
Dose rate threshold (µSv/h)	2.5	7.5	25	2,000	100,000
Max allowable Air Concentration (DAC)	0.3	1	80	4,000	
Dose Limit (effective dose)	15 mSv/a	50 mSv/a & 30 mSv/q	50 mSv/a & 30 mSv/q	50 mSv/a & 30 mSv/q	50 mSv/a & 30 mSv/q
Access Requirement			Written authorization from SPR	Written authorization from SPR	Written authorization from Centre Director

Table 4.2: CEA Radiation Protection Zoning

CEA Guidelines

CEA Direction Centrale de la Sécurité (DCS) issues guidelines that are more restrictive, to ensure that the Regulations will always be satisfied. The CEA guideline for maximum exposure of a radiation worker is 15 mSv per year and that for the average dose for all exposed workers is 2.5 mSv per year [8]. Such guidelines are issued periodically in the form of three-year plans for worker safety improvement. DCS-34 is the most recent three-year improvement plan issued by CEA. Besides, an ALARA optimisation has to be done at any time.

ITER Radiation Protection Approach

The ITER radiation protection framework is described in the ITER Generic Site Safety Report (GSSR), Volume VI. Project guideline for annual individual worker doses is 5 mSv.

The ITER project has proposed to satisfy the dose optimisation requirement, during the design stage, by a prioritised ALARA process. In the GSSR ALARA is a three-step process:

- The starting point of the ALARA process considers the ITER facility as a whole. At this stage of the design, the emphasis is on identifying the hazards and the systems posing the greatest risks.
- The second step is to identify ALARA assessment guidelines to help determine where effort and resources should be focused. Three so-called "ALARA guidelines" are defined:
    - if a system or activity is anticipated to require an individual to exceed 0.5 mSv/shift, then the system/activity is modified;
    - if an activity is performed in a dose rate > 100 µSv/h, then more detailed assessment is required;
    - if the collective dose for a system operation or maintenance activities is estimated to exceed 30 person·mSv, more detailed assessment is needed.

- In the third step, the guidelines are applied to the individual systems. Those exceeding the guidelines are designated as requiring analysis to find areas for improvement to reduce risk and exposure.

During design and operation phase, this iterative process will continue in reviewing the systems, and optimising them, reducing the ALARA thresholds.

The ITER project proposal for radiation zoning is presented in Table 4.3, partially reproduced from the ITER Safety Report (Vol. VI) and the Plant Design Specifications; it includes a system of radiation access control that has a direct correspondence with the system of radiation zoning.

Access Zone (Area Classification)	Access Limitations	Airborne Total Dose Rate Area Contamination Characteristics
Zone A	Unlimited Access	No airborne contamination Dose rate < 0.5 µSv/h WHITE
Zone B	Limited for NRW[14]	Unlimited for RW[15] Total dose rate (internal + external) < 10 µSv/h GREEN
Zone C	Limited Access for all workers	< 100 DAC < 1 mSv/h AMBER
Zone D	Restricted access areas	Airborne > 100 DAC or external dose rate > 1 mSv/h RED

Table 4.3: Area Classifications and Radiation Access Zones

Assessment of the occupational doses in many parts of the ITER plant and in different types of maintenance operations are being developed and continued work on this is still necessary. The list of systems, structures and components described in the GSSR Vol. II is proposed as a tool to be applied for the worker dose exposure calculation and a dose assessment report preparation.

In conclusion, for occupational safety in routine operation and maintenance, the ITER project has adopted dose guidelines that are in line with international practice. These are also generally in accordance with French regulatory requirements. The principle of maintaining doses below these guidelines and as low as reasonably achievable (ALARA) is a project approach that is also a regulatory requirement. At the present stage in the project the implementation of the ALARA process should continue and extend to all the systems. A radiation zoning and access control approach is adopted in ITER that is compatible with French regulations.

[14] NRW: Non Radiation Workers
[15] RW: Radiation Workers

4.2.2.3. Hazards analysis

The following chapters review the approach used in the ITER project to the identification of hazards and protection against them, the identification of postulated accidents and the analysis of selected reference events, and the assurance that radiological guidelines are complied with. This is based on the material presented in the Generic Site Safety Report (GSSR), in particular volumes VII, VIII, X and XI.

Identification and Control of Hazards

ITER processes and plant areas have been systematically reviewed to identify potential hazards (see section 4.2.3.1). From this analysis, protective measures or safety functions have been identified and assigned to implementing systems and components.

The inventories are restricted to the minimum in order to control hazards. The most fundamental safety function is the confinement of hazardous materials.

Identification of Reference Events

Techniques of deterministic safety analysis, used to determine a fully representative set of reference events, were supplemented by techniques of probabilistic assessment used to check its comprehensiveness. The selection of the reference events covers release of the main energy sources (plasma, coolant, magnet); decay heat removal, chemical reactions (hydrogen, ozone) and potential failures in the confinement (in-vessel and ex-vessel coolant leaks) are investigated. All plausible initiating events were systematically identified to provide a justification that the selected sequences are indeed the most severe ones of their category and the Postulated Initiating Events (PIE) family.

To help ensure that all aspects of plant operation have been considered, two complementary and fundamentally different approaches have been applied to identify potential initiators: these are the component-level (bottom-up) and the global (top-down) approaches. The concluding part of both the bottom-up and top-down studies is to show that identified accident sequences have been addressed by analyses that show that potential consequences are within acceptable guidelines.

Analysis of Reference Events

Reference events have been demonstrated in GSSR vol. X to cover all major systems; the radioactive inventories distributed amongst these systems and initiator types that have the potential to cause releases.

The postulated initiating events (PIE) and sequences of subsequent failures are categorised as Incidents or Accidents, and their consequences are evaluated and compared with safety objectives and release guidelines. Each reference event starts with a postulated initiating event, adds all consequential failures, and assesses the complete sequence up to environmental release. In most events, the loss of off site and others aggravating failures are postulated. The main safety functions and

associated safeguard systems are identified. The whole sequence of events is described including the actuation (time + logic) of safeguard systems.

Parametric studies of hypothetical sequences are used to investigate the ultimate safety margins of ITER and to demonstrate its robustness in terms of safety with regard to the project's objectives and radiological guidelines. Key notions in the context of hypothetical sequences are:

- Avoidance of "cliff edge effects"
- "No-evacuation" to demonstrate the safety and environmental potential of fusion power
- Robustness of defence-in-depth.

Two accidents are discussed hereafter:

- *Within the reference events, the large divertor ex-vessel coolant leak which leads to the highest amount of tritium release*.

