5 Safety

5.1 Introduction

It is a goal of ITER to demonstrate the safety and environmental potential of fusion and thereby provide a good precedent for the safety of future fusion power reactors. To accomplish this goal, ITER needs to address the full range of hazards and minimise exposure to these. However, it is necessary to account for ITER's experimental nature, the related design and material constraints, and the fact that not all design choices are suited for future fusion power reactors.

With regard to the above goal, the ITER work programme has always included extensive safety and environmental assessments and aimed at a design that can be sited in any of the sponsoring Parties' countries with a minimum of site-specific adaptations. There can be a number of acceptable safety approaches and different choices or emphasis made in implementing a safety approach to meet a particular country's concerns and regulations. This section describes a generic safety approach and its implementation for a generic site. The safety guidelines for the project are based on internationally recognised principles and criteria, most notably the International Commission on Radiological Protection (ICRP) and the International Atomic Energy Agency (IAEA) recommendations. The approach to safety[1] is summarised in the following sections in terms of safety objectives, safety principles, safety and environmental criteria, and the elements of the generic safety approach.

5.2 Safety Approach

5.2.1 Safety Objectives

With regard to the safety of individuals, society and the environment, the potential hazards in ITER from normal operation, off-normal operation and waste are addressed as follows:

(1) ensure in normal operation that exposure to hazards within the premises is controlled, kept below prescribed limits, and minimised;
(2) ensure in normal operation that exposure to hazards due to any hazardous effluents from the premises is controlled, kept below prescribed limits, and minimised;

[1] Plant Design Specification G A0 SP 2

(3) prevent accidents with high confidence;
(4) ensure that the consequences, if any, of more frequent events are minor and that the likelihood of accidents with higher consequences is low;
(5) 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[1], technical justification for not needing evacuation of the public (external hazards are site dependent, but are considered for the generic site);
(6) reduce radioactive waste hazards and volumes.

It is against these objectives that ITER has been assessed as summarised here.

5.2.2 Safety Principles

The principles[2] associated with the safety approach that have guided the design are listed below.

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

(2) Defence-in-depth is used so that all activities are subject to overlapping levels of safety provisions in 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. In addition, measures to mitigate the consequences of postulated accidents shall be provided, including successive or nested barriers for confinement of hazardous materials.

(3) 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.

(4) The safety approach shall be driven by a deployment of ITER's favourable safety characteristics to the maximum extent feasible: the fuel inventory in the plasma is always below 1 g so that the fusion energy content is small; plasma burn is terminated inherently when fuelling is stopped due to the limited confinement by the plasma of energy and particles; the plasma burn is self-limiting with regard to power excursions, excessive fuelling and excessive additional heating; the plasma burn is passively terminated by the ingress of impurities under abnormal conditions; the radioactive decay heat density is low; the energy inventories are relatively low; large heat transfer surfaces and large masses are available as heat sinks; confinement barriers exist and must be leak-tight for operational (non-safety) reasons.

However, the experimental nature of the facility shall also be addressed. A conservative, 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 shall not be assigned to experimental components, but faults in these will be considered as expected events in the safety assessments. The

[1] International Atomic Energy Agency, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115, IAEA, Vienna (1996).
[2] Plant Design Specification G A0 SP 2 00-11-21 W1.1, section 3.2

experimental programs and related machine modifications and operations shall be developed to take advantage of preceding operations.

(5) 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 off-normal operation, and waste.

5.2.3 Safety and Environmental Criteria

Since regulatory approval is required before the construction of ITER, preparations to assist in the future application for approval have been included in the design. In the absence of an actual ITER site, the design has followed international recommendations (ICRP and IAEA), in particular technology-independent ones, such as limits on doses to the public and staff.

The project guidelines for doses from occupational exposure[1] are given in Table 5.2.3-1. The work towards the establishment of best practices is consistent with the ICRP and IAEA recommendations. In particular, efforts have been made to design such that exposures during operation, maintenance, modification and decommissioning are ALARA.

Table 5.2.3-1 Limits and Project Guidelines for Doses from Occupational Exposure

Dose Limits	
ICRP[2] recommended limit for annual individual worker doses	20 mSv averaged over 5 consecutive years not to exceed 50 mSv in any year
Project Guidelines	
Annual individual worker doses	< 5 mSv/year
Individual worker dose for any given shift	< 0.5 mSv/shift

The project has established guidelines for radioactivity releases to the environment[3] given in Table 5.2.3-2. The design has aimed at ensuring margins between calculated values and project guidelines. The favourable characteristics of ITER can be further demonstrated if, even for hypothetical events (see section 5.5.5) 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[4] which is 50 mSv avertable dose within a period no more than 1 week. Following site selection, host country regulations will apply.

[1] Plant Design Specification G A0 SP 2, Table 3-2
[2] 1990 Recommendations of the International Commission on Radiological Protection, ICRP Publication 60, Pergamon Press, 1991.
[3] Plant Design Specification G A0 SP 2, Table 3-1
[4] International Atomic Energy Agency, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115, IAEA, Vienna (1996).

Table 5.2.3-2 Project Release Guidelines

Events or Conditions	Goal	Project Release Guideline
Normal Operation comprising events sequences and plant conditions planned and required for ITER normal operation, including some faults, events or conditions which can occur as a result of the ITER experimental nature.	Reduce releases to levels as low as reasonably achievable but ensure they do not exceed project release guideline for Normal Operation.	< 1 g T as HT and 0.1 g T as HTO and 1 g metal as AP and 5 g metal as ACP per year.
Incidents, comprising deviations from normal operation, event sequences or plant conditions not planned but likely to occur due to failures one or more times during the life of the plant but not including Normal Operation.	Reduce likelihood and magnitude of releases with the aim to prevent releases, but ensure they do not exceed project release guideline for Incidents.	< 1 g T as HT or 0.1 g T as HTO or 1 g metal as AP or 1 g metal as ACP or equivalent combination of these per event.
Accidents, comprising postulated event sequences or conditions not likely to occur during the life of the plant.	Reduce likelihood and magnitude of releases but ensure they do not exceed project release guideline for Accidents.	< 50 g T as HT or 5 g T as HTO or 50 g metal as AP or 50 g metal as ACP or equivalent combination of these per event.

HT: elemental tritium (including DT); HTO: tritium oxide (including DTO); AP: divertor or first-wall activation products; ACP: activated corrosion products.

In the absence of an actual ITER site, activated materials that do not meet the criteria for unconditional clearance following IAEA recommendations[1] after a decay period of up to 100 years are considered long-term waste.

5.2.4 Implementation of the Safety Approach

5.2.4.1 Overview

This section describes the current implementation of the safety design principles and elements of a generic safety approach[2].

The ITER safety approach has been implemented by:

- enhancement of fusion's intrinsic safety characteristics to the maximum extent feasible, which includes a minimisation of the dependence on dedicated 'safety systems';
- conservative design to accommodate uncertainties due to the limited operational experience and data with DT plasmas;
- integration of mitigation systems to enhance safety assurance against potentially hazardous inventories in the facility by deploying well established safety approaches and methodologies tailored for ITER;
- use of a step-by-step approach through progressive stages of operation to further validate safety data and analyses.

ITER is a research facility, and its experimental nature requires a design that permits flexible operation, facilitates experimentation, and can accommodate changes. Changes, for example,

[1] "Clearance Levels for Radionuclides in Solid Materials", International Atomic Energy Agency, IAEA-TECDOC-855, Vienna, 1996.
[2] Plant Design Specification G A0 SP 2, Section 3.4

could include testing of alternative divertor designs, different plasma-facing materials, etc. These needs drive the safety design in three ways:

- provision of a conservative safety envelope;
- minimisation of the safety role and influence of experimental components such as the divertor and blanket;
- minimisation of the safety role and influence of experimenters' equipment such as plasma diagnostics.

Protective measures and safety functions have been identified to ensure protection of the personnel, public and the environment. Design requirements have been derived from the safety principles and release guidelines by identifying the systems, structures, components, and procedural or administrative measures that can prevent or mitigate releases, and by allocating performance targets (both in terms of capability and reliability) to these. The design requirements are iterated and refined as a result of systematic safety analysis.

The elements of the implementation of the ITER safety approach are listed below.

- Design
  - confinement and protection of confinement
  - component classification
  - structural design
  - materials
  - quality assurance
- Ability to withstand events
  - earthquake and other common cause events
  - equipment (environmental) qualification
  - fire protection
- Protection of public and environment in normal operation
  - effluents
  - decommissioning
  - waste
- Safety of workers
  - radiation protection
  - hazardous materials
  - conventional hazards
- Legal and international obligations
  - security and proliferation
- Human Factors
  - organisational aspects, user interfaces, etc.

From a review of the nature of the hazards in ITER, protective measures or safety functions are identified and assigned to implementing systems and components.

<u>Restricting inventories</u> is an effective method to control hazards.

<u>Confinement</u> of hazardous materials is the most fundamental safety function, where confinement refers to all types of physical and functional barriers which provide protection against the spread and release of hazardous materials. Confinement is implemented by:

- sets of successive physical envelopes (including process enclosures, secondary confinement, port cells, containment volume, etc.) around each of the principal source terms (in-vessel, fuel cycle, heat transfer systems, hot cell, and in-vessel components during maintenance),

and

- systems (vacuum vessel pressure suppression system, tokamak vent system, TCWS vault coolers, and normal and standby vent detritiation systems) which provide the functions of pressure control in and removal of radioactive materials from these envelopes.

Releases would most significantly occur upon breach of barriers. Hence, protection of confinement and the following needs to be evaluated, taking into consideration ITER's safety characteristics:.

Control of coolant enthalpy to prevent damage to barriers from overpressure or underpressure.

Control of chemical energy to avoid energy release and pressurisation threats to confinement barriers.

Heat removal to protect against mobilisation of hazardous materials or damage to barriers.

Control of magnetic energy to avoid damage to confinement barriers from mechanical impact, pressurisation or electric arcs in the event of faults.

5.2.4.2 Inventories

The ITER project has set itself challenging guidelines for the maximum amounts of tritium and dust inside the vacuum vessel and the fuel cycle. Further design studies have to show their feasibility. These ambitious guidelines were set to push the design into a direction of minimising inventories. Should further studies show that the guidelines need to be increased to allow practical operation of ITER, this would not invalidate the ITER safety approach since wide margins are built into the confinement of inventories such that predicted releases are significantly below project release guidelines. More details of the radioactive inventories and energies in ITER are given in subsection 5.3.

5.2.4.3 Confinement

ITER protects personnel and the public by using confinement barriers. Successive physical and functional barriers protect against spread and release of hazardous materials. The confinement barriers are as independent from each other and passive where possible with minimal dependence on new components that cannot practically be tested in the appropriate service environment before construction. Every radioactive inventory is contained in its vessel, process piping, component, etc. which serves as the first confinement barrier. This confinement barrier is designed to have high reliability to prevent releases. Another barrier is provided, usually close to the first one, to:

- protect personnel and limit the spread of contamination from leaks;
- mitigate consequences in the event of failure of the first barrier to assist in meeting project release guidelines.

The design maximises structural and spatial separation and independence from the first confinement barrier to prevent a common failure mode of both barriers. Exhaust from rooms that can be contaminated is treated by filters and/or detritiation systems and is monitored.

Inventories of tritium or activated materials reside within the vacuum vessel, in the fuel cycle (vacuum pumping, tritium plant, fuelling), within the hot cell, and in the tokamak cooling water system (TCWS). The confinement approach for accidents for each of these is illustrated in Figure 5.2.4-1 and described below.

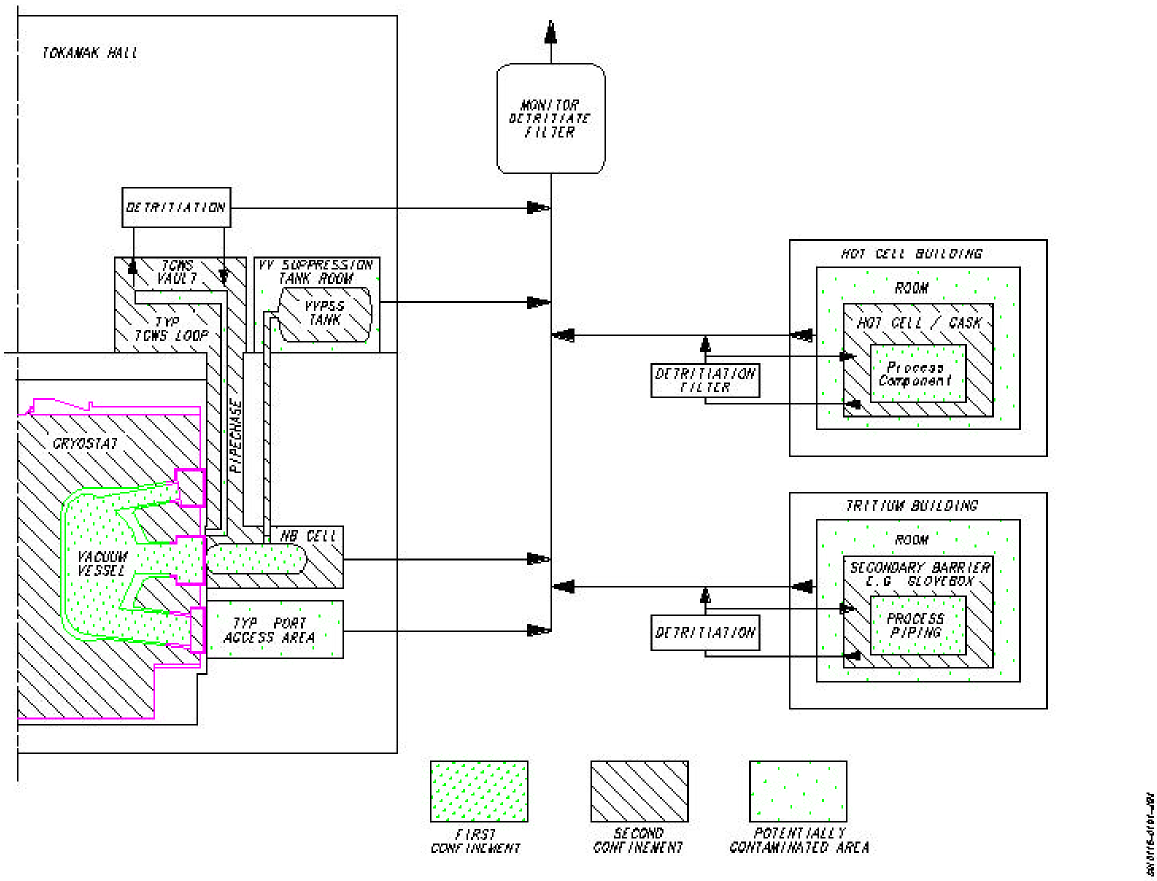

Figure 5.2.4-1 Schematic of Confinement Approach for Accidents Illustrating the Successive Confinement Barriers that are Available

The primary confinement for the source term within the vacuum vessel is the vacuum vessel and its extensions including the neutral beam (NB) injector vessels and confinement barriers in the radio-frequency (RF) heating systems and diagnostics. The TCWS piping forms the primary confinement for tritium and activated corrosion products in the loops of its constituent primary heat transfer systems (PHTSs). Since experimental components inside the vacuum vessel are not assigned a safety function, the TCWS must also confine the in-vessel source term for events such as an in-vessel coolant spill. The vacuum vessel pressure suppression system (VVPSS) reduces peak pressures inside the vacuum vessel if there is an ingress of coolant, and the parts of the VVPSS that are external to the cryostat would then also form part of the confinement barrier.

For the in-vessel and TCWS source terms, another barrier is available. Penetration confinement barriers or windows, typically at the cryostat closure plate, in the RF heating systems and diagnostics form part of this barrier. The TCWS vault, the area where the CVCSs are, the TCWS vault extension, the NB cell, the vertical pipe shafts, the upper and lower pipe chases and the lower pipe sump, collectively referred to as the containment volume, are designed to be leak tight and withstand over-pressures following coolant spills. Exhaust from the containment volume, port access areas and gallery areas, etc. that can be potentially contaminated (e.g. due to accidental leakage past confinement barriers or during maintenance) can be treated by filters and detritiation systems and is directed to the monitored plant exhaust.

The stainless steel piping used in the TCWS piping is ductile and incipient failures should be revealed by leaks before any crack reaches a critical crack size. For such materials and with a reasonable leak detection system, double-ended guillotine failures can be considered 'hypothetical events'. The containment volume, at minimum, is designed to confine the pressures resulting from a leak in the TCWS piping under any foreseen conditions. Due to the higher temperatures, pressures resulting from a leak during machine baking (when the coolant temperature is ~ 240°C) is limiting. A building design pressure of ~ 200 kPa (absolute) is adequate to confine any such leak with a margin to allow for computational uncertainty.

Nonetheless, the containment volume can also confine the pressure caused by any pipe failure during plasma operation up to and including a double-ended guillotine rupture. It is only during pulsed operation of the plasma (coolant temperatures ~ 150°C) that an ex-vessel piping failure could lead to in-vessel failures (due to resulting overheating of in-vessel components or disruption damage), and hence potentially release the in-vessel source term (tritium, dust) into the containment volume. A building design pressure of ~ 200 kPa (absolute) is also adequate to confine any such release with a margin to allow for computational uncertainty. This approach of confining breaks even up to double-ended guillotine failures during operation adds margin to the design and decreases the importance to public safety of being able to correctly predict critical crack sizes and leak rates, and of having a sensitive leak detection system capable of working under transient conditions such as during a pulse.