  The large divertor ex-vessel coolant leak investigated is initiated by a hypothetical double-ended pipe break in a divertor primary cooling loop inside Tokamak Cooling Water System (TCWS) vault at fusion nominal power (500 MW). This is also assumed to lead to an induced in-vessel break. The main safety concerns are the bypass of first safety barrier (vacuum vessel) and the pressurisation of the TCWS vault and the vacuum vessel that is likely to challenge the containment integrity. The sequence includes aggravating event (loss of off-site power).

  The main safety functions and associated safety systems involved allow limiting the consequences of the accidents. The total environmental releases, taking into account uncertainty on vacuum vessel tritium inventory, are 1.5 g-T, 0.07 g-dust and 0.6 g-ACP.

- *The tritium event illustrates the method to investigate the ultimate safety margins of ITER: a reference event is considered postulating an additional failure of the safety function*.

  The reference event is a failure of the tritium fuelling line, postulating coincident failures of both primary and secondary tubing. DT gas (the amount over the atmospheric pressure (> 100 kPa)) is released.

  Due to safety systems the tritium release is relatively limited compared to the sequence mentioned above: 0.17 g. Thus the mitigation of the consequences of this accident strongly relies on the efficiency of the confinement.

  In case of a postulated failure of the safety system elements (confinement), leading to a hypothetical event, the mobilised tritium inventory (13.4 g) is released.

Radiological Releases

The consequences of each reference event have been calculated in terms of environmental releases for the three main hazards (tritium, dust and activated

corrosion product), for several exhaust heights and different meteorological conditions, with or without ingestion. The results can be compared to the French guidelines to check the acceptability of the scenarios. In comparison to the release guidelines adopted by the ITER project itself, all calculated consequences fall well within the allowable range.

Safety Models and Codes

The analyses of transients and consequences arising from the reference events use a number of computer codes as well as analytical techniques. Information is given for each code:

- About its applicability, limitations, availability and physical modelling approaches and simplifications,
- Its verification and validation status.

Some verification and validation work is still underway aimed at validating the codes on fusion-specific analytical tests.

A main goal of ITER is to demonstrate from the viewpoint of safety the attractiveness of fusion. To accomplish this goal, ITER safety needs to address the full range of hazards and minimise exposure to these. The safety approach draws upon experience in nuclear and non-nuclear industries.

In conclusion, the analysis of ITER implements general and recognised safety principles. Reference events have been identified and analysed. The whole set of results obtained, although some of them may need to be deepened, leads to a robust safety demonstration. Verification and validation work is still in progress on the numerical tools that are used for the assessment of the accidental sequences.

4.2.2.4. External hazards

To assess the possible construction and operation of ITER in Cadarache, it is necessary to consider the totality of external hazards for which it would have to be designed. This section considers the different hazards, taking account the site characteristics, in relation to the requirements of the French safety rules.

The documents which have been used in following are prescriptions relative to the site [9], main safety rules "Règles Fondamentales de Sûreté" [10-14], calculation rules and unified technical documents "Règles de Calcul et Documents Techniques Unifiés" [15-17], ITER Technical Basis for the final Design Report and GSSR vol. IX.

The external hazards that must be taken into account in the building design have been considered and compared with ITER design assumptions where they have been specified. They are reviewed in the following:

Wind Actions

The study of wind actions on a construction has to consider prescriptions defined in the calculation rule [16]. Wind actions on a building depend on:

- The speed of the wind or base dynamic pressure;
- The category of the construction and its total dimensions;
- The position of the element considered in the construction and its orientation towards the wind (height effect, site effect, mask effect);
- Dimensions of the element considered.

The speed parameter or base dynamic pressure is defined according to the geographical implantation of the project. The speed of the wind is 35 m/s, **but to be compared with ITER design assumption of 38.9 m/s**, then the base dynamic pressure is 750 Pa. Furthermore, the verification of resistance and stability conditions of a construction under wind actions has also to envisage in calculations an extreme dynamic pressure; the ratio of that here to the basis dynamic pressure is taken equal to 1.75 increasing thus the pressure to 1310 Pa. Concerning the other parameters listed above that can amplify or decrease the base dynamic pressure; they can be defined only in a precise analysis of the project regarding the exact building geometry and their implantation with respect to each other.

Snow Actions

Snow actions studied on a construction have to follow prescriptions defined in rules NV65 and N84 referred in [16] and [17]. Actions of snow on a building depend on:

- The base normal overload;
- The altitude;
- Characteristics of the roof;
- The concomitance of effects of snow and wind.

The parameter base normal overload is defined according to the geographical project implantation. The basis normal overload is 350 Pa. Considering the altitude of the implantation site (310 m), the normal overload is then 460 Pa. Furthermore, the verification of resistance and stability conditions of a construction towards the actions of snow has to envisage in calculations an extreme overload that increases, in regard to the altitude of the site, to 780 Pa. The design assumption value given for ITER is about 1500 Pa. Concerning the other parameters listed above that can amplify or decrease the basis overload; they can be defined only by carrying out a detailed analysis of the project regarding the exact building geometry.

Aircraft Crash Risks

Risks linked to aircraft crash have to be studied with prescriptions of the RFS I.2.a referenced [10]. The RFS defines three aircraft families:

- General aviation;
- Commercial aviation;
- Military aviation.

For each site, an evaluation of the probability of crash on each target is undertaken for each of the three aircraft families above. For this evaluation, formulations introduce crash statistics according to the aircraft family, the nature of the aerial space, the phase of flight and the virtual surface notion of the buildings housing each safety function. Furthermore, the RFS fixes a limiting probability of occurrence of the event of 10^{-7} per year.

Concerning the Cadarache site, the "Présentation Générale de la Sûreté de l'établissement", PGSE, referenced [9], gives information in impact probability of the aircraft on buildings on this site that are the following:

- general aviation $1.6 \cdot 10^{-10}$ per year / m²
- commercial aviation $1.3 \cdot 10^{-12}$ per year / m²
- military aviation $1.2 \cdot 10^{-11}$ per year / m²

Regarding comments in the RFS applied to nuclear plants, whose geometrical characteristics are comparable to those of installations in the ITER project, it appears that only probabilities of aircraft crash of the general aviation reach values higher than 10^{-7} per year.

Installations must be then calculated to resist to their impact. It is admitted to consider two aircraft types considered representative of the different categories of aircraft of general aviation:

- CESSNA 210 single-motor;
- LEAR JET 23 twin-motor.

Buildings to be justified are defined by an analysis of safety undertaken in accordance with the RFS [10]. Walls protected by another wall designed to withstand the aircraft crash are not subject to this action. Regarding the RCC-G section 1.3.2.3-3 [15], characteristics of reference aircraft are as given in Table 4.4. Justifications can be led by using an equivalent static effort such as specified in the section 1.3.2.3-3 of the RCC-G [15]. This has been pointed out in GSSR vol. IX and detailed assessment will be needed.

	LEAR JET	CESSNA
Speed of impact	100 m/s	100 m/s
Aircraft mass	5.7 t	1.5 t
Impact surface of the aircraft	12 m² (1.2 x 10)	4 m² (0.5 x 8)
Motor mass	./.	200 kg
Impact surface of the motor	./.	0.5 m²

Table 4.4: Characteristics of Reference Aircrafts

Seismic Risks

Seismic studies in the framework of the ITER project have to consider prescriptions and methodologies defined in the RFS I.2.c [11] and V.2.g [14]. The reference spectra can be found in Chapter 3.3. Two studies have been done showing that the design assumption of 0.2 g can be complied with without major building modifications.

Industrial Risks

The principles are to take account of those risks specified in the RFS I.2.d [12]. The RFS presents three family sources of the potential hazards:

- Permanent industrial installations such as storage and units of production;
- Piping of transportation such as pipelines;
- Road, railway, fluvial and maritime.

Considering the geographical implantation of the ITER project, only the road is considered as a potential hazard source. Furthermore, the RFS states that risks to be studied are those linked to the following hazards categories:

- The wave of aerial pressure due to an explosion;
- The increase of temperature due to an external fire;
- The seismic wave associated with an explosion;
- Toxic or corrosive gas clouds, gases and consequent smoke of a fire.

For external explosions Accordingly to the RFS, the convention is to assume as the minimal plausible hazard a wave of aerial overpressure incident with triangular form and with a stiff front that comes from any horizontal direction. It presents the following characteristics:

- Maximum overpressure 0.05 bar
- Duration of overpressure 300 ms

The load to take into account on buildings considers phenomena of reflection and focalisation according to section 1.3.2.3-2 of the RCC-G [15].

For external fires, regarding the prescriptions of the RFS, there is no conventional load to consider. This external hazard has then to be studied in a particular safety study.

For seismic wave associated with an explosion, it is noted that dispositions of construction taken against seismic consequences in application of the RFS 1.2.c and V.2.g referenced [11] and [14] are in general estimated sufficient for the protection against the seismic wave effects due to explosions.

For gas emissions, there is no conventional load to consider. This external hazard has then to be studied in a particular safety study.

Flood Risks

The principles to take into account are specified in the RFS I.2.e referenced [12]. Regarding the expected geographical implantation of the project, this risk is not envisaged.

References

[1] Decree n° 86-1103

[2] Decree n° 88-662

[3] ICRP-26

[4] Directive 80/836

[5] Decree 98-1185

[6] ICRP-60

[7] Euratom Directive 96/29

[8] DCS-34

[9] Centre d'études de Cadarache – Présentation générale de la sûreté de l'établissement – 12/1994

[10] RFS I.2.a Prise en compte des risques liés aux chutes d'avions (Aircraft Crash Risks)

[11] RFS I.2.c Détermination des mouvements sismiques à prendre en compte pour la sûreté des installations (Seismic Inputs)

[12] RFS I.2.d Prise en compte des risques liés à l'environnement industriel et aux voies de communication (Industrial Environment and Roads Risks)

[13] RFS I.2.e Prise en compte d'un risque d'inondation d'origine externe (External Flood Risks)

[14] RFS V.2.g Calculs sismiques des ouvrages du génie civil (Seismic Calculations of Structures)

[15] RCC-G Ed. 1988 Règles de conception et de construction du génie civil des îlots nucléaires REP – Tome I – Conception (Conception of Nuclear Plant)

[16] DTU P 06-002 Règles NV 65 modifiées 1999 et annexes définissant les effets de la neige et du vent sur les constructions (Snow and Wind)

[17] DTU P 06-006 Règles N 84 modifiées 2000 – Actions de la neige sur les constructions (Snow)

In conclusion, the bases of climatic actions (snow and wind) considered in the ITER design studies are conservative towards the rules and applicable conditions of the Cadarache site. Concerning aircraft crash and industrial risks, in the framework of a safety analysis, targets to be protected must be identified and, if necessary, actions characterised that are not evaluated in RFS. The building will be constructed to avoid any radioactive release in case of Cessna or Lear Jet crash. The low inventory of mobilisable radioactive material is a key factor to reduce radioactive release in any accidental event. Finally, the seismic studies carried out in the framework of the design of the tokamak and tritium buildings have highlighted the need for an analysis of the seismic behaviour according to prescriptions and regulations in force in France for this type of building.

4.2.3. Effluents and Releases

The purpose of this chapter is to assess ITER effluents and releases of and look for an optimisation with the hypothesis of a construction at Cadarache on the basis of a comprehensive ALARA process. After reviewing general aspects on the implementation of the ALARA principle for optimising the effluent management systems of ITER, the strategy in the ITER design for effluents is described. Then the impact on the public exposures of effluents for Cadarache site conditions is detailed. Finally an assessment on optimisation of effluent management systems is given.

This chapter will refer to the individual dose limit for the members of the public. Following ICPR-60 [1] and Euratom Directive 96/29 [2], the following principles apply:

- in the context of optimisation, all exposures shall be kept As Low As Reasonably Achievable (ALARA), economic and social factors taken into account;
- the sum of the doses from all relevant practices shall not exceed the dose limits laid down separately for exposed workers and members of the public.

The application of the optimisation principle is defined in the French regulation [3]. As concerns the individual dose limit for the members of the public, the common practice for nuclear power plants in France employs a dose limit of the order of 10 µSv per year for normal operation. In the consideration of licensing ITER under French conditions, the design target for the effluent management is that consequences should be minimised for normal operation, maintenance and accidents. The studies show in particular that, for all foreseeable accidents, the release level will be far below the one for which any countermeasures or food restrictions are needed for the population at the site boundary.

4.2.3.1. General aspects on the implementation of the ALARA principle for optimising the effluent management systems of ITER

For the releases brought about by normal operation and planned maintenance activities, the ITER design process applies Project Release Guidelines (see Plant Design Specification, PDS). In the evaluations presented in the Generic Site Safety Report (GSSR, vol. IV) it has been shown that these guidelines have been met for the different types of effluents.

The application of the optimisation principle implies continuous efforts to be made to reduce the releases and the impact they may cause. The design of ITER has gradually evolved building on the experience at similar existing research facilities, such as JET. The adopted design process is in principle fully compatible with the general requirement of optimisation. The material contained in the present GSSR emphasises the most recent design features rather than providing a comparable description of the previous design variations.