Leaks through PHTSs heat exchangers can be isolated (e.g., isolation valves on the heat rejection system). The vacuum vessel primary heat transfer system (VV PHTS) is somewhat different from the others in that it is very well shielded from the plasma and neutrons by the in-vessel components. As a result the VV PHTS contains very low inventories of tritium and activated corrosion products even under accident conditions and hence does not require a secondary confinement barrier.

Heating, ventilation and detritiation systems provide additional confinement functions (isolation, detritiation, filtering) for the port cells surrounding the tokamak. These systems service rooms where a release of tritium or tokamak dust is possible. Releases to the environment would be reduced by providing filters to remove particulates, dryers to remove tritiated water vapour, or detritiation systems to remove elemental tritium. Releases are routed to a monitored release point.

The primary confinement for the fuel cycle tritium inventory is the process equipment and pipes of the vacuum pumping system, tritium plant, and fuelling system. These are surrounded by secondary barriers such as glove boxes, cold boxes, vacuum jackets, and other enclosures. For example in the isotope separation system, tritium is contained by the high quality piping and components of the distillation columns which are themselves within in a cold box under high vacuum for thermal insulation and which also forms another confinement barrier. Rooms are maintained at a negative pressure with respect to the external atmosphere, and the exhaust can be treated by detritiation systems and is directed to the monitored plant exhaust.

The hot cell source term includes tritium and activated materials that arise from the refurbishment and the storage of in-vessel components. Hot cells, casks, and process equipment (e.g. for tritium recovery from plasma-facing materials) form the primary confinement barrier, and the hot cell building exhaust can be treated by filters and detritiation systems, and is directed to the monitored plant exhaust.

Other buildings with radioactive materials include the low level radwaste building and the personnel building. Floor drainage from such radiologically controlled areas is collected to tanks where it can be monitored and, if necessary, treated before it is discharged to the environment. Ventilation from the potentially contaminated, radiologically controlled spaces in these buildings is routed to the monitored plant exhaust.

Using the vacuum vessel and cryostat as confinement barriers takes advantage of the inherent magnetic fusion characteristic that high quality and high reliability vacuums are needed for fusion operation. Failures of these barriers or leakage through them inherently (passively) terminate plasma or super-conducting magnet operation. The other confinement barriers are tailored to the nature of hazards in each compartment. The vacuum vessel and its extensions (including penetration barriers), the fuel cycle process piping, hot cells, etc. that form the primary enclosures for radioactive inventories, together with the building and detritiation/filter systems, may be considered a minimum set of systems to meet project release guidelines. The additional confinement barriers described in this section are available to reduce the spread of contamination in the facility and help protect personnel from exposures, and they also further enhance public safety.

5.2.4.4 Coolant Enthalpy

The use of pressurised water and cryogenic coolant requires consideration of the consequences of failure for the protection of confinement barriers, and design of the confinement barriers for accident pressures.

The vacuum vessel is connected to a pressure suppression system by ducts with rupture disks and bleed valves that permit connection at pressures below the rupture disk setpoint. Steam from an in-vessel coolant leak is routed though the ducts and condensed in the large tank of water of the VVPSS, which limits the pressure in the vacuum vessel to below its design pressure. Operation of the VVPSS is passive, relying only on pressure differences to open the rupture disks and force the steam through the water in the VVPSS tank to condense the steam. The bleed valves are provided to permit connection to the VVPSS at lower pressures which may avoid use of the rupture disks and limit recovery time and operator exposures. A drain line allows water to be drained from the vacuum vessel to the drain tank, limiting the generation of steam in the longer term. The VVPSS also provides vacuum vessel overpressure protection in case of cryogenic leaks into the vacuum vessel (e.g. from cryopumps).

The ex-vessel primary heat transfer system piping is surrounded by guard pipes, pipe chase structures, or is within the TCWS vault and/or NB cell (see section 5.2.4.3). Water and steam released if piping fails are routed to the vault. In the TCWS vault, coolers are provided to return the pressure to sub-atmospheric within 24 hours in the event of a steam discharge.

5.2.4.5 Hydrogen Inventories and Chemical Reactions

The potential for the following chemical reactions exists in ITER:
- hydrogen - air reaction with hydrogen isotope inventories in process systems in case of leaks from or into process piping;
- beryllium/carbon/tungsten - steam/air reactions inside the vacuum vessel at elevated temperatures following in-vessel coolant spills or air ingress;
- ozone formation (and the attendant explosion hazard) in liquid/frozen air caused by air in-leakage in a radiation field.

In the case of systems containing hydrogen, basic hydrogen safety design principles applied in the design include:

- preventing leakage of hydrogen isotopes (also required for radioactive material confinement);
- eliminating the formation of hydrogen/air mixtures by use of inert or vacuum second confinement or isolation;
- preventing the formation of flammable hydrogen/air mixtures in rooms by adequate ventilation;
- eliminating ignition sources.

Reduction of hydrogen generated by potential in-vessel chemical reactions consists of:

- limiting the quantities of chemically reactive dust on hot plasma-facing components of the divertor to less than 6 kg each of beryllium, carbon and tungsten;
- terminating fusion power to prevent heat up to high temperatures of in-vessel components by terminating the heat load in the event of an upset in their heat transfer systems to limit hydrogen production from Be-steam reactions[1];
- ensuring heat removal to reduce temperatures and hence reduce reaction rates - the design ensures that in-vessel temperatures remain below 385°C for the beryllium surfaces except during the few tens of minutes of initial transient after fusion power shutdown (see section 5.2.4.6).

The design of the cryostat must limit air leakage in the cryostat below an acceptable value to maintain thermal insulation for cryogenic cooling of the superconducting magnets, and this also prevents the potential ozone production (and hence explosion) hazard. Provision is made to detect cryocondensates on the cryostat interior cold surface. All cryogenic needs across the ITER facility (except inside the cryoplant) are met by helium, not by liquid nitrogen to avoid the formation of ozone.

5.2.4.6 Heat Removal

Heat must be removed during normal operation to prevent component failures and following events to limit chemical reactions, to limit source term mobilisation, and/or to help ensure confinement integrity is maintained.

The decay heat time evolution after shutdown is shown in Figure 5.2.4-2. At the end of life (conservatively assuming a fluence of 0.5 MWa/m^2 to maximise predicted decay heat) the ITER global decay heat amounts to ~11 MW at shutdown and decreases to 0.6 MW after one day.

[1] for example, M.J. Gaeta, B.J. Merrill, H.-W. Bartels, L.N. Topilski, and C. Laval, "Short term hydrogen production issues for ITER", Fusion Technology, Vol. 32 (1997), pp 23-34.

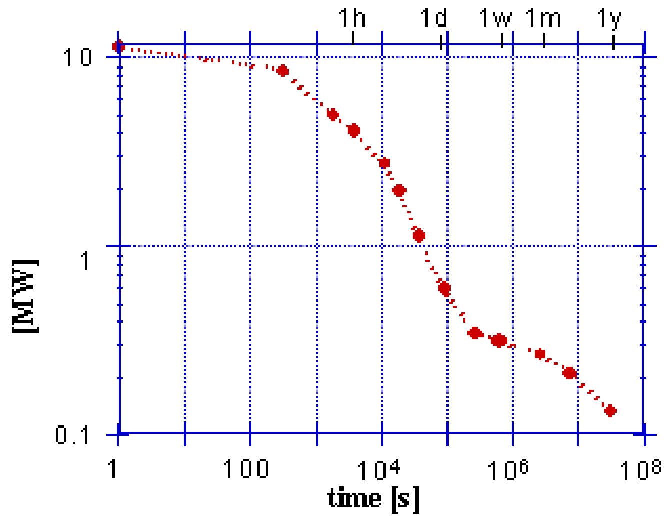

Figure 5.2.4-2 Global Decay Heat In ITER at End of Life (fluence 0.5 MWa/m^2)

Successive lines of defence of systems to provide heat removal are available:

- multiple normal heat removal paths and systems provided for machine operation (which are assumed not to function for safety assessment);
- two independent loops of the vacuum vessel cooling systems;
- a natural circulation capability of the vacuum vessel PHTS should electrical power for its pumps be lost;
- thermal radiation to the magnet structures and cryostat and ultimately to the environment;
- the possibility of assisting ultimate decay heat transfer by introducing gas into the cryostat, degrading the vacuum insulation.

The VV PHTS has a natural circulation capability and is divided into two independent loops. Each loop is capable of removing decay heat and slowly cooling down the structures in the long-term following an accident (~ 0.8 MW). Initially the in-vessel components cool off as the plasma heat load is removed. No cooling is needed for several hours (> 12 h) as temperatures remain < 330°C. Subsequently, the in-vessel components slowly heat up until the decay heat levels decrease to within the vacuum vessel heat transfer system capability (~ 17 hours). The acceptability of the intermediate heat-up is confirmed in accident analyses. In the safety assessments, only the vacuum vessel cooling system is assumed to provide the decay heat removal function.

For hypothetical sequences (when all in-vessel and vacuum vessel cooling is assumed not to be functioning), investigations conclude that filling the cryostat with helium even weeks after the loss of cooling and ensuring air convection at the outer surface of the cryostat is sufficient to limit in-vessel component temperatures to < 650°C, precluding failure of confinement barriers and limiting any potential for chemical reactions.

Decay heat removal by radiation and natural circulation is an example of passive safety integrated into the ITER safety approach.

5.2.4.7 Magnetic Energy

The function of the magnet system is to provide the toroidal and poloidal magnetic fields necessary to initiate, contain and control the plasma during the various phases of machine operation. Since the magnets occupy valuable space around the machine, and are difficult to repair, the primary design considerations are compactness and reliability. However, the magnets contain large amounts of magnetic energy, are subjected to large electromagnetic loads, and are in proximity to confinement barriers. Safety is therefore also an aspect that requires consideration at the design stage. The magnet system is designed to be built and operated so that its failures cannot cause damage to confinement barriers that would result in a release of radioactivity exceeding the project release guidelines.

Because of the need for high availability levels required for machine operation, the magnets have multiple levels of defence against operational faults, so that the likelihood of any fault developing to the point where permanent magnet damage occurs is very low. For the same reason, margins are provided in the magnet design criteria, and the quality assurance procedures to be applied during manufacturing, pre-testing and commissioning of magnet components must be very thorough. Again, for investment protection, multiple diagnostics and/or inherent design features are included to achieve an inherently low probability of faults and/or a minimisation of their impact. Where active protection is used, the instrumentation and intervention systems are fully separate from the fault location and entirely independent of it. Reliance on the functioning of single critical components or diagnostics to achieve the desired low probability of magnet damage is avoided.

Nonetheless, to assess the design performance for safety, the inherently low fault probability has been neglected, and faults have been postulated and analysed with particular attention to the failure of the manufacturing inspection and active protection systems. Potential damage mechanisms are:

- structural: mechanical impact on other components due to abnormal loads on magnets caused by postulated faults or due to postulated structural failure of magnet components;
- thermal: energy deposition from the magnets into other components due to arcs arising from postulated faults.

The magnets are massive components, and the magnet structural elements are distributed and contain large redundancy against failure. The design of these structural elements take into account normal and abnormal loading conditions to prevent structural failure or thermal damage to the point where the vacuum vessel or its extensions are damaged. Fault detection to switch off power to the PF coils is included to mitigate damage that may be caused by arcs. Damage from postulated arcs is limited to cryostat feedthroughs and there is no release of radioactivity as a consequence of magnet system failures.

5.3 Radiological Source Terms

ITER will contain radioactive and hazardous materials as well as energy inventories. They have been quantified as part of the safety assessment[1] since they are the basis for virtually all safety and environmental analyses.

The relevant source terms are:
(1) tritium,

[1] Generic Site Safety Report, Volume III, Radiological and Energy Source Terms, G 84 RI 3 00-12-14 W0.2.

(2) tokamak dust,
(3) activated corrosion products (ACPs).

The total site inventory of tritium will be < 3 kg. The ITER project has set an ambitious guideline for initial operation of 450 g tritium for the maximum mobilisable amount of tritium inside the vacuum vessel or the fuel cycle sub-systems. The cryogenic pumps have a maximum tritium inventory of 120 g. Thus the maximum amount of tritium in co-deposited layers and implanted in plasma facing material should be limited to less than 330 g by regular removal (see section 2.4). Tritium bred inside the beryllium of the first wall is considered to be immobile because of the limited temperatures in ITER off-normal operation. These ambitious guidelines were set to push the design into a direction of minimising tritium inventories. For safety assessments, larger values were used (1000 g in vessel and 700 g in the fuel cycle) to provide margins for measurement uncertainties. Because of the wide margins adopted in the confinement of these inventories, operating values above the guidelines would not invalidate the safety approach. The tritium inventory is segmented such that only a relatively small fraction is available for mobilisation and release during any specific event. Multiple confinement barriers including tritium cleanup systems are provided to mitigate the consequences of any tritium release.

The vast majority of the activation products are tightly bound in solid metallic structures and are not mobilisable during any credible off-normal scenario. ITER energy sources are controlled such that activation product mobilisation is generally limited to erosion (dust) and corrosion of structural materials.

The plasma-facing components are beryllium, carbon, and tungsten. A guideline to limit the mobilisable tungsten dust to 100 kg inside the vacuum vessel has been established due to its radiological hazard. For safety assessments, larger values were used (350 kg in vessel) to provide margins for measurement uncertainties. Dust particles are prone to chemical reactions with steam, forming hydrogen in off-normal situations when coolant is leaked into the vacuum vessel. To limit this potential source of hydrogen, guidelines are set for the maximum amount of dust on the divertor surface. These surfaces can be hot enough in off-normal situations to cause oxidation of dust and subsequent hydrogen production. The guideline is 6 kg dust each for carbon, beryllium and tungsten (see section 5.2.4.5). Dust production and removal is further discussed in Section 2.4.

Some ACP will be present in the various in-vessel and vacuum vessel coolant loops as well as in coolant loops for the test modules, auxiliary heating and diagnostic equipment exposed to a neutron flux. The ACP affect occupational exposure, routine effluents to the environment and potential releases due to accidents. The amount of ACP has been assessed to be < 10 kg as deposits, < 60 g as cruds and ions per cooling loop. The hazard potential of the ACP in the vacuum vessel (VV) HTS is < 1% of the hazard potential of the ACP in the primary-first wall/blanket (PFW/BLK) HTS because the neutron flux is reduced by several orders of magnitude inside the vacuum vessel. Specific radionuclide concentrations used in the safety assessments are presented in Table 5.3-1.

Table 5.3-1 Activity of Tungsten and FW/shield Activated Corrosion Products (ACPs)

Tungsten activation (plasma surface layer: 25 micro-m)			**ACP deposits (steel)**			
isotope	half life [y]	activity [Bq/kg]	isotope	half life [y]	deposit activity [Bq/kg-deposit]	Ion and cruds in solution activity [Bq/kg-ion/crud]
W 187	2.72E-03	5.24E+14	Fe-55	2.73E+01	2.07E+12	9.61E+11
W 185	2.06E-01	3.71E+13	Mn-54	8.55E-01	9.86E+10	3.49E+11
W 185m	3.17E-06	3.64E+13	Mn-56	2.94E-04	1.35E+12	1.19E+13
W 181	3.31E-01	1.43E+13	Co-58	1.94E-01	1.06E+11	3.92E+11
Re188	1.94E-03	6.01E+12	Co-60	5.27E+01	1.41E+11	2.39E+11
Re186	1.03E-02	2.20E+12	Cr-51	7.59E-02	1.14E+11	4.54E+08
Re188m	3.54E-05	5.79E+11	Ni-57	4.11E-03	4.52E+10	8.85E+10
W 179	7.13E-05	2.56E+11	Co-57	7.44E-01	2.64E+11	4.96E+11
Ta182	3.14E-01	1.54E+11				
W 179m	1.22E-05	1.02E+11				
Ta186	2.00E-05	6.34E+10				
Ta183	1.39E-02	6.18E+10				
Ta184	9.92E-04	4.34E+10				
Ta182m	3.04E-05	2.88E+10				
Ta179	1.61E+00	2.74E+10				
Re184	1.04E-01	1.99E+10				
Ta180	9.22E-04	1.15E+10				
Hf183	1.22E-04	9.64E+09				

5.4 Normal Operation

The assessment of normal operation addresses potential effluents and emissions to the environment, occupational safety of personnel working at the facility and radioactive materials generated during operation and decommissioning of ITER. The design and operation of ITER strives to reduce effluents, occupational exposure and wastes in accordance with the ALARA principle. Efforts to ensure very low levels will continue, but no concerns have been identified by the assessments.

5.4.1 Effluent Sources and Control

The ITER design incorporates many features to ensure that environmental impact during normal operation will be low. The hazards are known and the control technologies are well established. According to the analyses performed, the facility can be operated to satisfy the conservative safety and environmental guidelines that have been established by ITER. Continuing application of the process to implement the ALARA principle may further reduce the estimated normal effluents.

This section discusses effluents that may be expected during normal operation. The primary materials of concern are tritium and activated materials. In addition, the facility will produce electromagnetic fields at the site boundary less than the earth's natural magnetic field and will reject thermal energy. Normal operation includes operation of all systems required to carry out the experimental programme, as well as maintenance, hot cell operation, and radioactive materials management operations.