4.2.3.2. Strategy and options for the effluents and releases management in the ITER design (GSSR vol. IV)

The effluents and releases from ITER dealt with here are gases, airborne particulates and liquids that are released during normal operation and maintenance from the ITER facility (see section 4.2.3.1) and which contain substances that could have a

detrimental environmental impact. The objective is to keep these releases to at least below the permitted levels and in general to levels considerably less than these.

Gases and Airborne Particulates

The plan for ITER is to have a single discharge point for all gases arising from the operational areas of ITER. At present this discharge point is at the level of the highest point of the ITER buildings, i.e. the roof of the main tokamak building. The exact position and height is site dependent and will be optimised taking into account the local dispersion conditions and the wake effect of ITER buildings and possible other nearby facilities. The gases will be entrained in the exhausts from all the Heating, Ventilation and Air Conditioning systems (HVAC), which are routed to this single discharge point.

This discharge point will be monitored both for control purposes and compliance monitoring, that is both to indicate to operators what is being discharged (control monitoring) and to measure what has been discharged for reporting to the relevant environmental agency (compliance monitoring). Gases that are expected to be discharged are tritium in various chemical forms and activated air products. It is possible for airborne radioactive and chemically toxic particulates to be released from this discharge point.

Methods to minimise Releases

Particulates: These will be minimised by filtering of all air streams of the HVAC systems before discharge using standard filtering techniques. These filters have a high efficiency only allowing some fine particulates to pass through, which will be monitored at the discharge point. Good housekeeping techniques in the relevant areas (i.e. Hot Cell) will assist in reducing the source of particulates.

Tritium: ITER is to be equipped with an extensive set of Atmosphere Detritiation Systems (ADS) to remove as much tritium as possible from gaseous streams that arise from normal operations, maintenance procedures and accident situations. The design of these systems is closely linked to the design of the confinement arrangements and HVAC systems. The concept of all these systems is well proven in tritium handling facilities such as JET or the Tritium Laboratory Karlsruhe TLK and basically consists of converting elemental tritium (HT) or hydrocarbons such as tritiated methane (CH2T2) into tritiated water vapour (HTO) and carbon dioxide by catalysts and then collecting the HTO on molecular sieve beds. Any pre-existing HTO will also be collected this way. These molecular sieve beds will be regenerated either routinely or as required and the resulting water will be collected and handled according to the level of tritium. The way tritiated water is handled will be described later. Each ADS has a high detritiation factor to reduce the tritium releases to the environment to very low levels.

Design of the ADS

The number of ADS systems and the parameters of each individual ADS have been calculated according to the estimated amount of tritium it will have to deal with, atmosphere flow rates, duty (normal, standby or emergency) and other relevant factors, such a the physical location of the system (i.e. Hot Cell, Tritium Building). The design produced for the complete system has been assessed and

found to be satisfactory. Certain refinements are suggested, such as standardisation of major components and cross-linking of systems to add greater flexibility and to expand the possibilities for dealing with abnormal situations. These should be assessed and incorporated during the detailed design stages.

Liquid Effluents

Liquid effluents from ITER will primarily be contaminated water. The type and level of contaminants will determine how they are managed. Tritium contaminated water arises from various sources:

Regeneration of the Molecular Sieve Beds of the ADS Systems

Water arising from the regeneration of molecular sieve beds in ADS systems will be measured for tritium content and then transferred to one of four water collection tanks depending on the level of tritium. The water detritiation system WDS will process water from these tanks on a campaign basis. There is no liquid waste product from the WDS system. The WDS produces hydrogen and oxygen gas. The tritium-depleted hydrogen fraction is passed through a flame arrester and then discharged to stack. The oxygen is discharged through an ADS after processing in a drier bed. The tritium-enriched hydrogen is fed into the Isotope Separation System (ISS) of the Tritium Plant.

Releases from the Tokamak Water-Cooling Systems due to Leakage, Maintenance, Accidents

Any releases from the tokamak cooling water system (TCWS) will be collected in a drain system installed in the tokamak building. This water will be tested and either sent to site waste or reused back into the TCWS after processing.

In addition, potentially tritium contaminated water could arise from:

Condensation within the different HVAC Systems

Water condensed by local atmosphere coolers and by the HVAC systems will only be tritium contaminated under abnormal conditions. This water is collected in tanks containing the lowest category of tritium concentration and is considered for direct discharge after assaying. Experience at tritium processing facilities such as JET or TLK has proven the validity of the above statements.

Showers/Laundry

It is expected that the water arising from these activities will have very low levels of contamination and will be discharged to the site waste system. However, this water will be collected and assayed before any discharge is initiated.

Trade Waste including Sewage

It is possible for the trade wastes to contain trace levels of tritium and it is expected that this will be dealt with according to site requirements and practices.

4.2.3.3. Evaluation of the public exposures caused by the normal operational effluents

In the GSSR, estimates of the gaseous and liquid releases into the environment have been presented and compared to the ITER project guideline releases. However, the licensing process of the ITER facility in France requires site-specific evaluation of the arising radiation exposures via different pathways, including the ingestion doses from the consumption of locally grown foodstuffs.

Gaseous Releases

In the safety and licensing considerations for the Cadarache site no specific estimation of the gaseous releases was carried out. Consequently **a conservative case of 1 g of tritium as HT and 0.1 g as HTO and 1 g of activation products AP (SS316), and 5 g of activated corrosion products (ACP)** was taken as the source-term for a global radiation dose evaluation in normal conditions. The different isotopes activities have been deduced from GSSR data in vol. III and vol. IV and the total annual releases for different radio nuclides corresponding to the chosen source term have been taken as input for the site-specific dose calculations. Population distribution around Cadarache, weather conditions, food habits and others parameters defined for Cadarache site have been used.

The annual effective doses for an adult in the closest habitation (Château) near Cadarache have been estimated to be less than 2.5 µSv per year after 10 years of continuous D-T operation. In all the calculated cases, for the releases in normal operation the individual dose for the adults, the children and the babies for all the studied groups is lower than10 µSv per year. For tritium the ingestion pathway is dominant, whereas for activation products (AP) and activated corrosion products (ACP) the dose is dominated by external exposure from deposited activity as it is shown in the Figure 4.1.

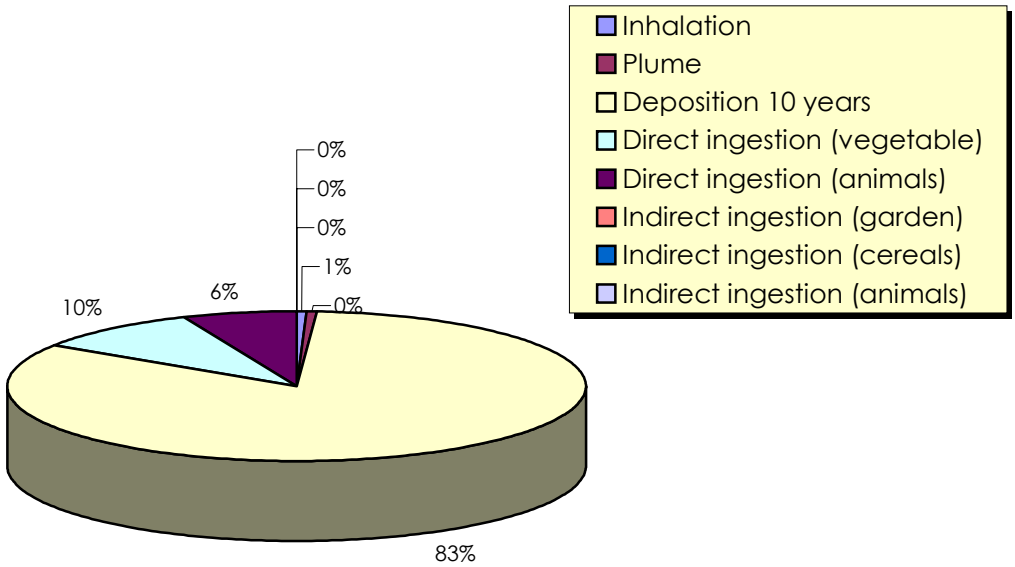


Figure 4.1: Contribution to dose due to gaseous release pathways, at the Cadarache Château after 10 years of exposure due to ITER normal operational effluents

Liquid Releases

The source term for the annual liquid releases are taken from table IV.3.1-1 of GSSR IV. This corresponds to the total estimated values of liquid releases: 0.14 TBq per year tritium and 0.77 g per year of ACP. Based on this source term the total annual releases for different radionuclides corresponding to the estimated liquid releases are taken from GSSR vol. III for the dose impact calculation. For the Cadarache site the maximum annual effective dose after 50 years was calculated to be about 0.1 µSv per year in the nearest village (Saint-Paul lez Durance).

4.2.3.4. Assessment of optimisation of effluent management systems

Evaluation of the current status of the design

In the current ITER design, the process followed has taken into account the current international radiation protection requirements. For the optimisation of effluents from ITER operations the ALARA process has involved the following steps (GSSR Vol. IV):

- Design approaches were surveyed at other fusion devices as TFTR and JET and tritium plants, with wide operating experience in control of releases. Analysis of these good practices and equipment used for effluent control in these facilities has benefited the design of the proposed effluent management of ITER.
- In addition, for normal operations, CANDU reactors have extensive experience in control of tritium releases, and the ITER design utilises this experience in establishing design specifications for many components.
- Establishment of stringent project effluent guidelines that are sufficiently conservative to be acceptable by most national regulations.
- Estimation of possible effluents. A systematic review of systems, activities and pathways has been undertaken to estimate releases and to ensure that all release pathways are accounted for and the obtained estimates for releases are lower than the project guidelines.
- In the design these release pathways are analysed to identify the main contributions and possibilities of design improvements for further reduction of the releases. The release reduction process is carried out in parallel with the general design evolution.

The last mentioned step is a key element in the application of the ALARA principle. Two main targets of current application of ALARA to reduce releases from normal operations are:

- Design improvement or changes to reduce process effluents;
- Design improvement for maintenance releases.

The ALARA process allows a reasonable guarantee that, at the current design status of ITER, the radiological and environmental impacts of the effluents will comply with the requirements established for the Cadarache site, and that they will be ALARA.

In addition, the process of reviewing the design is continuing, with the goal to incorporate or improve new equipment, which may lead to a greater optimisation of the effluent management.

The following conclusions can be drawn concerning the optimisation of gaseous and liquid effluent management:

- The different ADS systems as designed and planned for ITER will accomplish their objective of ensuring that tritium releases to the environment are within design guidelines for normal operations, maintenance and design basis accident scenarios. This is accomplished by the application of established techniques derived from experience from existing plants.
- The liquid effluents arising from ITER operations will be handled according to the concentration of tritium. The objectives will be to recover as much tritium as possible from these liquids for reuse in the ITER fuel cycle and simultaneously reduce the tritium concentrations in the other product streams to very low levels. These other product streams, either gaseous or liquid, will only be discharged to the environment if they comply with the site requirements for both concentration and total discharge.

Evaluation of the Detailed Design

In a detailed engineering study of the ITER project, a thorough review of the design using ALARA criteria should be carried out. Following the usual procedures in this field, a manual for the entire facility, which should include a checklist of ALARA criteria to comply within specific areas, should be prepared.

Even though a wide operating experience of most systems and equipment have been used for the treatment and discharge of the effluents (including a high degree of standardisation), this detailed ALARA review would allow to guarantee that all ALARA aspects have been taken into consideration. This review is of special interest given the unique nature of the facility.

Evaluation of Machine Operation

Operation of the ITER device comprises different phases (H-H, D-D and D-T) with a gradual increase of the effluents to be discharged. The treatment systems have been designed to handle all the effluents produced during all of these phases of operation being base on expected conditions at the end of nominal life.

This progressive increase in the production of effluents will permit an assessment of their impact on the environment and of the corresponding adaptation of the facility, should it be required, for the incorporation of new equipment or improvement of existing ones.

References

[1] ICRP-60

[2] Euratom Directive 96/29

[3] Decree 98-1185

The values of effluents and releases, which are expected according to the GSSR evaluation, are well below internationally accepted levels and are compatible with the current authorisation of the Cadarache centre.

The assessment has shown that the best available means have been used in the design for minimising the effluents. However, the installation of future nuclear facilities in the Cadarache Centre has to be taken into account.

Preliminary evaluation of doses to the public from atmospheric release is low, less than 10 µSv per year at the end of life of ITER.

Concerning the tritium concentration in water, limits are driven by the fact that water from the Durance and the EDF Canal are used as drinking water.

4.2.4. Waste management and decommissioning

The purpose of this chapter is to give an overview of the radioactive waste from ITER, regarding the management, streams and characteristics. The waste is both from operational activities as well as the decommissioning phase. ITER is a completely new machine regarding construction and performance; therefore the base of experience is very limited for the estimation of amount of waste and level of activity in the waste. In the development of design, analyses of materials and preparatory assessments have been performed with a focus to minimise the radioactive waste. This work was already started from the previous concept of the 1998 ITER design. The material selected for different components are with present knowledge and technology the best, from technical performance during operation and radiological waste points of view. In the following the waste classification is presented (4.2.3.1), then the possible repositories (4.2.3.2), and finally the dismantling operations foreseen (4.2.3.3). References are given at the end of this chapter.