5.4.1.1 Summary of Effluents Estimates

ITER has set restrictive project release guidelines, as given in Table 5.2.3-2. In addition to meeting prescribed host country limits, the ALARA principle is recommended by international nuclear safety experts and incorporated in national regulations of many countries. The International Basic Safety Standards[1] explain ALARA as the objective to keep all exposures to values such that further expenditures for design, construction, and operation would not be warranted by the corresponding reduction in radiation exposure. For ITER this process involves a systematic review of systems, activities and pathways with a potential of effluents, estimating effluents, and examining ways to reduce the main contributors.

Sources of potential effluents have been identified, discharge pathways determined, and design features and active discharge control systems assessed for expected end of life conditions which is assumed to include extensive maintenance and refurbishment in the hot cell. Conservative assumptions are made so as not to underestimate potential effluents. Effluents pathways are controlled and monitored through the plant exhaust, the liquid discharge pathways, and the heat rejection system. Table 5.4.1-1 provides a summary of estimated annual effluents and compares these to the project guidelines. Annual effluents will increase with time from the start of operation as activation levels increase and tritium permeates, and will vary from year to year depending on activities undertaken, but are unlikely to exceed the values in Table 5.4.1-1 given the conservative assumptions used in making the estimates.

Table 5.4.1-1 Estimates of Effluents at Expected End of Life conditions and Comparison with Project Guidelines

<table>
<tr><th>Species</th><th>Estimate</th><th>Project Guideline</th><th>% of Guideline</th></tr>
<tr><td>Tritium – as HTO in air</td><td>0.05 g tritium /a
(18 TBq/a)</td><td rowspan="2">0.1 g tritium /a</td><td rowspan="2">50</td></tr>
<tr><td>Tritium in water</td><td>0.0004 g tritium /a
(0.14 TB/a)</td></tr>
<tr><td>Tritium – as HT in air</td><td>0.18 g tritium /a
(67 TBq/a)</td><td>1 g tritium /a</td><td>18</td></tr>
<tr><td>Activated dust</td><td>0.25 g metal/a</td><td>1 g metal/a</td><td>25</td></tr>
<tr><td>Activated corrosion products (ACP)</td><td>0.85 g metal/a</td><td>5 g metal/a</td><td>17</td></tr>
<tr><th colspan="4">Species with no specified project guideline</th></tr>
<tr><th>Species</th><th>Estimate</th><th colspan="2">Notes</th></tr>
<tr><td>Activated gases
^{41}Ar
^{14}C</td><td>
<1 TBq/a
10 MBq/a</td><td colspan="2">Release limits not specifically established for these isotopes but not significant at these levels</td></tr>
<tr><td>Direct radiation at 250 m</td><td>4 µSv/a</td><td colspan="2">< 0.5% of background</td></tr>
<tr><td>Beryllium (not activated)</td><td>~ 0.1 g/a</td><td colspan="2">This is well below current limits for public protection as based on reviews of existing regulatory limits, official investigations and other publications.</td></tr>
<tr><td>Magnetic fields at 250 m</td><td>< 20 µT</td><td colspan="2">Less than Earth's magnetic field (25– 65 µT)</td></tr>
<tr><td>Thermal</td><td>411 MWt (pulsing at 25% availability)
180 MWt (shutdown)</td><td colspan="2">Not significant</td></tr>
<tr><td>Cryogen – Helium</td><td>~ 1-3 t/a</td><td colspan="2">Not significant</td></tr>
</table>

[1] "Clearance Levels for Radionuclides in Solid Materials", International Atomic Energy Agency, IAEA-TECDOC-855, Vienna, 1996.

A detailed effluents assessment was performed for each system to ensure that no significant discharge pathway was missed. This detailed information has been further analysed to provide:

- a breakdown of effluents by discharge points (Figure 5.4.1-1);
- a comparison of effluents from operations and maintenance (Figure 5.4.1-2).

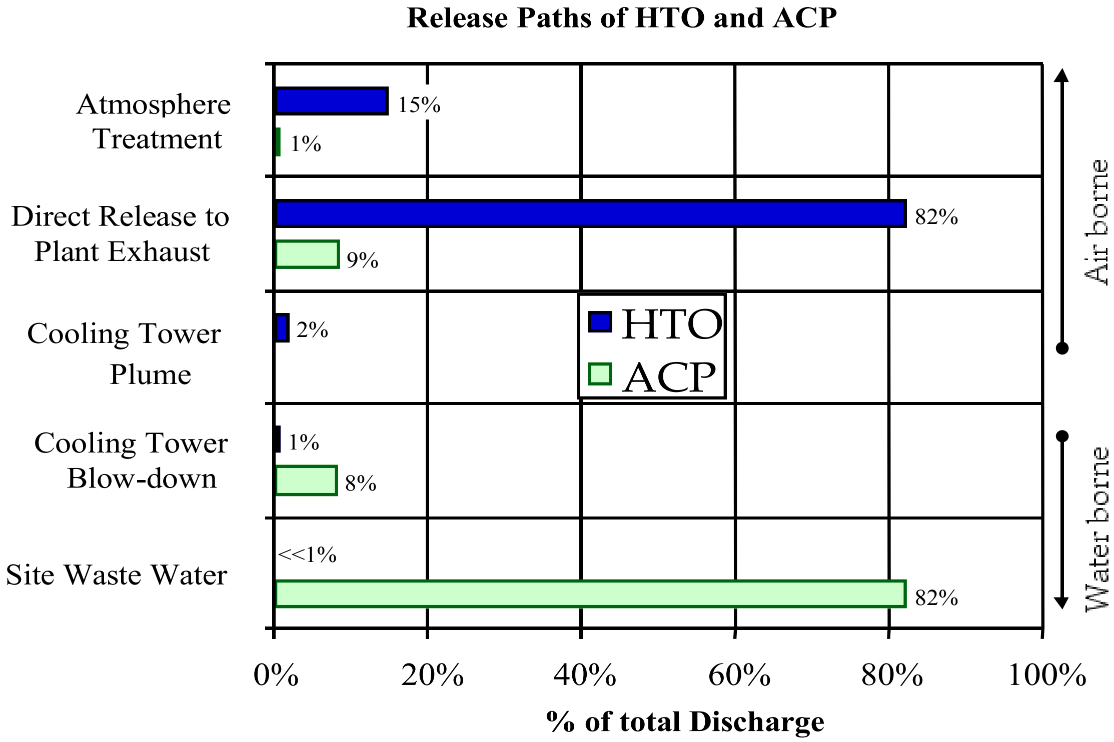

Figure 5.4.1-l Fraction of the Total Amount of Tritium (HTO) and of Activation Corrosion Products (ACPs) Effluents from Specific Release Points or Paths

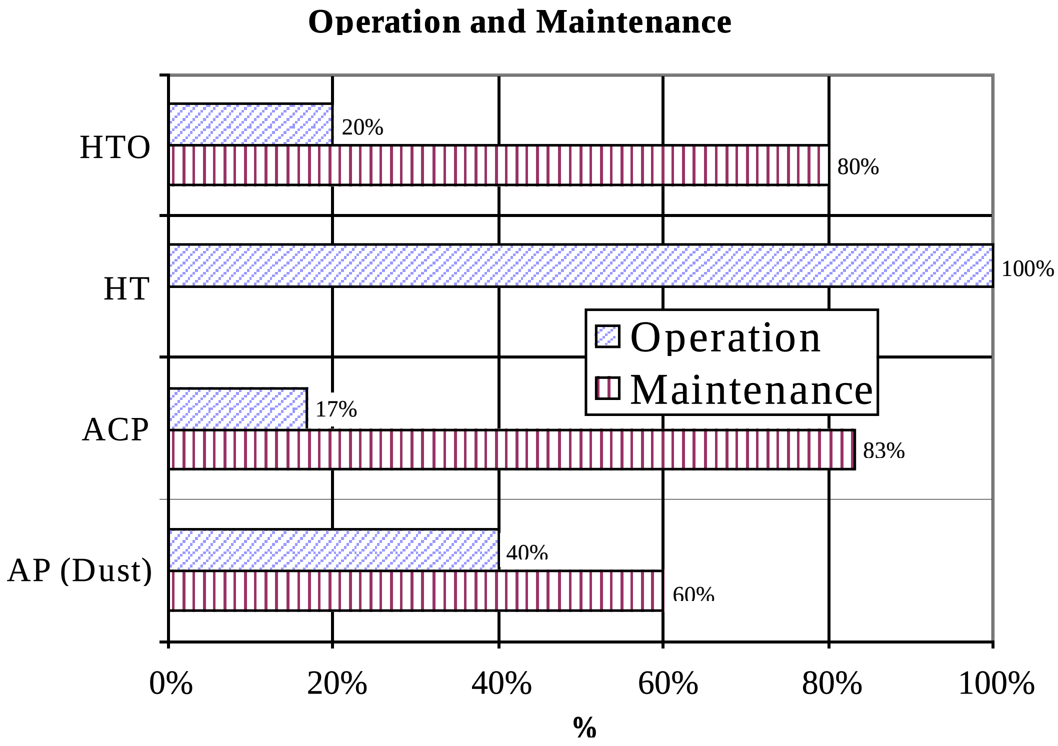

Figure 5.4.1-2 Fraction of the Total Amount of Tritium (HT or HTO), Activated Corrosion Products (ACP) and Activation Products (AP) Effluents during Operations or Maintenance

There is some uncertainty in any estimate of effluents since ITER is a first-of-a-kind experimental facility that will be resolved only when ITER begins operation. However, any radioactive effluents will increase slowly over the project lifetime, so there would be adequate time to upgrade discharge control systems if needed to maintain effluents at low levels. The pulsed nature of the operation also provides some flexibility in scheduling operations with potential effluents to ensure that effective control measures are in place beforehand. The preliminary estimates of effluents, and flexibility in the design to add additional or upgraded discharge control measures, provide confidence that the design can meet the project guidelines. Furthermore, a process is in place to ensure that effluents will be ALARA.

5.4.1.2 Tritium

There is a potential for tritium effluents from the following monitored release points:
- discharge from atmosphere detritiation systems through the plant exhaust;
- low concentrations discharged through the HVAC directly to the plant exhaust;
- discharge through the cooling tower plume;
- low contaminated water discharged with site waste water.

For each of these, effluents for normal operation and maintenance have been assessed and are summarised above in Figure 5.4.1-1. The tritium loss through HVAC to the plant exhaust results from numerous maintenance operations of large components where each individual discharge is small and further reduction by additional control systems is not warranted. The small amount of discharge from operational atmospheric detritiation systems results from the high removal efficiency, 99.9%, of these systems.

A small amount of tritium in the cooling tower blowdown as liquid effluent is included in the estimate, which is only due to the conservative assumption that the heat exchangers may have chronic leaks.

5.4.1.3 Activation Products

The primary contributors to operational effluents with activation products are the following:
- corrosion products from primary heat transfer systems (PHTSs);
- activated dust from the vacuum vessel;
- activated gases.

ACPs from Coolants

Activated material enters the PHTSs coolant by corrosion of the activated cooling channels in the components (e.g., blanket modules) within in the vacuum vessel. The isotopes entering the coolant depend on the material at the water interface. The three loops for the primary first wall/blanket (PFW/BLK) PHTS have stainless steel at the water interface. The vacuum vessel also has a steel/water interface but the neutron fluence is significantly lower. The divertor/limiter (DIV/LIM) PHTS loop has steel and copper at the interface. This is also the case for the NB PHTSs but the neutron fluence is lower for those systems.

The corrosion rate and solubility are strongly affected by the water chemistry and the extent of the water treatment during operation. The treatment during operation is performed by the Chemical and Volume Control Systems (CVCSs) described in section 3.3. This system employs chemical control, filtering, and ion exchange to remove ACPs. Table 5.4.1-2 lists the most significant isotopes for the coolant systems with stainless steel at the interface.

Table 5.4.1-2 Key Isotopes for the Coolant Systems with SS316 at the Water Interface

^{51}Cr	^{54}Mn	^{55}Fe	^{56}Mn	^{57}Ni	^{57}Co	^{58}Co	^{60}Co

The pathways for effluents are by leakage from the TCWS via the heat exchangers to the heat rejection systems, and leakage from components into the TCWS vault. Potential pathways and release points include the following:
- water discharged with the site waste water,
- water leakage into the TCWS vault and released by the HVAC system,
- leakage through the heat exchangers and loss to the cooling tower.

Operational effluents directly from the TCWS components not including heat exchanger leakage are primarily through the TCWS vault atmospheric control systems or from releases not captured by dryers or filters. During operation, the airflow in the vaults is continuously circulated through air filters with only a small stream extracted to maintain negative pressure. The resulting decontamination efficiency is very high, so losses will not be significant. During maintenance operations in the TCWS vault, when worker doses are an important consideration, air flow may be increased by directing flows directly to the plant exhaust filters. ACP effluents from the cooling tower blowdown is only due to the conservative assumption that the heat exchangers may have chronic leaks late in their lives.

The estimated ACP effluents are dominated by discharges from the site waste water originating from the PHTSs. The waste water is filtered, but these effluents have been conservatively estimated without crediting filtering.

Activated Dust

The pathway for dust from the vacuum vessel includes the following steps.
- Dust is carried outside the vacuum vessel, for example during movement of in-vessel components, it is likely that some dust will remain in crevices of the component, even after cleanup. During the cutting and rewelding required for divertor removal, some activated particulates could enter the contamination control areas.
- Dust is lofted into the air. A significant fraction of the dust is of small particle size (< 2 μm) and conservatively assumed be lofted into the air during maintenance and cleanup operations.
- Dust transported through high efficiency filters. During the maintenance operations, contamination control barriers will be in place with air filtration. Any dust by-passing these barriers would be in controlled areas served by the plant exhaust filters. Dust removal by high efficiency filters is a function of particle size, and removal efficiency is expected to exceed 99.9% for particles > 0.3 μm in diameter.

In-vessel components are brought into the hot cell for refurbishing that could produce significant dust. For each divertor replacement, ~ 8 t of plasma-facing material could be processed, and the yearly amount of dust produced could be several tens of kilograms. Since all operations are done in the hot cell that is continually vented to maintain a negative pressure, the primary pathway for effluents is the ventilation system. During cutting operations, the dust produced is captured by high-flow air cleaners and filtered with multiple banks of high efficiency filters.

Activated Gases

Analyses have been performed to estimate the production and release of various activated gases that may be produced during ITER operation. These releases are small and include:
- activated air between the cryostat and bioshield - the main isotope of concern is ^{41}Ar;
- activated divertor gases - nitrogen, argon, and/or neon may be pumped into the divertor to produce line radiation leading to ^{37}S, ^{41}Ar, etc. The half-lives for the most significant isotopes are short, so that hold-up in processing of less than a day before discharge would reduce the estimated effluents by an order of magnitude.

5.4.1.4 Non-nuclear Effluents and Emissions

Beryllium

Non-activated beryllium effluents would come primarily from installation of plasma-facing components and refurbishing the divertor cassettes in the hot cell. After the machine is activated, installation would be done in the circulating gas that is maintained within the vacuum vessel. Particulate filters with an efficiency of better than 99.9%, would maintain effluents to very low levels. It is estimated that a few tens of kg of beryllium dust could be produced from preparation of replacement in-vessel components in the hot cell, which would lead to a discharge below 0.1 g/a.

Cryogens

Cryogenic systems are required by ITER to cool the superconducting magnets and operate cryopumps for hydrogen pumping. A large cryoplant will be on site to provide refrigeration and storage of cryogenic helium. Losses of cryogenic helium and nitrogen will be minimised for economic reasons. The environmental impact of these discharges should be minimal.

Heat

The heat rejected from ITER operations originates from nuclear heating (primarily during plasma pulsing with only a small amount of decay heat between pulses) and non-nuclear systems (during pulsing, the dwell between pulses, and periods between pulsing scenarios). The non-nuclear systems include additional heating systems, component cooling, and chilled water. The thermal discharge will be primarily from the cooling towers; however, there will be some direct losses (see section 3.3). The design and selection of an appropriate ITER site should allow dissipation of these heat loads without adverse environmental impact.

5.4.2 Occupational Safety

ITER has established a program for personnel protection against hazards anticipated during construction, operation and maintenance activities. The program addresses radiation protection and conventional hazards protection. The objective of this work is to ensure that occupational safety is adequately considered in the design of systems and components and, thereby, to gain confidence that a high level of worker safety will be achievable during operation and maintenance activities. Such confidence can be built only by developing a sound occupational safety program and performing ongoing assessments.

Programs and guidelines have been established for both radiological and conventional hazard protection. A preliminary occupational safety assessment has been performed, and the requirement for continuing analysis as the design progresses is recognised. The level of detail is commensurate with the information available and the relative safety risks, and further development is anticipated as the design progresses. Since ITER is a first-of-a-kind and an experimental facility, there are no definitive benchmarks for the guidelines and uncertainty cannot be completely avoided during the assessment. This uncertainty is balanced by conservatism in the analysis, and the time delay for the build-up of radiological hazards. Hazard sources will increase slowly with successful experimental campaigns, allowing the future operating organisation to acquire maintenance and operating experience during lower hazard conditions. This experience will provide a basis for improving the maintenance methods for enhanced worker safety. The current assessment does not credit improvements anticipated with experience nor the initially lower radiological hazards. The hazards considered in these occupational safety assessments include:

- radiological hazards from activated components, tritium, tokamak dust and activated corrosion products;
- industrial hazards, such as exposure to beryllium, electromagnetic fields, cryogenic fluids, inert gases, high voltage, and other hazardous substances.