4.2.4.1. Waste classification: methods and assessments

French waste classification criteria

In GSSR vol. V, it is said that "*work and results reported in this volume largely address the concept of clearance, a process recommended by the IAEA for releasing solid materials with very low level radioactivity from regulatory control and assuring that the consequences to the public would be trivial. Natural background radioactivity provides a reference level for man-made radioactivity. Relative to this reference, doses around 10 µSv per year are considered insignificant and have been used by IAEA for quantifying the concept of unconditional clearance*."

Since the concept of "clearance" cannot be applied in France, it is necessary to review all the wastes and to define which repository has to be adopted in each case according to French regulation criteria.

In France, the radioactive waste is classified in four categories, depending on their activity and the half live of nuclides:

- **TFA waste**: "Très Faiblement Actifs", very low activation waste,
- **A-Type waste**: Low and Intermediate Waste;
- **B-Type waste**;
- **C-Type waste** (no type C waste in ITER case).

Each of these categories has its own acceptation criteria.

- **TFA Waste**

  Data are taken as working values from official documents under preparation, and have to be used as preliminary guidelines waiting to final approbation.

  - The "Indice Radiologique d'Acceptabilité en Stockage", IRAS (radiological acceptance index in storage) must be less than 1:

$$IRAS < 1, \text{ where} \qquad IRAS = \Sigma A_i/10^{Ci}$$

A_i is the radionuclide activity and Ci its class (from 0 to 3)

- Maximum limits for acceptance related to water transfer (LMAE, Limites Maximales d'Admissibilité liées aux transferts par Eau) must be complied.

- **A-Type Waste**

  The CSA (Centre de Stockage de l'Aube) criteria are used as reference since all shallow land disposals will have the same type of criteria as CSA.

  - Maximum limits for acceptance (LMA, Limites Maximales d'Acceptabilité) must be complied.

- **B-Type Waste**

  All the waste that cannot be stored in A-type waste disposal.

The demonstration that a beryllium threshold is respected during decommissioning processes is required for the acceptability in shallow land disposal of wastes suspected to be polluted by beryllium.

As a first hypothesis, three zones can be defined to identify nuclear and non-nuclear parts of the installation:

- **Zone 1**: zone without radioactivity such as office building.
- **Zone 2**: non-contaminated zone such as bioshield. None contamination is foreseen in this zone in normal operation. In case of contamination in incidental situation, the zone can be decontaminated.
- **Zone 3**: radioactive zone.

An added radioactivity on natural radioactivity in the range of 1 Bq/g is considered as the limit between zone 2 and zone 3.

Review of Waste Streams and Classification

The overall dominating amount of radioactive waste, both regarding masses as well as activity is the decommissioning waste. The operational waste will only be a minor part. Some components are exchanged during lifetime of the installation; the major masses are from the blanket system and the divertor. The activity in the construction material of the machine has been calculated with a 1-D activation model. With these data and the physical data of the components, the activity in each component has then been estimated [1]. This allows having for the different zones used in 1-D calculation and then for each component, the material, the mass and the activity characteristics. The management of wastes containing beryllium is foreseen in a French law (13/07/92) concerning special industrial wastes. The amount of non-active beryllium, produced during ITER construction and during component replacement, has to be assessed.

Operational Waste

The radioactive operational waste will come from activated waste material from normal operation, and will be maintained within controlled areas. The major part of the waste consist of:

- Exchanged components;
- Vacuum pump oils;
- Process residues (waste by-products such as resins, slurries, dusts, etc.);
- Ion exchange resins and filter masses;
- Used or defective tools and equipment;
- Protective clothing;
- Materials from decontamination operations;
- Process (hot) area office trash such as paper, towels, laboratory samples, etc.

The total amount of operational waste from the hot cell and waste treatment facility is estimated to 1,400 ton with a total activity of 60 TBq.

In a first hypothesis, the process waste can be finally disposed as TFA (20 %), type A (75 %) and type B (5 %). A finer estimation of these wastes must be performed.

All the exchanged components (about 750 tons) must be classified as B-type even if their decay time is longer than 30 years. If detritiation processes are performed on only tritiated wastes (tritium plant equipment, fuelling system equipment…), this mass of B-type could be slightly reduced.

Decommissioning Waste

During operation, the tokamak and surrounding equipment will be activated and contaminated, and the activity will remain after final shutdown. The activity decays by time and the total activity in the waste will then be reduced.

Decommissioning will produce about 30000 tons of wastes.

Comments on Tritium Contamination due to Permeation

The steel of in vessel components and vacuum vessel will contain tritium not only from transmutation reaction (activation) but also from permeation from the plasma side (contamination). Blankets and divertor have to be considered as B-type because of transmutation reaction products, so tritium contamination has no major impact on their classification. The tritium from permeation alone increases tritium inventory but would not change the basic judgement that detritiation is not necessary. Concerning vacuum vessel, the tritium contamination does not change classification into A-type.

Tritium plant equipment, fuelling system equipment and remote handling equipment, which are assumed to be only tritiated, will be B-type due to the tritium content and will need to be detritiated if another storage will be foreseen.

The presence of tritium, from activation and permeation, will require specific performances of the packaging to allow a controlled tritium degassing.

Waste management strategy

In order to reduce A- and B-type wastes, processes such as decontamination and detritiation are foreseen. Furthermore an interim storage is considered to allow radioactive decay. If these measures are not applicable or they provide no advantages an evacuation of waste to a repository directly after plasma shutdown is planned.

The Table 4.5 shows the total mass of waste distributed by type at shutdown for operation and decommissioning materials (Hypothesis 1). Hypothesis 2 shows their distribution after decay or specific handling. After 30 years 4924 tons of type A waste can be declassified to TFA and 1 ton of type B to type A. After 50 years 387 tons of type B can be considered type A. After 100 years 4598 type A tons are acceptable as TFA.

Optimisation studies will confirm if further reduction from type A to TFA are beneficial.

	Phases	Masses (tons)			
		TFA	A-Type	B-Type	*Total*
Hypothesis 1	All wastes at shutdown	10577	19492	3502	*33570*
Hypothesis 2	Part at shutdown	10577	9970	3114	*23660*
	Part after 30 years	4924	1		*4925*
	Part at 50 years or detritiation simulation		387		*387*
	Part at 100 years or CWS decontamination simulation	4598			*4598*
	Total	**20099**	**10358**	**3114**	***33570***

Table 4.5: Breakdown of waste at shutdow or after decay

4.2.4.2. Waste management options

Hypotheses on repositories

Hypotheses are made on waste repositories and packaging. Different repositories can be considered for the storage of waste packages:

- Specific industrial waste disposal;
- On-site buffer interim storage waiting for a definitive repository (the date of the interim storage end must be determined at the beginning of storage);
- Industrial dump;
- TFA dedicated repository;
- Shallow land final disposal for type A wastes;
- Cells dedicated to tritiated wastes in shallow land disposal.

Long-term interim storage: the duration is not determined for the moment but should be lower than 100 years. The possible functions of interim storage are B-type

declassification to A-type. From the costing point of view, B-type waste management is the most expensive but for this waste there is no change expected in its classification since the plasma shutdown until the 100 years interim storage availability except for only tritiated wastes.

Deep final disposal: only for wastes that cannot be stored in A-type shallow land disposal and can not be declassified.

Waste Management Strategy

For each type of waste, the following strategy is proposed:

Conventional Waste

These wastes come from civil engineering structure dismantling. Depending on their nature (concrete, steel…), management systems are defined for these types of wastes.

TFA Waste

The wastes are classified in this category after final shutdown or after a decay time at maximum of 30 years, which allow type A waste declassification to TFA. They can be treated using management modes currently developed for this type of waste as buffer interim storage in on-site cells for 30 years at maximum to allow declassification to TFA. Waste waiting for packaging or transport could also use this storage.

The interest of direct storage of large massive part of components must be investigated. The current reference solution for transport to a TFA storage centre is to use ISO containers.

A-Type Waste

Compacting could be used to reduce waste volume. For this type of waste, the waste matrix generally used is a hydraulic binder. The use of large dimension container allows reducing operations on metallic pieces. The reference solution, which is selected for calculation, is a concrete package with 3 m^3 waste volume, external volume of 5 m^3.

B-Type Waste

No declassification of type B waste will be possible after 100 years decay except for only tritiated wastes.

- *Activated and Tritiated Waste (stainless steels, Cu and Ag)*

  The management modes of these wastes must take into account the large dimensions and the weights of the components and the presence of tritium. The principal stake of the packaging is the control of tritium degassing: the external

waste casing must warrant a tritium gas confinement level satisfying the repository requirements. This has to be investigated

The waste packaging must also satisfy:

- Durability,
- Mechanical stability,
- Handling properties.

A concrete packaging with concrete waste matrix can be used. A square base fibre concrete package will be proposed for calculation with 3 m^3 waste volume, external volume of 5 m^3.

For this waste, the envisaged repository is deep disposal, as they cannot be accepted in A-type repository even after long-term interim storage for radioactive decay. Nevertheless, a long-term interim storage could be used before deep disposal to allow tritium decay for the highly tritiated wastes. The facility may need an atmosphere detritiation.

- *Waste with Beryllium and Highly Tritiated*

ITER will use 13 tons of Be in blankets first wall (10 mm thickness). Due to its chemical toxicity, it is necessary to reinforce protection lines. The maximum allowance performance, related to safety constraints, will be put on the packaging. Different points have to be investigated:

- **Specific treatments before packaging**: the possibility of Be recycling have to be studied;
- **Packaging**: specifications for a waste matrix with resistance in particular to lixiviation and a packaging with double confinement barrier (stainless steel internal container and concrete external container for example);
- **Repository**: deep final disposal.

A double barrier package with 3 m^3 waste volume and external volume of 5 m^3 is proposed.

There are also components with tungsten and Carbon-Carbon fibre composite (CFC) in divertor first wall. They will collect tokamak activated dust and will be polluted by Be. They can be pulverulent and for CFC they can contain an important quantity of tritium.

The management of such waste will need:

- **Specific treatments before packaging**: detritiation or Be decontamination for example,
- **Packaging**: the Be container could be used. This allows reducing the typology of the containers. This is the reference solution for calculation,
- **Repository**: deep final disposal.

- *Only Tritiated Waste*

This waste comes essentially from operation and dismantling of the tritium plant, fuelling system and remote handling equipment. The interest of detritiation will be investigated. Nevertheless, a long-term interim storage of about 100 years (8 half live of tritium) can be envisaged if after this period, due to tritium decay, the repository of the waste can be optimised. It is necessary to take into account the management of potential gaseous outlets.

For information, the maximum tritium-degassing rate for packages stored in CSA is currently 2 Bq/g/day.

For this type of waste, storage in specific cells in shallow land disposal (with or without preliminary radioactive decay storage) could be envisaged as definitive repository. Studies are currently performed to determine the acceptability of solid tritiated wastes coming from CEA.

A package with 3 m^3 waste volume and external volume of 5 m^3 is proposed for calculation.

In conclusion, after a first analysis, all the radioactive materials of the ITER machine appear to be manageable.

Nevertheless, two points are sensitive:

- **Mixed wastes (chemical and radiological wastes such as beryllium irradiated wastes) are a key issue. For these wastes, the maximum allowance requirements, related to safety constraints, will be put on the packaging that has to be carefully designed and qualified.**
- **The technical processes used during the operation and deactivation periods, are a second key issue since the classification of waste packages will depend on guarantees (mainly demonstrations on contamination, beryllium level…) provided by the process.**

The following should be investigated:

- Definition of dismantling and cutting processes which guarantee a contamination below threshold for beryllium;
- Interest of detritiation, Be decontamination, compacting…
- Definition of the packaging (controlled tritium effluents and beryllium handling must be guaranteed);
- Study of an in-situ long-term interim storage if a final repository is not available.

4.2.4.3. Decommissioning

Decommissioning deals with the operations from first dismantling to the packaging of produced waste in order to evacuate them to the appropriate repository. This is a preliminary approach to a possible solution.

Several parameters must be considered:

- Dates when wastes reach the allowed levels in the different repositories (particularly mass activity);
- Dose rates at the same key dates, and also at other important dates corresponding to levels to define (sorting by zone: green, yellow… for example);
- Physical access;
- Radiological access (dose rate of the surrounding areas);
- Mechanical availability of parts to be dismantled.

Removed parts must be drummed. In order to perform this packaging, it is necessary to cut to fit to the drums characteristics (size and weight first) and to fit to the repository conditions (dose rate, chemical limitations…). This packaging is performed on the production site. This operation is among the heaviest regarding the needed human works and materials, taking into account the dose rates which can be high and taking into account the characteristics of this kind of packaging (repository and transport regulations are severe). The buildings concerned with the additional activity are tokamak hall, hot cell, radwaste and tritium building.

It is assumed that all other buildings are free of additional activity after phase 1 decommissioning.

The reference solution containers are those described in the previous chapter.

Dismantling Operations

The purpose of the studied dismantling is to reach the third level of the IAEA classification. The three level characteristics are indicated in Table 4.6.