5.4.2.1 Radiation Protection Program

Objectives

The radiation protection program (RPP) provides a set of systematic processes that identify and control both the radiation hazards and the exposure of personnel to these hazards. The RPP is based on internationally accepted radiation protection principles and well-established precedents. The objectives of the RPP are to:

- prevent occupational doses over legal limits;
- maintain personnel doses As Low As Reasonably Achievable (ALARA);
- prevent unplanned exposures;
- minimise the spread of contamination.

To meet these objectives, the RPP employs the principle of ALARA, the establishment of exposure limits and guidelines, and processes for controlling the movement of personnel and materials.

ALARA

ALARA is a key safety principle for ITER[1] and is applied to occupational safety as follows:
"The radiation protection practices shall be consistent with the IAEA and ICRP recommendations and should make use of best practices. In particular, efforts shall be made to design such that exposures during operation, maintenance, modification and decommissioning are ALARA, economic and social factors being taken into account."

The ALARA principle underlies each element of the RPP and is applied in the analysis of the maintenance scheme of each system. This is done using an ALARA process which is described in Section 5.4.2.2.

Exposure Limits

Limits for occupational exposures are taken from the ICRP recommendations[2]. The annual limit for radiation workers (RW) is 20 mSv measured over 5 years, not exceeding 50 mSv in any single year, and the annual limit for members of the public, non-radiation workers (NRW), is 1 mSv.

These limits are not directly used for design or operation, but they are used as a basis to derive the guidelines given in Table 5.2.3-1 for exposure control. It is anticipated that during operation, ITER will operate within guidelines that will be a fraction of the regulatory limit of the host country.

[1] Plant Design Specification G A0 SP 2, Section 3.3
[2] "1990 Recommendations of the International Commission on Radiological Protection", ICRP Publication 60, Pergamon Press 1991.

Table 5.4.2-1 Area Classifications and Radiation Access Zones

Access Zone (Area Classification)	**Access Limitations**	**Airborne / Total Dose Rate / Area Contamination Characteristics**
Zone A (Non-Supervised Area)	Unlimited Access.	· No airborne contamination. Dose rate < 0.5 μSv/h; · WHITE contamination control zones only: No surface or airborne contamination and no reasonable possibility of cross-contamination.
Zone B (Supervised Area)	Limited Access for NRW. [a] Unlimited Access for RW. [a]	· Total dose rate (internal + external) < 10 μSv/h; · GREEN contamination control zones acceptable: No loose contamination tolerated. May be subject to temporary surface or airborne cross-contamination, airborne should not exceed 1 DAC.
Zone C (Controlled Area)	Limited Access for all workers. Access requires planning and an appropriate level of approval for the hazards and the class of personnel requiring access.	· < 100 DAC and < 1 mSv/h; · AMBER contamination control zones acceptable: Airborne and loose surface contamination tolerated but must be identified and controlled. Contamination levels shall be maintained ALARA taking into account the risk of exposure, capability of available protective equipment, possibility of contamination spread, and cost. Airborne contamination in AMBER zones should not exceed 100 DAC.
Zone D (Controlled / Restricted Area)	These are restricted access areas, entry occurs only with a high level of approval from both an operational and a radiological safety view. These areas shall have physical barriers to prevent inadvertent personnel entry.	· Airborne >100 DAC or external dose rate > 1 mSv/h; · RED contamination control zones are only tolerated in Zone D. These areas have permanent or higher than AMBER levels of contamination.

[a] Personnel performing work requiring exposure to radiological hazards will be designated as Radiation Workers (RW). All other personnel, including non-designated visitors, will be treated as Non-Radiation Workers (NRW).

Notes: DAC = Derived Air Concentration: unprotected exposure to 1 DAC = 10 μSv/h

1 DAC HTO = 3.1×10^{5} Bq/m^3 = 8.4×10^{-6} Ci/m^3

For internal dose rate, hazard defined in DAC of airborne contamination

For external dose rate, hazard defined as μSv/h

Access and Zoning

A process to ensure safe personnel access is fundamental to meeting the RPP objectives. This requires a well-designed and properly implemented system of radiation access zones and the operational procedures to ensure proper use. This also forms a framework for contamination control, dose control, and prevention of unplanned, acute over-exposures. Access zones are defined[1] in Table 5.4.2-1. Hazard levels will build with operation. The zoning has been determined to be consistent with expected conditions at end of life to ensure adequate design provisions are in place at the start of operation to avoid the need for design modifications as operation progresses. Actual zoning during operation will be determined to be consistent with the hazards present and will change with time.

All rooms or areas are classified based on the exposure and contamination conditions estimated for 24 hours after shutdown for maintenance. This allows the accessibility of an area to be

[1] Plant Design Specification G A0 SP 2, Table 3-3.

determined when access would be normally required. Any activities or operating conditions that can cause an otherwise accessible area to have prohibitive occupational hazards will cause such an area to be Zone D. Because of the changing hazard conditions and the potential for error that this creates, these locations will have a physically interlocked system that has two requirements (1) to prevent the particular activity or operating condition if an affected location is occupied, and (2) to physically prevent human access for the duration of the activity or condition. A formal change control process will be implemented to allow the re-designation of locations from restricted to accessible.

Ultimately, accessibility of an area will depend on the zoning, actual hazard level, worker classification and current worker exposures. The RPP includes requirements, guidance, and good design practices to assure control of exposures and contamination during operation. Basic hazard control and monitoring, and radiation protection administration and management, are also included.

5.4.2.2 Radiation Protection Assessment

A radiation protection (RP) assessment process has been developed to gain confidence that a high level of worker safety will be achievable during operation and maintenance activities. The highest exposure and risk areas are identified as highest priority for potential design improvements aimed at reducing overall exposures and assuring good contamination control. Since ITER is not yet an operating facility, the assessment is of the planned activities for the given design and the proposed maintenance schemes. More refined assessments are performed as the design progresses, and greater design and operational detail is available. In addition, there will be several years of operation prior to introducing tritium for further learning and optimisation.

The radiation safety assessment involves two steps. The first is to review exposure conditions to assure that the access control and contamination control zoning requirements are met for the anticipated activities. The second is to apply the ALARA process which requires an estimate of collective occupational exposure. The assessment methodology for ITER is graphically presented in Figure 5.4.2-1. The iterative process begins with the review of the entire facility to prioritise systems for further detailed assessments. It then proceeds to the development of goals and criteria, the application of these goals to the specific systems, and possible modification of the facility or activities. Each repetition may result in a new set of goals and criteria being developed as the assessment is refined and the design is optimised. Dose reduction methods are first applied to those areas where it is identified that the impact would be the greatest. The process continues throughout design and operation to optimise the activities and reduce risk.

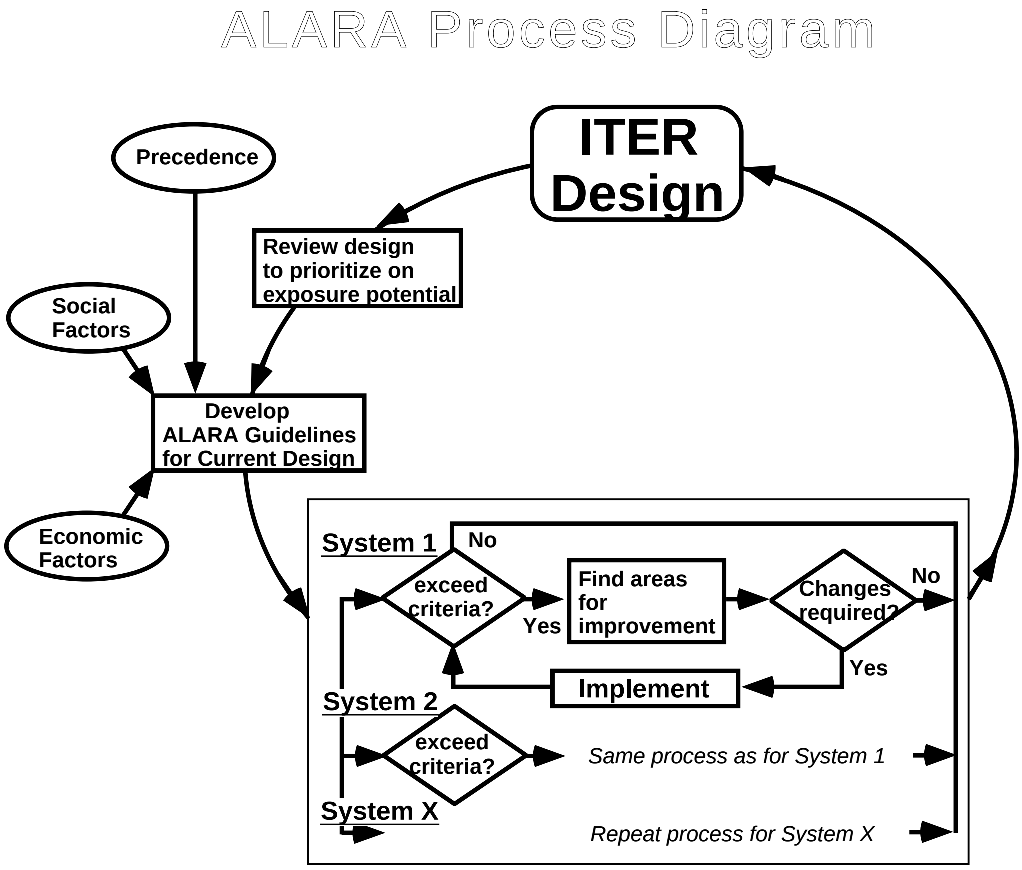

Figure 5.4.2-1 ALARA Process Diagram

Radiation Hazard Information and Zoning

The estimated radiological hazard information is used with the zoning criteria to define the access zone maps for ITER. The radiological conditions are based on neutron activation analysis modelled for the end of 20 years operation. This modelling is based on the reference operating scenario of 0.3 MWa/m^2 modified with 6 days added operation at 25% duty cycle at the end of each 10-year period. This is used for both neutron activation of in-vessel components and structures (see section 2.14) and for calculation of ACPs in the coolant systems. Tritium permeation and production in the coolant is considered along with component location and leakage rates, based on heavy water fission reactor (CANDU) experience, to estimate the potential levels of tritium in air. Finally, sources of contamination are identified along with potential pathways. The above is used to prepare access zone maps that illustrate accessibility of the various locations and ensure the viability of the proposed maintenance and operation schemes.

Access zone maps have been developed for the tokamak building, the tritium building, the hot cell building, the low level radwaste building, and the personnel and access control building.

ALARA Implementation

The ALARA process for ITER has already gone through one iteration, and the estimated occupational exposures reduced. A larger number of systems are being assessed in a more detailed manner as the design and operational and maintenance schemes become more detailed, and dose estimates become more refined. The process has been executed in three steps.

1. Preliminary qualitative review: Each system was initially reviewed for problems in accessing components for operation and maintenance activities, possible deterioration of contamination control, or significant potential for individual or collective exposure during required operation and maintenance activities.

2. Judgement of occupational radiological safety risk of systems: An initial judgement was made on the relative radiological risk posed by the maintenance of the various systems to prioritise the subsequent more detailed reviews. The systems in Table 5.4.2-2 were selected for more detailed review.

Table 5.4.2-2 Systems Selected for ALARA Review

WBS No	System
1.6	Blanket
1.7	Divertor
1.8	Fuelling and wall conditioning
2.6	Cooling water system
3.1	Vacuum pumping
5.1	Ion cyclotron heating and current drive
5.2	Electron cyclotron heating and current drive
5.3	Neutral beam heating and current drive
5.4	Lower hybrid heating and current drive
5.5	Diagnostics
5.6	Test blanket modules
6.3	Hot cell processing and waste treatment

3. Detailed analysis: The prioritised systems were scrutinised using the criteria as described below. The criteria are not absolute levels below which there is no further concern, but as described previously, the process is an iterative one that will continue into the operations stage, and the screening criteria reviewed and revised (downward) in subsequent iterations.

(a) The design is reviewed to identify any locations where the exposure conditions exceed 100 μSv/h, and it is determined whether the access and associated exposure warrants efforts to either reduce the dose rates or to change the maintenance procedure to reduce worker exposure to these conditions.
(b) The design is reviewed and if an activity is identified that would require an individual to exceed 0.5 mSv in one shift, then either the activity or the system is modified.
(c) The collective dose required for operation and maintenance activities for the system is estimated. Dose reduction methods are considered for systems with annual activities that exceed a total of 30 pers-mSv, or for tasks performed less often than annually if the occupational exposure for the task is estimated to exceed 30 pers-mSv. Steps to reduce the estimated exposure are taken, unless it is judged that "deployment of resources is seriously out of line with the consequent reduction."[1]

[1] 1990 Recommendations of the International Commission on Radiological Protection, ICRP Publication 60, Pergamon Press, 1991 (para 117).

This iterative process is used to optimise shielding and maintenance procedures during design evolution. It has led to stricter project guidelines and significant reductions in the estimated occupational exposure for systems which have undergone additional iterations. The initial analyses were performed for divertor cassette replacement and tokamak cooling water system (TCWS) maintenance, and the improvements are presented below. Analyses for the other systems in Table 5.4.2-2 (such as RF heating and current drive and diagnostic systems) are underway.

Tokamak Cooling Water System (TCWS) Maintenance

TCWS occupational exposure estimates have been ongoing during the EDA, following the design as it evolves and demonstrating a reduction in estimated exposure of two orders of magnitude. The estimated exposure has been systematically reduced from 1,400 pers-mSv estimated in 1996, to 250 pers-mSv for the 1998 ITER design, and to ~65 pers-mSv for the current design. This is the result of changes in the baseline operating scenario, improvements in the TCWS and CVCS designs, improvements in the inspection and maintenance programs, and improved accuracy of the code modelling and exposure assumptions. Table 5.4.2-3 shows the evolution of the TCWS occupational exposure analysis.

Table 5.4.2-3 Evolution of TCWS Occupational Exposure Estimates

Design Version	Major Changes	Collective Dose per Year person-mSv
1996	Initial design and assessment assumptions	1,400
Early 1997	CVCS decontamination factor increased from 2 to 10	500
1998 ITER Design	CVCS decontamination factor increased to 50, modification of CVCS units, improved inspection and maintenance plan	250
ITER	Reduced number of components, increased pipe-wall thickness, adopted realistic operating scenarios and worker positions	~65

The evolution of the analysis, as well as of the design, has resulted in reduced estimates and in increased confidence in the estimates.

Divertor Replacement

The assessment of the divertor replacement has been performed several times during the design evolution to optimise the shielding design and the maintenance procedure. The latest occupational exposure for divertor replacement was estimated based on dose rates in the divertor maintenance ports of about 100 μSv/h outside the VV closure plate but inside the cryostat and 50 μSv/h outside the cryostat but within the bioshield.. The dose rates used in the estimates reflect those achieved in other port maintenance areas.

The considerable reduction in estimates of occupational exposure is illustrated in Table 5.4.2–4. Compared with the initial analysis which used conservative gamma field levels for the tokamak design before optimisation, the current estimate shows almost two orders of magnitude improvement. The estimates are lower due to the introduction of shielding into the divertor port to reduce neutron streaming, and improved maintenance procedures.

Table 5.4.2-4 Example of ALARA Application for Divertor Maintenance

Design Version	Major Changes	Collective dose per year, pers-mSv
Initial	Initial design and assessment assumptions	404
1998 ITER Design	Shielding improvements including increasing port wall thickness and replacing 3 cm gas seal with 15 cm plate between VV and blanket	120
ITER	Improved maintenance procedure for pipe cutting and port seal cutting	~6

RF Heating and Diagnostic Maintenance

The most recent occupational exposure assessments are for the port plugs of the three RF heating systems plus diagnostics. These systems have similar remote maintenance requirements and maintenance procedures. For example, the standard port plugs are taken to the hot cell for refurbishment. The preliminary results indicate that much of the estimated exposure is due to pipe cutting or to activities immediately behind the port plugs. Current efforts are focussed on reducing the hands-on time for these two activities by optimising the use of remote cutters. These enhancements are expected to not only benefit the RF heating and diagnostic component maintenance, but also potentially reduce divertor, test blanket, blanket, and cryopump maintenance occupational exposures.

Overall Assessment

The project's commitment to ALARA ensures the continuing review, analysis, and improvement of design and maintenance procedures to ensure that exposures and radiological risks are not only below the guidelines, but also maintained as low as reasonably achievable. The current assessment of all major systems points to ITER successfully maintaining occupational exposures below the project guidelines. The reduction of estimated occupational exposure for the TCWS and the divertor systems has been successful enough to move the focus from these systems to the RF heating and the diagnostic systems. The iterative ALARA process systematically analyses and improves the systems of greatest concern. The widening application of systematic reviews ensures that potential contributors to occupational exposure are identified and addressed. Assessments of systems already reviewed are continuing with the objective of further decreasing the expected exposure and increasing confidence in the results.

5.4.2.3 Conventional Occupational Safety

Occupational safety protective measures will be implemented to assure worker protection against non-radiation hazards in compliance with the industrial safety standards of the host country.

Beryllium Hazard

Similar to radiation zoning, beryllium zoning and design measures for monitoring and control have been developed using JET experience[1]. The beryllium zoning criteria are defined in Table 5.4.2-5.

[1] Code of Practice for the Safe Use of Beryllium at JET, July 1989.

Table 5.4.2-5 Beryllium Zoning

Zone	Hazard Level		Access and Control Conditions
	Airborne	Surface	
Uncontrolled Zone	< 0.01μg/m^3	< 0.1 μg/m^2	No reasonable possibility for beryllium exposure. Time unlimited access areas with no protective or monitoring devices. Control Rooms, ordinary offices, all other areas not directly or indirectly connected to operations with beryllium.
Controlled Zone	0.01 μg/m^3 – 0.2 μg-/m^3	0.1 μg/m^2 - 10 μg/m^2	Beryllium measured or anticipated. Only designated beryllium workers are allowed access. Access time and protective equipment will be determined by the activities with beryllium bearing equipment and the potential for airborne beryllium.
Respiratory Protection Zone	> 0.2 μg/m^3	> 10 μg/m^2	Airborne beryllium is either measured or expected to exceed levels requiring respiratory protection. These areas must be physically enclosed and outfitted with appropriate ventilation to ensure the areas outside this zone do not experience elevated beryllium due to the work within the zone. Workers will require respiratory protection commensurate with the work and the hazards measured.

Electromagnetic and Radiofrequency Field Exposures

Electromagnetic (EM) and radiofrequency (RF) field exposure limits presented in - Table 5.4.2-6 and Table 5.4.2-7 are consistent with a survey of national regulations and international recommendations. With the poloidal field coils energised, working restrictions will be required in the tokamak hall and possibly in portions of the buildings immediately surrounding the tokamak hall. Design and administrative measures have been foreseen to monitor and control workers' electromagnetic exposure.

Table 5.4.2-6 Electromagnetic Field Zones and Conditions

Zone Name	B [mT]	Access and Control Condition
Uncontrolled Zone	< 10	Unlimited access.
Controlled Zone	> 10,< 100	Access is limited during tokamak operation such that individual exposures to EM fields are ≤ 60 mT–h per day.
Prohibited Zone	> 100	Entry is prohibited, aside from exceptional circumstances where up to 60 mT-h may be allowed with a high level of approval.

Table 5.4.2-7 Guidelines Related to Radio-Frequency Powers

Leakage at ECRH joints	< 5 mW/cm^2
Leakage at ICRH joints	< 1.0 mW/cm^2

Industrial Hazards

Other hazards that are likely at the ITER facility during construction and/or operation include high voltage, cryogens, confined spaces, fires, chemical hazards, mechanical hazards, rotating machinery, and lifting equipment (cranes). These hazards will be treated according to the industrial safety regulations and practices of the host country.

5.4.3 Radioactive Materials, Decommissioning and Waste

In the frame of ITER's safety objectives, the issue of radioactive materials, decommissioning and waste is carefully considered. It is important in this context to discriminate between the radioactive materials generated and that materials fraction actually remaining 'waste' after the decay time selected. All activities associated with radioactive materials and waste will comply with the host country regulations and practices. The management will be co-ordinated between the ITER project and the host country.

5.4.3.1 The Concept of Clearance

In the absence of an actual host country for ITER, the ultimate waste amounts are estimated provisionally on the basis of 'clearance'[1]. By this process, radiation sources can be released from regulatory control (i.e. control is removed). The "Interim Report for Comment" in the reference provides "unconditional clearance levels" for radionuclides in solid materials. According to the IAEA concept, materials with lower activation levels can be cleared, irrespective of how and where they may be used in the future.

Natural background radioactivity provides a reference level for the assessment of man-made radioactivity. Relative to this reference, doses around 10 μSv/a are commonly considered as insignificant, called[2] "trivial". The perception that a dose of 10 μSv/a appears to be insignificant has been used by IAEA for quantifying the concept of 'clearance' by the evaluating credible scenarios for the potential use and the associated exposure hazard of man-made solid radioactive materials after removal from nuclear facilities. The result of conservatively evaluating these scenarios is dose rates and doses, from which in turn "clearance levels" are derived to categorise radioactive materials. The radionuclide-by-radionuclide clearance levels are supplemented by a linear summation rule to be used for nuclide mixtures which are the most common case in practice. The numerical values of the clearance levels[3] have been supplemented by calculated values[4] consistent with the reference. In the actual evaluation, each isotope is characterised by its "clearance index" (specific radioactivity divided by the clearance level). A material for which the sum over the clearance indices of all constituent isotopes is above unity after the decay time considered cannot be cleared.

---

[1] "Clearance Levels for Radionuclides in Solid Materials", International Atomic Energy Agency, IAEA-TECDOC-855, Vienna, 1996.
[2] "Principles for the Exemption of Radiation Sources and Practices from Regulatory Control", Safety Series No. 89, IAEA, Vienna, 1988.
[3] "Clearance Levels for Radionuclides in Solid Materials", International Atomic Energy Agency, IAEA-TECDOC-855, Vienna, 1996.
[4] R.A. Forrest, J-Ch. Sublet: "EAF-99 biological, clearance and transport libraries", Report UKAEA FUS 410, December 1998.

5.4.3.2 Scope and Scale

The radioactive materials arising during operation and remaining after final shutdown include activated materials (due to fusion neutrons) and contaminated materials (due to activated tokamak dust, activated corrosion products and tritium) and mixtures thereof. Almost all radioactive materials present at final shutdown would be considered as waste if immediate disposal in final repositories was necessary. Actually, decay and decontamination will reduce the radioactivity with time after final shutdown. Therefore, not all radioactive materials may need to go into waste repositories; rather a significant fraction has the potential to be ‘cleared’. Since this fraction increases with time, a criterion with regard to time has to be set. At present the ITER project provisionally assumes that radioactive material not allowing for clearance after a decay time of up to 100 years is ‘waste’ needing disposal in a repository. The time period beyond 100 years will be called the ‘waste time scale’ in the following.

To guide the analysis of radioactive materials, assessments have been performed in parallel to the evolving ITER design. These 1D analyses are generally consistent with the more detailed neutronic analyses in section 2.14. The scale is shown by preliminary numbers in Table 5.4.3-1. About 90 % of the radioactive material is predominantly activated, the rest is predominantly contaminated. The waste masses do not include the bioshield for two reasons: Firstly its radioactivity is not due to activation by fusion neutrons, rather it stems from the natural radioisotope ^{40}K in the impurity element potassium in the concrete; secondly the clearance index of the concrete including reinforcing steel is always below unity, around 0.1. Relative to the indicative numbers, the actual ones will vary to a certain extent due to changes of the developing design, analyses details, dismantling possibilities, etc.

Table 5.4.3-1 Rounded Masses and Mass Fractions of Radioactive Materials

Total radioactive material	31,000 [t]
Material remaining as waste after a decay time up to 100 years*	6,100 [t]

* The waste mass results from assuming that component parts with clearance indices[1] above unity can be separated from the parts with indices below unity.

A sensitivity assessment indicates that the material remaining as waste after a decay time of only 30 years is about twice that after 100 years.

The mass of operational waste from component component maintenance, repair and replacements has been estimated to be about 750 t. This mass and the associated volumes are such that on-site accommodation in the hot cell storage area up to the final shutdown of ITER can be envisaged.

5.4.3.3 Conclusions from the Analyses

Relevant Isotopes

The vacuum vessel is a major contributor to the radioactive masses. The surface separating the non-clearable inner materials from the more peripheral, clearable ones is located inside the vacuum vessel walls. Therefore, the plasma side and rear side of the vacuum vessel have been selected as representative locations for most of the scoping studies.

[1] "Clearance Levels for Radionuclides in Solid Materials", International Atomic Energy Agency, IAEA-TECDOC-855, Vienna, 1996.

Figure 5.4.3-1 shows specific activities vs. decay time after final shutdown (total fluence 0.5 MWa/m^2) of the transmutation products in the ITER reference structural steel (SS 316 L(N)-IG with 0.01 and 0.05 wt.% of Nb and Co, respectively) at the plasma-side surface of the outboard vacuum vessel.

Figure 5.4.3-2 shows clearance indices vs. decay time after final shutdown of the transmutation products at the plasma-side surface of the outboard vacuum vessel.

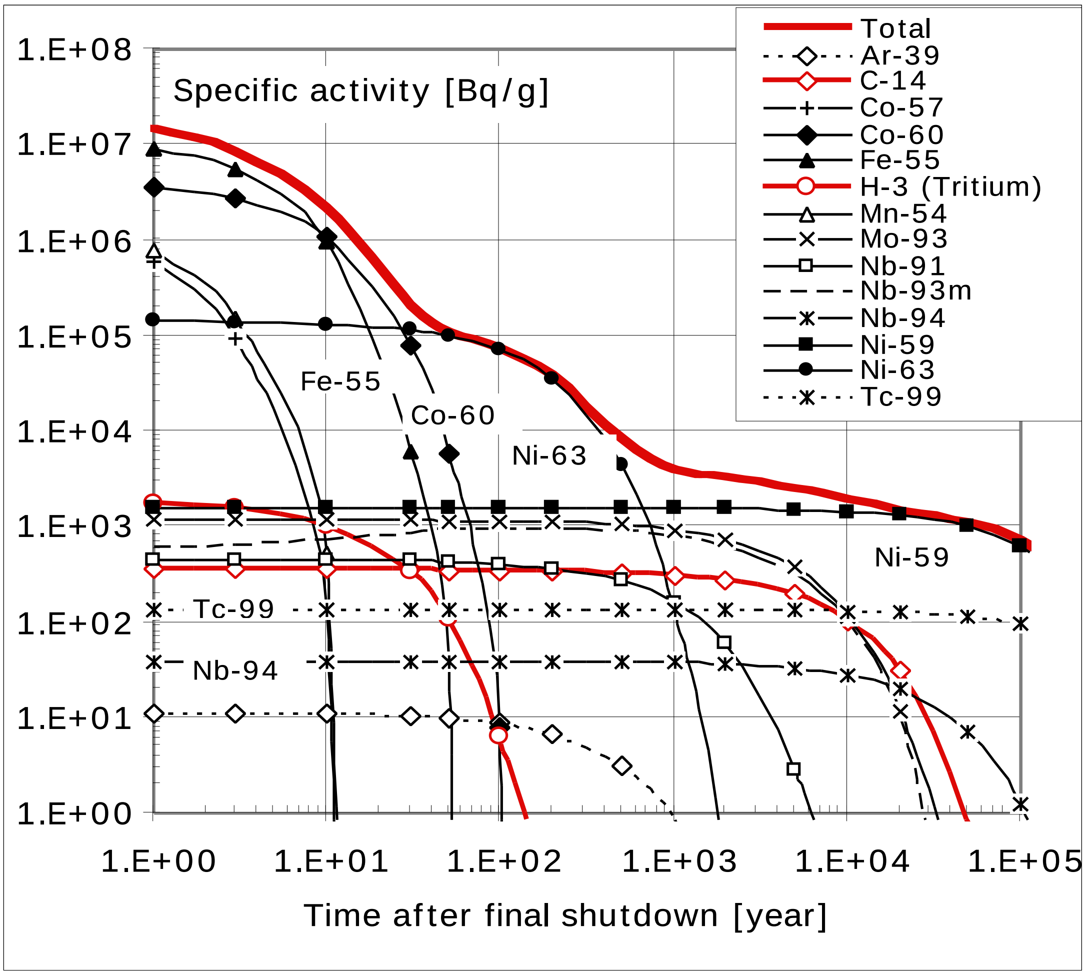

Figure 5.4.3-1 Specific Activities (vs. decay time after final shutdown) of the Transmutation Products in the Structural Steel (SS 316 L(N)-IG with 0.01 and 0.05 wt.% of Nb and Co, respectively) at the Plasma-side Surface of the Outboard Vacuum Vessel.
The legend is ordered alphabetically. The four most important Isotopes are marked by solid symbols and explicit labels (such as 'Co-60') are attached to the respective curves.

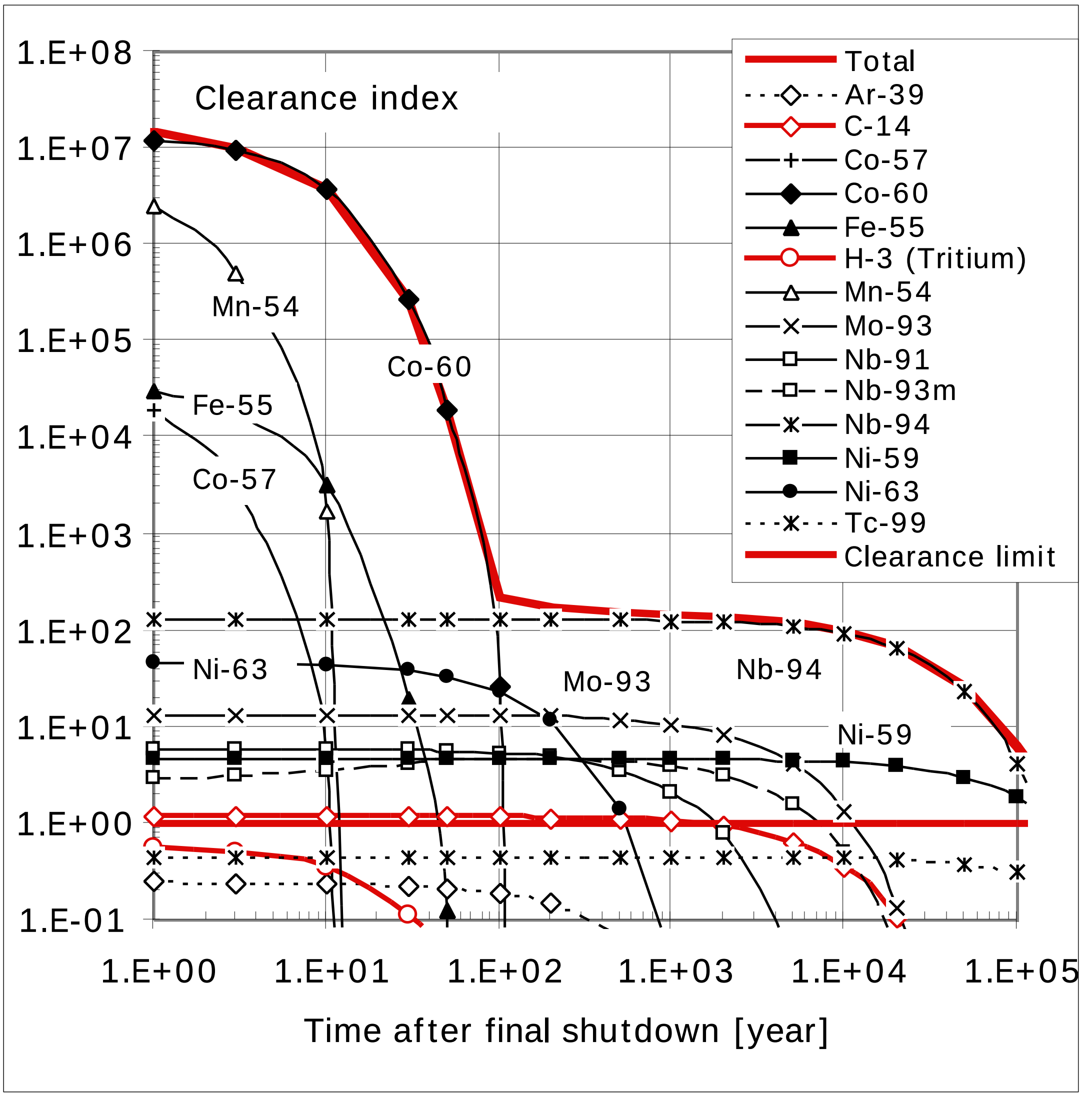

Figure 5.4.3-2 Clearance Indices (vs. decay time after final shutdown) of the Transmutation Products at the Plasma-Side Surface of the Outboard Vacuum Vessel (SS 316 L(N)-IG with 0.01 and 0.05 wt.% of Nb and Co, respectively).
The legend is ordered alphabetically. The five most important isotopes are marked by solid symbols and explicit labels (such as 'Co-60') are attached to the respective curves.

The steel alloying elements Ni, Mo and N in the plasma-side part of the outboard vacuum vessel give rise to the transmutation products ^{63}Ni, ^{93}Mo, ^{59}Ni, ^{14}C, ^{93m}Nb, and ^{99}Tc which are relevant since they significantly contribute to the activation level on the waste time scale. Some ^{94}Nb is also produced from the alloying element Mo by threshold reactions. The contribution of alloying elements to the total activation cannot be markedly reduced in the frame of ITER without having an impact on the properties of the reference structural steel.

The impurity elements Co and Nb are the only relevant ones in the reference structural steel since the other impurities and the so-called trace elements do not significantly contribute to the activation. The upper limits of 0.01 wt.% and 0.05 wt.% specified, respectively, for the Nb and Co concentrations in the reference structural steel are reasonable with regard to clearance and are technically feasible. The impact of these specifications on cost is negligible.

Table 5.4.3-2 shows a list of the most relevant radioisotopes (in alphabetic order), based on analyses of both the vacuum vessel front wall and the blanket first wall at 100 years after final shutdown. The most relevant isotopes with regard to clearance indices (shown in Figure 5.4.3-2) are also most relevant for the specific ingestion dose which is a measure of the radiotoxicity potential.

Table 5.4.3-2 The Most Relevant Radioisotopes for ITER Steels 100 Years after Final Shutdown

^{39}Ar	^{14}C	^{60}Co	^{93}Mo	^{91}Nb	^{93m}Nb	^{94}Nb	^{59}Ni	^{63}Ni	^{99}Tc

The relevance is determined by the concentrations of the parent elements in the reference steel together with the long half lives of the daughter radioisotopes. The only exception is ^{93m}Nb. It makes a noticeable long-term contribution in spite of its relatively short half-life (16.1 years) due to the continuous generation of ^{93m}Nb by capture of an orbital electron by ^{93}Mo nuclei.