	Level 1	Level 2	Level 3
	Closing and survey	Conditional and partial liberation	Full dismantling
Status of the plant and of the equipment	All barriers in operation Limited access Monitored confinement	Reinforcement of the containment Only one barrier Free access around the containment	Everything is removed[16]
Survey	Heavy	Light	No
Inspections	Often	Scheduled	No
Identified stage	DSD (definitive shutdown)	New INB (basic nuclear installation)	Absence of additional activity

Table 4.6: Levels of Dismantling

[16] Everything with any additional activity is removed. This classification does not deal with the non-additional activity materials. Sorting between elements with additional activity or not is under the responsibility of the site owner.

The main phases established on the basis of the IAEA recommendations and on the basis of the French regulation are:

Phase 1

Phase 1 is under the responsibility of the plant operator. And therefore it is not studied here. Works to be performed during this phase are described in GSSR vol V and Decommissioning Plan.

This phase concerns removal of tritium and removal of in-vessel components (blanket and divertor).

The duration of this phase is about 5 years in the generic ITER decommissioning planning.

Phase 2

According to the general regulation the following items are dismantled:

- What can be dismantled for physical and radiological reasons (man access);
- What can be evacuated to an identified repository (conventional, TFA, type A and type B);
- If possible, dismantling begins with the lowest levels of activity and ends with the highest level.

Gauging should be done immediately to avoid dispersion. It may not be possible to gauge some parts for several reasons and these will be dismantled on a specific area to be created. This area may be the assembly hall with some adaptations (means and tools). Use of the hot cell is not excluded.

The necessary sorting (materials, radiological, destination) must be performed as soon as possible in order to limit pollution, interim storage, and handling. The packaging is performed on another area to be defined. Particularly, it is necessary to allow the needed radiological measures on the closed parcels. The ambient dose rate in the gauging zone may be important. The hypothesis that it is not necessary to perform decontamination is made:

- The main contamination is due to tritium into the vessel, this tritium is removed as good as possible during phase 1;
- The contamination part of the dose rate is low compared to the activation level;
- Repository acceptance criteria for contamination would be respected without decontamination operations;
- Hand cleaning is possible also in order to limit the dispersion.

The creation of a new INB, “Installation Nucléaire de Base”, could be necessary. This stage is not absolutely imposed by the regulations. It could be used to wait for the decay. In the case of ITER in France, waiting does not seem to be useful. Nevertheless packages could have to be removed away. Duration of this phase

will depend on the building flow on the site and on the evacuation flow. Works on the non-nuclear part of the site may be performed in parallel.

Phase 3

The purpose of this phase is to liberate the site, every additional activity materials being removed. The main part of the work is performed during phase 2. At least the dismantling of the parcels interim storage building must be considered during this third phase.

Other Radioactive Data (Equipment outside the Pit)

The systems outside the pit (Water Cooling System and vacuum system) are supposed to be decontaminated during phase 1. As the detritiation performed during phase 1 is supposed to have removed all the tritium, these circuits can be sorted, as TFA or type A. Beryllium pollution is not considered in this document.

A potential problem of the hot cells is the beryllium pollution. The hypothesis is that all this pollution is removed during phase 1. These rooms that are not highly contaminated are sorted as TFA or conventional waste. Their dismantling schedule is independent of the tokamak one.

The radwaste building contamination would only be due to incidents during the operation. The necessary cleaning is assumed to have been done so that the rooms are not sorted as TFA. Phase 1 includes a detritiation of this building, and thus these zones are considered as highly decontaminated. Dismantling of the tritium building is independent of the others buildings.

Dismantling of In-Pit Equipment

The purpose of this study is to develop a feasible scheme for dismantling and sorting equipment located in the pit. Further study and optimisation are needed to complete the actual plans.

The equipment will be cut in some large parts, using remote tools if necessary. Then they will be removed from the pit using the existing heavy crane and placed in a special pool. The special pool is built in the assembly hall. This allows reducing the dose to the personnel, and reducing the contamination dispersion. In this pool, the large parts are cut to fit the chosen drums, using simple remote tools. Regarding the results of the sorting analysis, the dismantling of the large parts is performed following mechanical criteria only. The general schedule of dismantling is to follow the reverse schedule of the assembly. The initial sorting cannot be performed in the pit because of the difficult mechanical access (interlocked structures) and the weight of the components, which exceeds the crane capacity. The pool will be equipped with a circuit that allows filtering and purifying the water. The water is not replaced during the complete dismantling phase. Small parts are handled from the pool in air, to the sewage or drying zone. After sewage or drying, the parts are handled in the packaging zone.

The assembly hall will be adapted to the dismantling operations requirements.

Packaging is done as soon as possible. There will be a buffer zone for drums waiting for the transport. An interim storage of all the drums has not been considered. The packaging will be done regarding the foreseen criteria of acceptance and transport (limits on size and weight for road and rail transport). If an interim storage will be decided, for any reason, the packaging could be delayed but has been done. Packaging should be done as soon as possible. The packaging zone is foreseen to fill the drums, and to perform all post-packaging operations (measurements, identifications, documents…). The packaging zone is installed in the assembly hall, near the pool, but with a physical separation. After packaging, the drums are sealed and handled to the interim storage zone. This zone will also be established on the assembly hall.

Other scenarios

Other solutions have been studied:

- The previous study was based on main guidelines given by the Decommissioning plan (annex DP to the PDD) [5]
- Another solution has been considered which consists, in removing the internal parts of vacuum vessel after a radioactive decay of about 25 years, using a robot and in evacuating waste to the hot cells. The other components are dismantled, cut in the pit and progressively evacuated to the assembling hall by the cryostat access points. Cryostat, vacuum vessel and coils are cut directly at the right size from top to bottom by moving a specific working platform built inside the pit. With this scenario, the duration of decommissioning will be about 12 years.

The best strategy will be chosen after complementary studies.

References

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[2] CD-ROM ITER task D451, subtask 1, 1 D activation calculation of ITER, D.G. Cepraga, G. Cambi, M. Frisoni, ENEA, July 2000

[3] Transport de matières radioactives, réglementation internationale, O. Kimmel, Préventique-Sécurité, n°31, Janv-Fév 1997

[4] Spécifications d'acceptation des colis de déchets radioactifs au Centre de l'Aube (INB n°49) – Spécifications générales – Spécification Technique Générale, ANDRA, ACO SP ASRE 99.001

[5] Decommissioning Plan, Annex DP to the PDD, N 23 FDR 1 01-05-17 W0.1

The main goal of the dismantling is to minimise the type B wastes volumes and weights. Known technology allows performing the sorting.

Depending on the scenario, dismantling could start immediately after phase 1 and could last 16 years or could start 25 years after phase 1 and could last 12 years . Feasible schemes for dismantling of ITER in line with French regulations and practices have been developed.