Tritium in the Vacuum Vessel Walls from Transmutation and Permeation

Tritium generated by transmutation reactions in the steel of the plasma-side part of the vacuum vessel (see Figure 5.4.3-1) makes a negligible contribution to the total activation on the waste time scale. This implies that the same conclusion is true for the entire vacuum vessel.

Tritium due to permeation into the plasma-side parts of the vacuum vessel is not very relevant if considered only from the radioactivity point of view since the high activity of these parts is dominated by neutron activation. An impact is expected, however, from permeated tritium on decommissioning and disposal (outgassing, leaching, etc.), expressed by associated national requirements. The concentration of permeated tritium in the rear sections from the coolant side is expected to be very low but remains to be assessed.

Figure 5.4.3-3 shows a tritium concentration profile across the plasma-side vacuum vessel wall for an operation scenario that leads to a total neutron fluence of 0.5 MWa/m^2 after 17 years of operation. The peak tritium concentration (at the plasma-side surface) is about 16 mg tritium per m^3 of steel corresponding to a peak specific activity of 0.7 MBq/g. The total tritium inventory is about 0.25 g in the wall, causing an average specific activity of 0.2 MBq/g. Parametric analysis shows only moderate sensitivities on assumptions thus providing confidence in the calculated orders of magnitude.

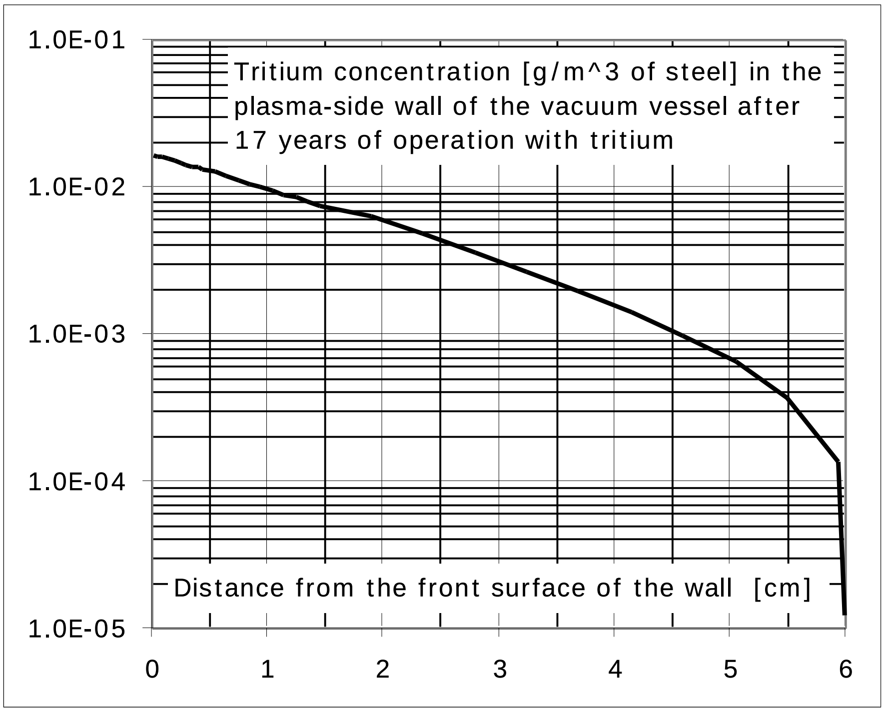

Figure 5.4.3-3 Tritium Concentration Profile across the Plasma-Side Wall of the Vacuum Vessel at Final Shutdown
(for an operation scenario that leads to a total neutron fluence of 0.51 MWa/m^2 after 17 years of operation)

Similar to the situation in the vacuum vessel walls, transmutation reactions of fusion neutrons with the elements in the in-vessel materials generate tritium. The specific activities of this tritium have been determined by the one-dimensional activation calculations.

The clearance indices in the blanket steel and copper are significantly above unity for at least the front half of the blanket up to 10 years. This implies that de-tritiation early after removal of the components from the device in the course of replacements or final decommissioning would be beneficial from the radiation protection point of view, possibly also with regard to tritium economy. The specific activity of tritium in divertor tungsten is about one order of magnitude above the clearance level, whereas the tritium activity in the first wall beryllium is much higher.

For tritium due to permeation into steel of the in-vessel components, the inventories calculated for the vacuum vessel front are used as a zero order approximation. These values are somewhat lower than the blanket front steel tritium activities from transmutation reactions which implies that the tritium from transmutation reactions alone warrants de-tritiation of in-vessel components after removal from the device, so that possibly higher tritium concentrations from permeation may add an additional incentive for de-tritiation but would not change the basic judgement on its necessity.

Relevance of Carbon-14

^{14}C is considered with increasing scrutiny due to carbon's major role in the biosphere. Dedicated work with regard to doses (both collective and individual ones) in particular from fusion device decommissioning and waste disposal has been launched recently, and provisional results have been obtained[1].

Figure 5.4.3-1 shows the specific activity of ^{14}C in the vacuum vessel versus time after final shutdown. The value at shutdown is about 600 Bq/g: the clearance level of 300 Bq/g is reached after about 5,000 years. Figure 5.4.3-2 shows that the contribution of ^{14}C to the total clearance index is masked by the much larger long-term contributions from ^{94}Nb, ^{63}Ni, ^{93}Mo and ^{59}Ni, but nevertheless the ^{14}C clearance index remains around unity for a long time. Eventually, dedicated considerations may be required with regard to specific regulations of the ITER host country.

Clearance Potential of the Vacuum Vessel

The profile of the total clearance index across the outboard vacuum vessel wall is displayed in Figure 5.4.3-4. About 30 cm at the plasma side of the outboard vacuum vessel will remain above the clearance limit beyond 100 years. Virtually the same thickness holds for the inboard vacuum vessel. The activity distribution implies that a significant fraction of the outboard vacuum vessel has the potential for clearance. A breakdown into clearable and non-clearable parts would require that the vacuum vessel be disassembled at decommissioning. A disassembly concept which exploits the vessel's shell structure and applies plasma cutting has been addressed in the design. The associated reduction in radioactive mass which would be waste is about 3000 t.

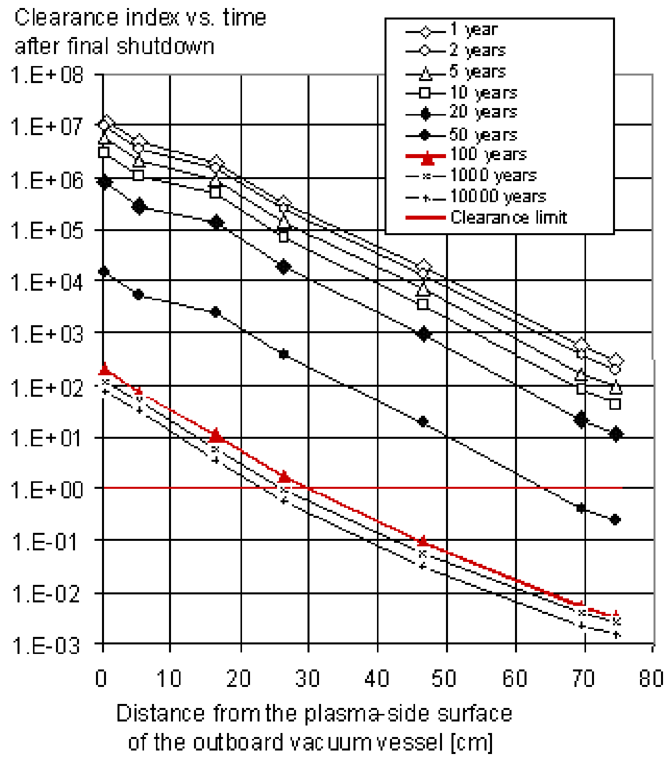

Figure 5.4.3-4 Profile Across the Outboard Vacuum Vessel Wall of the Total Clearance Index in the Structural and Shielding Steels for Various Decay Times After Final Shutdown

[1] T. Hamacher et al.: "Radiological impact of an intense fusion economy", Proceedings of the 21st Symposium on Fusion Technology (SOFT), Madrid, Spain, September 2000.

Clearance Potential of the Toroidal Field Coils

The toroidal field coils are major contributors to the mass of radioactive materials since they become activated by neutrons in spite of the shielding by the in-vessel components and the vacuum vessel. The long-term activation is dominated by ^{94}Nb. It is generated in particular from the Nb which is inherent to the superconductor in the coil winding packs and also from the Nb impurity in the cryogenic structural steels.

Virtually the entire inboard winding pack has clearance indices (ranging from about 100 down to about one) above the clearance limit beyond 100 years so that no clearance potential exists. The index for the outboard plasma-side of front winding pack is only marginally below the clearance limit at 100 years and beyond. The index in the rear part of the winding is more than two orders of magnitude below the clearance limit so that a potential for clearance exists.

The clearance indices of the coil cases show the usual cross-over of two different decay curves around 100 years: the curve up to 100 years is dominated by ^{60}Co, the curve beyond 100 years is dominated by ^{94}Nb. For the outboard field coil case, the concentration of the Nb impurity in the coil cryogenic steels, specified to be 0.01 wt.%, is sufficiently low from the clearance index point of view, with a margin of more than two orders of magnitude. The situation is less clear for the inboard coil case since its clearance index between 0.1 and 1 at 100 years is only marginally below the clearance limit.

Overall, the activity distribution inside the toroidal field coils is such that part of the coil cases and possibly the outboard winding pack have a clearance potential. A breakdown into clearable and non-clearable parts would require that the toroidal field coils can be disassembled at decommissioning. Therefore, a concept for the removal of the winding packs from the coils has been addressed in the design. The associated reduction of the radioactive mass which would be waste is about 2000 t.

Contact Dose Rates

Figure 5.4.3-5 shows contact dose rates vs. time after final shutdown at various vacuum vessel locations. These are average values since they have been calculated by a 1D model. The dose rates at the outboard rear surface of the vacuum vessel are particularly important since, together with the extent of remote operations and handling times, they determine staff exposure during decommissioning. At present it is assumed that dose rates around 10 µSv/hr may allow for prolonged hands-on operations. On average, the vacuum vessel's outboard rear surface reaches this value after about 10 to 20 years. Of course, more detailed calculations are required for an actual decommissioning program to determine the expected local dose rate peaks.

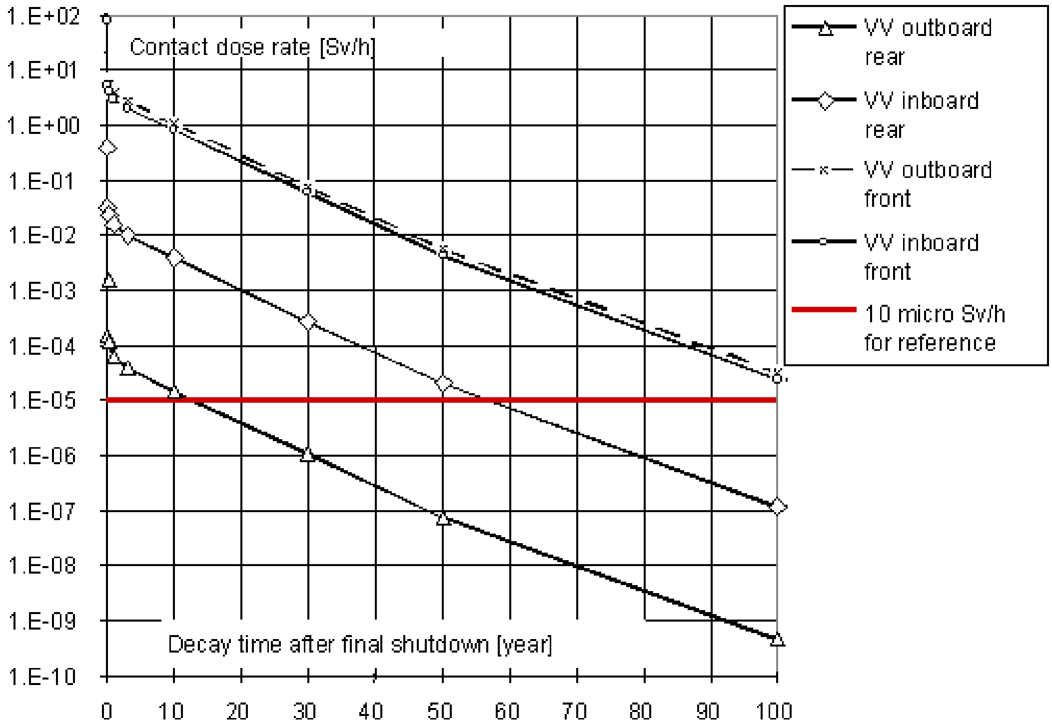

Figure 5.4.3-5 Contact Dose Rates at Various Vacuum Vessel Locations vs. Decay Time after Final Shutdown

Decay Heat

Nuclear decay heat has been calculated for the time point 100 years after final shutdown which is the starting time of the ITER waste time scale. A typical value is 30 μW/kg for blanket modules.

Radiotoxicity Potential

‘Radioactivity’ is supplemented by the important notion of ‘radiotoxicity potential’. Total radioactivity (in Bq) from operation, replacements and decommissioning remaining after final shutdown is not an adequate measure for the hazard associated with a nuclear facility. For example, the total radioactivity from component activation in a steel-based model fusion power reactor[1] - after a decay time of 100 years - is about four orders of magnitude higher than the total radioactivity of the ash from an equivalent coal-fired plant, whereas potential doses associated with the two plants differ by only an order of magnitude or so after 50 years. Therefore, it has become common practice[2] to use a ‘radiotoxicity potential’ (expressed by doses) to characterise the hazard potential of nuclear installations. The radiotoxicity potential is the hypothetical dose due to incorporation in the body of all radioactive material under assessment.

[1] J. Raeder I. Cook, F.H. Morgenstern, E. Salpietro, R. Bünde, E. Ebert: “Safety and Environmental Assessment of Fusion Power (SEAFP)”, Report of the SEAFP Project, European Commission, EURFUBRU XII-217/95, Brussels, 1995.

[2] H. Röthemeyer, A.G. Herrmann, H. Salewski: “The influence of radioactive waste disposal on natural activity, heat production, and radiotoxicity”, Kerntechnik 61, 5 – 6, 1996, p.245 – 250 (for example).

The radiotoxicity of ITER is dominated by the in-vessel components and decreases by several orders of magnitude within 50 to 100 years followed by a much slower decrease afterwards due to longer-lived radioisotopes. After about 100 years, the radiotoxicity of ITER is comparable to the total ash from a large coal-fired power plant. The most relevant isotopes are ^{54}Mn, ^{55}Fe and ^{60}Co in the shorter term and ^{63}Ni and ^{93}Mo in the long term.

5.4.3.4 Decommissioning Assumptions

It is assumed that the ITER organisation at the end of operation will be responsible for starting the machine decommissioning through a de-activation period after which the facility will be handed over to the responsible organisation inside the ITER host country.

The envisaged decommissioning scenario (see section 6.8) accounts for the impact of the essential drivers, such as availability of equipment, facilities, staff, activation decay time, etc. Flexibility is provided by the use of two separate phases. The duration and activity of each phase can be modified to a certain extent to accommodate organisational requirements and constraints.

During the first phase, immediately after shutdown, the machine will, in particular, be de-activated and cleaned by removing tritium and any removable dust from the in-vessel components. The necessary activities will be carried out by the ITER organisation by using the remote handling facilities and staff existing at the end of the project. At the end of the first phase, the ITER facility will be handed over to the responsible organisation inside the host country that will be responsible for the subsequent second phase of decommissioning and disposal. The scenarios demonstrate the feasibility of decommissioning considering worker and public safety.

5.5 Event Analysis

To assess the potential for public radiation exposures and the effectiveness of implementation of the safety requirements and functions in the ITER design, a comprehensive analysis of reference events has been performed. Failures and combinations of failures are postulated to show that the design is tolerant to such failures and to ensure a conservative operating envelope is achieved. The analyses included conservative assumptions of initial facility operating and off-normal conditions and examined possible ways for tritium, activated corrosion products in coolants, and neutron-activated tokamak dust, to be released to the environment. The results showed that radioactive releases for all of these reference events are well below the project release guidelines in Table 5.2.3-2. In addition, ultimate safety margins of the facility were examined by analysis of hypothetical events. Analysis shows that safety margins degrade gradually, and that the ITER design provides a high level of public protection even for these hypothetical events. The assessments provide confidence that the operation of ITER will result in no significant risk to the general public from postulated accidents.

5.5.1 Determination and Categorisation of Event Sequences

In the safety analyses, events and plant conditions are categorised as follows:

- **normal operation** - events and plant conditions planned and required for normal operation, including some faults and events which can occur as a result of the experimental nature of ITER;

- **incidents** - deviations from normal operation, event sequences and plant conditions not planned but likely to occur one or more times during the life of the plant as the result of components failures (excluding normal operation events);
- **accidents** - event sequences and plant conditions not likely to occur during the life of the plant but postulated to assess the safety of the facility.

The basic principle in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiation exposure to the public, and those conditions having the potential for the greatest exposure to the public should be those least likely to occur.

5.5.1.1 Reference Events

Table 5.5.1-1 shows the 25 different reference events that have been analysed in-depth with their respective categories. These were selected to cover the major systems, the radioactive inventories distributed amongst these systems, and initiator types that have the potential to cause releases, as well as the above event categories. The appropriateness of this set of reference events is confirmed by detailed event sequence studies as described in the following sections.

Table 5.5.1-1 Reference Events Analysed in Detail

Plasma events	
	Loss of plasma control/exceptional plasma behaviour (i, a)
Loss of electrical power	
	Loss of off-site power for up to 1 h (i)
	Loss of off-site power for up to 32 h (a)
	Loss of off-site power and on-site class III power for up to 1 h (a)
In-vessel events	
	In-vessel first wall pipe or coolant channel leak (i)
	Multiple first-wall pipe or coolant channel damage (a)
	Loss of vacuum through a vacuum vessel penetration line (a)
Ex-vessel HTS events	
	Loss of heat sink in divertor HTS (i)
	Pump trip/loss of flow in divertor HTS (i)
	Pump seizure in divertor HTS (a)
	Vacuum vessel HTS break (a)
	Large ex-vessel divertor HTS break (a)
	Heat exchanger leakage (i)
	Heat exchanger tube rupture (a)
Maintenance events	
	Stuck divertor cassette in transport cask (a)
	Maintenance accident on vacuum vessel (a)
Tritium plant and fuel cycle events	
	Tritium process line leakage (i)
	Transport hydride bed mis-handling (a)
	Isotope separation system failure (a)
	Fuelling line with impaired confinement (a)
Magnet Events	
	TF short (a)
	Magnet arc (a)
Cryostat Event	
	Air ingress (a)
	Water/air/helium ingress (a)
Hot Cell Events	
	Failure of confinement (a)

(i) Incidents are deviations from normal operation, event sequences or conditions not planned but likely to occur during the life of the plant (see Table 5.2.3-2).

(a) Accidents are event sequences or conditions not likely to occur during the plant life but are postulated to demonstrate the safety of the plant (see Table 5.2.3-2).

The reference events are analysed using detailed quantitative modelling and integrated system simulation codes and conservative or bounding conditions. In addition, all accidents are analysed assuming a coincident loss of off-site power and additional failures in mitigating systems. The plasma behaviour is addressed in a conservative way to show the limited effects of loss of plasma control or exceptional plasma behaviour. Loss of power is investigated to determine if there are requirements for emergency power. Many events are grouped around the cooling water systems which are a key element to demonstrate the safety approach. Air and water ingress into the vacuum vessel and cryostat under various off-normal plant conditions are investigated. Potential events during maintenance of the vacuum vessel are considered since maintenance will be a typical state of the ITER plant. Safety of the tritium plant with its significant inventory is addressed in four reference events. Magnet system structural integrity and consequences of arcs are examined. Consequences of failures in confinement and decay heat removal in the hot cell are investigated.

5.5.1.2 Enveloping by Reference Events

For the analysis to be complete, it must address the comprehensive list of fault conditions, or initiating events, covering all conceivable hazards arising in the facility. Thus sequence identification studies have been carried out with this objective, and more specifically to confirm that an appropriate selection has been made of the postulated events which are analysed in detail.

To help ensure that all aspects of plant operation have been considered, two fundamentally different approaches have been applied to the identification of potential initiators. These are the component-level (bottom-up) and the top-down approaches. The former is based on the application of systematic methods which seek to catalogue all potential faults in the plant components and sub-systems, and to consider the conceivable consequences of these faults. The focus is on the failure of individual components, and it is based on the design in as much detail as is available.

In contrast, the top-down approach starts at the plant level, and takes a global view of the potential hazards and the safety functions which provide protection. By considering the abnormal events which would have to occur to realise these hazards, a list of event initiators is again produced, in terms of system or in some cases component faults.

The concluding part of both the bottom-up and top-down studies is to show that all identified sequences have in some way been addressed by analyses which show that all potential consequences are within acceptable limits. The catalogue of events and the event sequences resulting from these studies identifies the radioactive inventory at risk, the confinement barriers challenged, the mitigating systems that must fail for a hazardous plant state to occur and the release pathway and have been evaluated to ensure that each one is either clearly insignificant or is covered, directly or indirectly, by a detailed analysis. It has been confirmed that the consequences of all identified sequences are enveloped by the assessed consequences in one or more of the analyses of reference events in Table 5.5.1-1.

5.5.1.3 Component level studies

The principal technique used in the bottom-up studies was failure modes and effects analysis (FMEA). The level of detail of the application has varied by system, because studies have proceeded in parallel with the evolution of the ITER design, resulting in systems being studied at various stages of their design maturity. FMEA catalogues all conceivable failures in every plant component, and the potential consequences. It allows frequencies to be assigned to each identified fault based, where possible, on failure frequency data gathered from operational experience on existing similar systems.

The outcome of FMEA studies is lists of fault conditions which could initiate an accident sequence. These are sorted into groups of initiators referred to as postulated initiating events (PIEs) which would lead to similar plant response. Some other, less formal, assessments of initiating events have also produced lists of PIEs. Table 5.5.1-2 summarises those systems that have been studied and the resulting number of PIEs listed for the normal operation phase.

Table 5.5.1-2 Postulated Initiating Events Identified in ITER Systems

System	Identification Method[1]	Number of PIEs
Magnet systems	IE list	1
Vacuum vessel	FMEA	15
First wall//blanket cooling water system	FMEA	13
Divertor cooling water system	FMEA	9
Fuelling system	IE list	8
Divertor remote handling	FMEA	5
Cryostat	FMEA	9
Heat rejection system	FMEA	6
Vacuum pumping	IE list	4
Tritium plant	FMEA	9
Cryoplant	FMEA	2
Coil power supplies	FMEA	6
Electrical power network	IE list	3
Neutral beam	FMEA	11
Hot cell	FMEA	6

1. IE list Initiating Event list (by informal method)
FMEA Failure Modes and Effects Analysis

Each PIE was assessed to determine event sequences which may result from it, and the possible consequences. A selection of PIEs was made for development using event trees to depict event sequences in which successive lines of defence of systems and equipment providing the required safety function were considered to fail. These show the potential hazardous consequence of each sequence, for example the release of tritium, tokamak dust or activated corrosion products, but in many cases the sequence frequency is extremely low, in the 'hypothetical' range.

5.5.1.4 Top-down Study

One of the motivations for performing a top-down study, to complement the already voluminous bottom-up studies, was that it can provide a clarity of presentation. Another benefit of this approach is to ensure completeness in identifying hazardous situations that could lead to the event of unacceptable radioactive release to the environment. By expanding this unacceptable event into all possible categories of events that could cause it to occur, the same initiators should be identified as with the bottom-up approach, ensuring the consistency and completeness of the analysis.

In order to start with a plant-level, safety functional approach, and work down to initiating faults, a global fault tree is required. To implement this, a master logic diagram (MLD) has been developed.

The top levels of the MLD are shown in Figure 5.5.1-1, and as an example the expansion of one branch in Figure 5.5.1-2, this representing the release of tritium from plasma-facing components. The top-level event in the MLD is 'excessive off-site release', meaning the release of radiological materials which exceed the project release guidelines. Subsequent levels in the diagram show the confinement barriers and supporting safety functions that protect against the release, and finally the potential initiators of accident sequences. Several failures must occur to give rise to a release based on the logic of the ITER MLD, indicated by the Boolean “AND” gates in the diagram (for clarity, "AND" gates are filled black in the diagram, "OR" gates are not). For example in Figure 5.5.1-2, an ingress of coolant into the vacuum vessel will create the conditions in which mobilisation of tritium could occur, but a large release to the environment would only result if, additionally, other failures occurred resulting in loss of confinement.

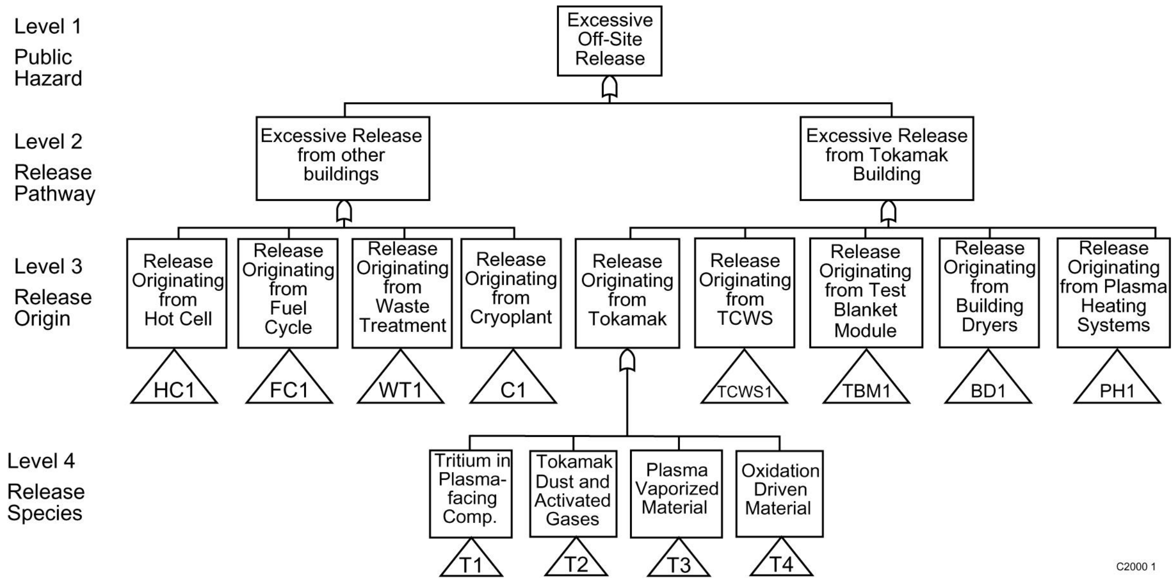

Figure 5.5.1-1 Top Levels of the Master Logic Diagram

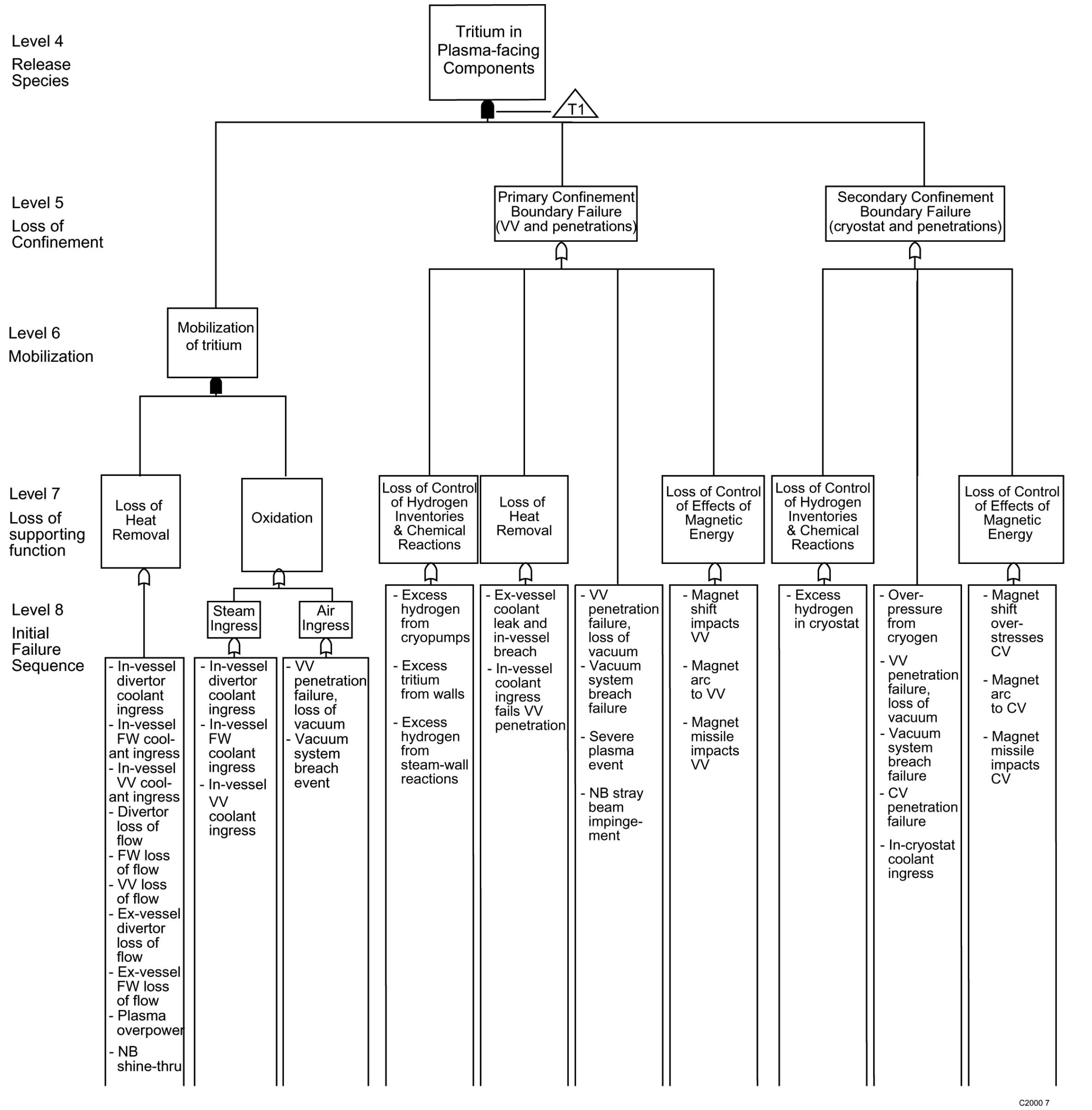

Figure 5.5.1-2 Branch of the Master Logic Diagram Representing the Release of Tritium from Plasma-Facing Components

As with the bottom-up studies, the initiating events can then be developed into sequences. The event sequence development leading to environmental releases is more straightforward in the top-down approach because all failures leading to environmental releases have been identified during the process of identifying initiating events.

5.5.2 Acceptance Criteria

Project release guidelines for tritium and activation products, which are the top level acceptance criteria used, are given in Table 5.2.3-2. The project imposed restrictive release guidelines so that the design would not have to be significantly modified after site selection. They will be reviewed after a site for ITER is chosen.

Other acceptance criteria adopted for these analyses are:

- maximum pressure inside the vacuum vessel is below 200 kPa (absolute);
- stresses in the cryostat are below allowable (maximum pressure below about 300 kPa (absolute));
- maximum containment volume pressure is below 200 kPa (absolute);
- flammable mixtures of hydrogen and air are avoided; the following acceptance criteria quantify this further:
  - hydrogen production inside the vacuum vessel is below 4 kg;
  - maximum first wall beryllium short-term (< 1h) temperature during plasma burn (excluding disruption) is < 700°C to avoid excessive hydrogen formation in case of beryllium/steam reactions if a water ingress into the vacuum vessel were to occur;
  - maximum long term (beyond one hour after shutdown) first wall beryllium temperature is < 385°C to avoid hydrogen formation in case of beryllium/steam reactions if a water ingress into the vacuum vessel were to occur;
- maximum temperature of components in casks once outside the vacuum vessel is 150°C to limit tritium outgassing;
- ozone formation in the cryostat is below 50 g to limit the explosive hazard.

5.5.3 Reference Events

The estimated releases are below the project release guidelines for all reference events, in most cases by orders of magnitude. The large margins are an indication of the favourable safety characteristics of ITER and the extent to which the safety approach has been successfully implemented in the design to further reduce the consequences of potential accidents to very low levels. Considering the wide breadth and depth of the cases analysed, the conclusion is drawn that there would be no significant risk to the public from accidental releases of radioactivity resulting from the operation of ITER.

Table 5.5.3-1 summarises important parameters and phenomena in the key event analysis. The key events in the table are those events where the fraction of release (a ratio of the predicted release to the release guideline) is larger than ~1% and events where the analysis demonstrates that confinement integrity (H_2 in vacuum vessel, cryostat pressure).

This section summarises the results of the analysis for the most important PIE families shown in Table 5.5.1-1.

5.5.3.1 <u>In-vessel Coolant Leak</u>

A family of potential accidents is grouped around the ITER cooling systems. Sequences starting with coolant pipe failures in various cooling loops and locations have been investigated and show the high degree of robustness in the design.

The thermal loads of a disruption of unexpectedly large magnitude can potentially threaten the integrity of the first wall and divertor cooling channels. Since the in-vessel components are not assigned a safety role, the consequences of a water ingress have been assessed. Different first wall damage sizes are investigated ranging from a single first wall coolant pipe failure (~ 1.6 cm^2) to multiple first wall pipe failure sizes up to 0.2 m^2, which is the maximum effective break size due to limited flow though upstream piping. This is done to show that the ITER design can accommodate the worst possible damage of in-vessel components

.

Table 5.5.3-1 Important Parameters and Phenomena in the Key Event Analysis

PIE Family	Important parameters	Important phenomena
In-vessel events In-vessel coolant leak	VV pressure and temperature	Effect of discharge of water from loop into the vacuum volume (flashing , impingement)
	Break water flow rate, loop water pressure and temperature	Break coolant flow to vacuum
	Flow rate into VPSSS, volume's pressure and temperature	Condensation in VVPSS
	In-vessel component's surface temperature	Heat transfer by radiation, conduction, convection in air/steam/vacuum
Loss of vacuum	Flow rate through the penetration line	Air exchange flow
	Dust/tritium concentration	Dust/tritium mobilisation and transport
	Hydrogen production rate at Be/air reaction	Be/air reaction
	In-vessel component surface temperature, VV pressure (long term)	Radiation, convection and conduction in air
Ex-vessel HTS events Large DV ex-vessel coolant pipe break	HTS Vault pressure and temperature	Effect of discharge of water from loop into volume with air
	Break flow rate, loop pressure and temperature	Break flow to volume with air
	Flow rate through the failed DV loop between VV and HTS vault	Air/steam exchange flow
	Dust/tritium concentration	Dust/tritium mobilisation and transport
Heat exchanger leakage	Leak rate through HX tube cracks	Assumption (assumption based on fission industry experience)
Maintenance Stuck DV cassette	Component's temperature due to decay heat	Radiation, convection and conduction in air
	Aerosol/tritium concentrations	Aerosol/tritium mobilisation and transport
Tritium system Isotope separation system failure	Tritium inventory ISS system pressure Leak rate	Tritium transport
Failure of fuelling line	Fuelling rate, isolation time	Tritium transport
Cryostat event Cryostat water and helium ingress	Cryostat pressure and atmosphere temperature	Heat transfer by radiation, conduction, convection in air/steam/vacuum
	Magnet structures and thermal shield temperatures; condensed (frozen) water masses	Effect of discharge of water and liquid He into the cryostat volume (flashing, condensation)
	Break water flow rate, loop's water pressure and temperature	Steam condensation on cryogenic surface; ice formation in presence of He
Hot Cell event	Component's temperature due to decay heat	Radiation, convection and conduction in air
	Atmosphere temperature in Hot Cell	Radiation, convection and conduction in air/wall
	Aerosol/tritium concentrations	Aerosol/tritium mobilization and transport

While it is assumed that such events could cause substantial damage to components in the vacuum vessel, the vacuum vessel pressure remains well below 0.2 MPa and any off-site releases remain very small. This is because part of the tritium and most of the corrosion products and dust will enter into the coolant leaked into the vacuum vessel and remain in the liquid pool in the pressure suppression system. All releases remain more than two orders of magnitude below the project release guidelines for incidents in Table 5.2.3-2.

5.5.3.2 Ex-vessel Coolant Leakage

A range of postulated coolant leaks outside of the vacuum vessel was investigated to ensure that the consequences and releases from such events will be acceptably small. The most severe case is discussed below.

During plasma operation, a double-ended pipe rupture of the largest pipe (0.4 m^2 flow area) is postulated to occur in the ex-vessel section of a DIV/LIM PHTS coolant loop. Coolant is discharged at a high rate into the containment volume. The fusion power is terminated to limit heat up of in-vessel components either passively by tiles that detach above the bond melting temperature or by an active system that injects impurities and uses signals from TCWS parameters. Even if the plasma is terminated, there is a possibility that the in-vessel cooling channels could be damaged because of a disruption that could follow the abnormal plasma termination. Since in-vessel components are not assigned a safety role, an in-vessel water ingress is also postulated to occur to demonstrate that any off-site consequences would be below the below the project release guidelines.

Only small amounts of hydrogen are generated by steam reacting with the uncooled plasma-facing components (PFCs) because the maximum first wall temperature always stays below 350°C. The containment volume pressure remains below 0.2 MPa. Failure of cooling channels in PFCs would cause pressurisation within the vacuum vessel. The vacuum vessel pressure peaks at 150 kPa, which is the setpoint for opening the pressure relief lines connecting the vacuum vessel and the pressure suppression system. Some fraction of the in-vessel inventories is transported to the heat transfer system vault by thermally driven exchange flow through the failed divertor loop, although most remains in the water pool of the suppression system or in condensed steam in the vault. The corrosion products, dust and tritium in the vault may leak to the environment before the vault pressure is restored to sub-atmospheric pressure within 24 hours after the event. Releases are about a factor of three below the project release guidelines for Accidents.

5.5.3.3 Loss of Vacuum

Although vacuum vessel penetrations are designed with care to provide two confinement barriers, the large number of these penetrations suggests that failure of a penetration line should be investigated to demonstrate the tolerance of the design to such failures. A loss of vacuum event results from this. Failure of windows/valves in a vacuum vessel penetration line (0.02 m^2 cross-sectional area) was selected to encompass all kinds of postulated loss of vacuum events. The penetration line is assumed in the analysis to be connected to the gallery with air atmosphere. Air ingress into the plasma chamber terminates the plasma with a disruption. Loss of off-site power is also assumed to coincide with the initiating event and last one hour.

The vacuum vessel and room pressures equalise about 25 minutes after event initiation. The air in the vacuum vessel heats up but stays below 200°C during the event. Chemical reactions do not occur due to the limited temperatures. In-vessel tritium and dust are mobilised by the air ingress, and some of them are transported to the vacuum vessel pressure suppression system. No mobilised radioactivity is transported out of the vacuum vessel due to the operation of the tokamak venting system, which pumps out the air in vacuum vessel through the normal vent detritiation system to prevent a back flow. Environmental releases are below project release guideline for Accidents by a factor of eight.

5.5.3.4 Tritium Process Pipe Leakage

Failures of the tritium processing equipment have been investigated because of the large tritium inventory in these systems. In such an event, the process gas will be released into the secondary confinement if the primary system pressure is higher than the confinement pressure, or the confinement atmosphere will equalise inside of the primary component if initial pressure is lower. There would be no flammable hydrogen mixtures since the secondary boundary will either contain an inert gas or be evacuated. The tritium release is the total of tritium losses from the secondary confinement (e.g. glove box) to the process room and the releases through the glove box atmosphere detritiation system in series with the vent detritiation system. The expected tritium atmospheric release is below the expected daily normal effluents.

The extremely unlikely event of failures in both the process and secondary confinement piping was postulated to ensure the fault tolerance of the design. The bounding case is a failure of the fuelling header between the tritium plant and the fuelling system valve boxes in the tokamak building. Tritium will leak into a room in the tritium building. The room will be isolated from normal ventilation and connected to the standby vent detritiation system (S-VDS). A time delay of 30 between detection and isolation was assumed. The total tritium content of the room is about 13 g at 30 s when the header is isolated from the pump for the bounding case. The tritium inside the room atmosphere is detritiated by the S-VDS. Instrumentation and control systems are provided with un-interruptible power, and the S–VDS can be supplied with emergency class III power that would be available within 30 seconds if a loss of off-site electrical power were also to occur. The total expected tritium release is about 0.17 g, which is one order of magnitude below project release guidelines for Accidents in Table 5.2.3-2. No flammable hydrogen-air mixtures are formed in the room.

Failure of the cryogenic distillation columns and helium coolant lines in the isotope separation system was examined since it has the highest tritium inventory (~220 g). Pessimistic assumptions for initial conditions and confinement equipment performance are used, and additional independent failures are postulated to demonstrate the tolerance to failure of the design. The tritium inventory spilled into its secondary confinement leads to a small tritium release to the environment (~ 54 mg), which is two orders of magnitude below project release guidelines for accidents. Releases from the isotope separation system are contained within secondary confinement and no flammable hydrogen-air mixtures are formed.

5.5.3.5 Magnets

The magnets are designed to ensure that the energy stored in them is controlled and off-normal events in the magnet system do not jeopardise confinement barriers. However, events that could potentially damage confinement barriers are examined in section 2.1 because of the large energy stored in the toroidal field coils (~ 40 GJ) and the geometrical location of the magnets near confinement barriers (vacuum vessel and cryostat). Pessimistic assumptions are made throughout to construct bounding scenarios to demonstrate that faults in the magnet systems do not create safety concerns.

One event postulated is a full terminal short of a TF coil, which is a extremely unlikely event that requires two ground faults in the coil busbar circuit, one on each side of the TF coil, while undergoing a fast discharge, plus the failure of the monitoring systems to detect these faults. This event is analysed to investigate the potential for structural damage to the confinement barriers. The maximum current induced in a shorted TF coil during a fast discharge is limited by both inductive coupling with the other coils during the fast discharge, and by the quench as the superconductor is driven to critical conditions. Extensive local plastic deformation can be expected to occur in the TF case (in the shorted coil and the adjacent coils) and intercoil structures. There may be some shear key and bolt failure and a loss of cryostat vacuum due to thermal shield damage, but gross structural failure is not predicted. The deflection of magnets is limited, and they do not touch the vacuum vessel or cryostat wall (or their thermal shields). There may be damage to the CS coil possibly leading to limited short circuits but arcing is confined to the coil. No radiological consequences are expected.

Another event postulated is an arc inside a coil. The arc is postulated to develop as a result of a failure (or more probably an inability) to discharge the coil when a quench occurs. The effects of arcs in the TF coils is contained within the massive coil case. External shorts on the CS and PF busbars (superconducting or normal) can potentially lead to molten material generation in the coils themselves (not significantly in the busbars) due to the coupling of extra energy into the coil, followed by a quench in a coil that cannot be discharged. In only three of the outer PF coil winding packs will the quench propagate slowly and local conductor melting, followed by the development of arcs, be possible. The melted material produced by the coil internal arcs (about 750 kg) may not be contained by the thin coil casing and would be spread over components in the cryostat in the vicinity of the shorted coil. Due to arc movement in the winding pack, the melted material will be distributed around the circumference of the coil, covering several square metres. It is possible that external arc energy associated with the coil short is sufficient to melt the conductor of the superconducting busbars (depending on the location of the initial short and the action of the coil power supplies) and cause local melting around the cryostat feedthroughs No failure of primary confinement barriers is predicted and no radiological consequences are expected.

A combined leak of helium and primary coolant into the cryostat was analysed to bound potential damage from the magnet system. The maximum pressure reached in the cryostat depends on the number of helium and HTS loops postulated to be damaged. In the bounding case, the cryostat pressure reaches a maximum value of 220 kPa in 82 s. All most steam condenses on cold surfaces in about 200 seconds after initiating the event. After 1,000 s, the pressure drops to 150 kPa due to cooling by cold surfaces such as the TF coil cases. Pumping of the cryostat volumes can restore sub-atmospheric conditions within 24 hours. Due to the condensation, the environmental releases are about three orders of magnitude below project release guidelines for Accidents in Table 5.2.3-2.

5.5.4 External Hazards

In addition to the reference events that envelope the safety consequences of internal failures, an assessment has been made to show that ITER is adequately designed against site hazards consistent with the ITER generic site requirements and site design assumptions, specifically earthquakes. An assessment of site-specific natural and man-made external hazards will be carried out following site selection to confirm the ability to perform the required safety functions.

An infrequent, severe earthquake (SL-2) which, although unlikely to occur during the lifetime of the facility, is assessed to demonstrate adequate protection of the public. This earthquake is assumed to have a peak ground acceleration of 0.2 g horizontal and vertical with a return period of ≥ 10,000 years. The design requirement is to ensure that required safety functions are provided and prevent releases from the facility.

Analysis of the tokamak and tokamak building have been carried out using the response spectra based on the assumed design basis ground motions. A 3D finite element shell-beam model has been built to perform seismic analysis as described in section 2.12. There is a high probability that a seismic event that leads to in-vessel motion would trigger a disruption, for example, due to dust falling into the plasma, loss of magnetic signals, false (motion induced) signals, or loss of plasma control. Control failures under some circumstances may lead to VDEs which in turn lead to higher loads on structures. For these reasons, a disruption or VDE load is assumed in addition to that from the earthquake.

The first horizontal eigenvalue of 2.8 Hz corresponds to the entire tokamak horizontal rocking on top of the gravity support. The vacuum vessel-magnet horizontal mutual rocking eigenvalue is 6.6 Hz. Displacements obtained from the model are relatively small and within the 10 mm radial build value allocated. The computed loads at the gravity support and vacuum vessel support under deadweight loads and SL-2 seismic loads are within acceptable limits.

A thorough seismic safety assessment will be undertaken as part of the site-specific adaptation, based on site-specific seismic hazards, final design and layout of the equipment. The preliminary seismic safety assessment demonstrates that ITER can be adequately designed for earthquakes. A system-by-system review of the consequences of a SL-2 level earthquake found possible releases that would not exceed release guidelines for Incidents in Table 5.2.3-2.

Seismically-qualified systems will survive, and their safety function will not be impaired. In particular, there will be no TCWS pipe ruptures (although some minor leakage may occur). Adequate radioactive confinement barriers will remain intact and sufficient heating, ventilation and air conditioning systems are qualified to ensure control of releases, if any.

It is concluded that ITER can be designed such that postulated earthquakes would not cause releases. Design changes of the tokamak building can be introduced to accommodate seismic loading up to 0.4 g, if judged necessary for a site specific design.

5.5.5 Ultimate Safety Margins

ITER is expected to be 'safe' with little dependence on dedicated 'safety systems' for public protection because of the fail-safe nature of the fusion energy reaction, modest mobilisable radioactive inventories, multiple layers of confinement, and passive means for decay heat removal. However, the inventories are large enough to warrant a detailed assessment of events that are very low in probability, so called hypothetical events, to demonstrate the tolerance to failures in the design and the lack of sharp increases in consequences if further degradation is postulated ('cliff-edge effects').

Hypothetical scenarios are developed by considering reference events (section 5.5.4) and postulating additional independent failures. Some bounding cases are analysed for safety functions, such as decay heat removal without any coolant or failure of all active plasma shutdown. For confinement, scenarios are investigated to show that the internal energy sources do not have the capacity to fail confinement and that enough margin is built into confinement to exclude cliff-edge effects.

Analysis of hypothetical events shows that the ITER design has sufficient safety margins to provide a high level of public protection and to meet the 50 mSv no-evacuation criterion.

Releases for the events considered are summarised in Table 5.5.5-2. The most limiting cases are the confinement bypass events. Releases in these events are below or near 20 g of tritium which is roughly 25% of the no-evacuation threshold. Significant environmental releases can only occur in ITER if several confinement barriers fail independently of each other. So far, no single event has been identified which can simultaneously damage the multiple layers of confinement in ITER.

The absence of cliff edge effects and robustness of the defence-in-depth approach are demonstrated in the postulated confinement bypass events which lead to the most limiting releases. In hypothetical confinement bypass events in the fuel cycle (where failure of two barriers is postulated) the amount of tritium released into the last confinement room is limited to less than 14 g tritium by process isolation. Postulating failure of room isolation would lead to an environmental release of this tritium.

Another extreme situation postulates common cause failure in vacuum vessel penetrations which would lead to significant transport of radioactivity into connecting rooms. Use of detritiation systems would limit environmental releases for these events. The largest environmental releases are expected for vacuum vessel bypass events which are combined with an extended blackout situation lasting one shift (8 hours). Postulating such a situation in combination with an in-vessel loss of cooling event would lead to environmental releases of about 20 g tritium and about 15 g of activated tungsten dust. Condensation inside the connected room plays an important role in limiting the environmental release of tritiated water. In a dry bypass event less radioactivity is transported from the VV to the connected room because there is no driving pressure pushing mobilized tritium and dust out of the VV.

Another hypothetical event is the postulation of failure of all cooling systems in ITER when removing decay heat. Even for this extreme situation the first wall only heats up to a maximum temperature of about 650ºC.

Postulating damage of both the vacuum vessel and cryostat boundary by some unidentified magnet energy release event would not lead to large environmental releases because the cryogenic surfaces would effectively capture most of the mobilized source term.

Table 5.5.5-2 Summary of Releases for Hypothetical Events

Event (Type of Release)	% of No-evacuation Threshold Average weather	% of No-evacuation Threshold Conservative weather
Wet bypass	< 1	<25
Dry bypass - loss of vacuum accident	< 1	<10
Tritium plant, fuel cycle	~ 1	15
Decay heat	< 1	< 1
Ex-vessel/In-vessel loss of coolant	< 1	< 1
Magnet energy release	< 1	< 1

Safety margins are maintained in ITER for these hypothetical events because of the following.

- Structural margins in the design of the vacuum vessel maintain integrity for the range of potential in-vessel coolant leaks from one cooling channel to a hypothetical break of all in-vessel pipes.

- The structure is robust such that there will be no catastrophic failure (or gross loss of geometry) due to plasma or magnetic forces.
- Inherent passive shutdown of the plasma occurs when the first wall reaches 700°C - 1100°C due to mechanical failure or beryllium evaporation, and intrinsic fail safe termination of the plasma occurs in case of in-vessel water or air leaks.
- Mobilisable radioactive inventories of tritium and activation products are modest. Therefore, ultimate performance of confinement barriers that must be assured in hypothetical accidents is not very stringent - about one order of magnitude release reduction for tritium and mobilisable metallic dust.
- Releases from the vacuum vessel even in the case of an in-vessel coolant leak with a bypass of the confinement barriers are limited due to limited inventories, isolation of the affected room, filtration and detritiation of leaks, and operation of the vacuum vessel pressure suppression system.
- A passive heat removal system (the VV HTS) exists.
- Long time scales are required for component heat-up following events because of heat removal by radiation from in-vessel components to the vacuum vessel and by natural circulation of the vacuum vessel heat transfer system.
- The decay heat density is low enough not to melt the structure even if all the heat transfer systems including the VV HTS are hypothetically assumed to be lost. Injection of helium (or air) into the cryostat within a few days will bridge the thermal isolation between vacuum vessel and the huge heat sink of the cryogenic structures of the magnet system. The maximum temperatures in such a case are low (< 650°C).
- The design is tolerant of failure of mitigating systems, such as failure of penetration line isolation and failure of the fusion power shutdown system.