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Health Physics Considerations for the ITER

J. J. Bevelacqua
Bevelacqua Resources, 343 Adair Drive, Richland, WA 99352 USA
bevelresou@aol.com

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Abstract
Health physics aspects associated with operation of the ITER are investigated. ITER has health physics elements in common with existing facilities as well as unique features. Fusion facility neutron radiation is similar to that encountered in a low energy accelerator. The tritium hazard has radiation protection issues in common with the Canadian Deuterium reactor. Completion of the final ITER design, its performance characteristics, component lifetimes, and maintenance requirements will determine the actual health physics hazards.

Keywords
ALARA, fusion energy, ITER, fusion radionuclides

1.0 Introduction

The ITER initially named the International Thermonuclear Experimental Reactor is an international project organized to build and operate the largest and most advanced experimental tokamak fusion reactor [1-13]. It is under construction at Cadarache, France. ITER has a goal of transitioning plasma production from experimental status to a full scale 500 MW fusion power facility. The project's members are the United States, European Union, Japan, China, South Korea, India, and Russia, and 34 countries support this enterprise.

Given the significance of the ITER and its potential for advancing power production technology, an investigation of its specific health physics characteristics is warranted. This investigation also has merit since the ITER has a defined technical basis and facility design, and is currently acknowledged as the facility that will first demonstrate fusion power operations [2,5,8,9,11-13].

This paper investigates the health physics aspects of the ITER facility. Since the ITER has a specific design basis, its health physics elements are explicitly addressed. A general discussion of various health physics aspects of fusion power facilities have been addressed by a number of authors [1,2,14-31] but a comprehensive health physics treatment of ITER has yet to be published .

A specific design basis also permits a comparison of ITER to commercial fission power facilities. This paper compares the radionuclides of interest, operational considerations, ALARA characteristics, design basis accidents, and anticipated radiation levels in conventional fission power plants and the ITER. Additional discussion regarding the unique aspects of ITER and its features in common with contemporary facilities are also provided.

The radiological hazards associated with a fusion power facility, anticipated sources of radiation exposure from this facility, and possible ALARA measures to reduce the occupational doses are reviewed. This paper also reviews the basic physics principles and relationships that govern the fusion process. These relationships define the basic plasma properties, govern the nuclides interacting to form the plasma, and determine their energy. These radiation types and associated energies significantly influence the health physics characteristics of the ITER.

2.0 Overview of Fusion Power Production

Fusion energy offers the potential for cheap, clean, and abundant energy. It also offers a number of significant advantages when compared with fission technology. In particular, fusion facilities do not encounter many of the issues associated with fission reactors including reactor safety, high-level waste generation, storage of spent reactor fuel, vulnerability to terrorism, and nuclear proliferation [14,25,29-31]. These factors offer considerable motivation for replacing fission reactors with fusion reactors once a viable fusion power design is defined, tested, and licensed. ITER offers the potential for verifying a scalable fusion power facility design.

At the ITER, the fusion reaction or process occurs within a plasma composed of deuterium and tritium nuclei. The ITER uses magnetic confinement to facilitate the fusion process. Magnetic confinement has been the best option for a fusion facility, but recent success at the National Ignition Facility at the Lawrence Livermore National Laboratory is renewing interest in inertial confinement [6,32]. The fusion confinement method influences the radiation types and fuel materials that must be controlled by the health physicist.

The ITER D-T fusion processes differ substantially from those encountered in fission reactors because it does not produce actinides or fission products (e.g., radioactive isotopes of iodine, cesium, strontium or noble gases). ITER does produce a variety of fusion and activation products and these products depend on the selected fusion process, the reaction energies, and the materials of construction selected for the facility. Fusion and activation products present a challenge for the health physicist responsible for worker radiation protection. Both internal and external radiation challenges are present. Since ITER uses a D-T process, tritium fuel material presents an internal hazard in its initial state prior to introduction into the reactor.

2.1 Fusion Process Candidates

Fusion involves the interaction of two light systems to form a heavier system. A variety of fusion processes are possible and could be applicable to power production. The term fusion process is used instead of fusion reaction because a fusion event that is used to produce power involves not only the reaction of individual light ions, but also their density, confinement time, mode of confinement, presence of other plasma constituents that inhibit or catalyze the light ion fusion, and methods to initiate, sustain, and energize the plasma. The term reaction is reserved for specific nuclear events [e.g., (n, γ), (γ, n), (n, α), (n, p), (n, 2n), and (n, 2n α)] that result from the fusion process.

Candidate fusion power processes for the ITER include the following reactions [2,8,9,25,26,29-31]:

$$\mathbf{D+D \xrightarrow{50\%} T(1.01\ MeV)+p(3.02\ MeV)} \qquad (1)$$

$$\mathbf{\xrightarrow{50\%} {}^{3}He(0.82\ MeV)+n(2.45\ MeV)} \qquad (2)$$

$$\mathbf{D+T \rightarrow {}^{4}He(3.50\ MeV)+n(14.1\ MeV)} \qquad (3)$$

where D is deuterium (^{2}H) and T is tritium (^{3}H). For consistency with the literature, D and ^{2}H and T and ^{3}H are used interchangeably. The particle energy at the reaction threshold is provided in parenthesis.

At ITER, fusion is initially realized through the D-T process. At a later stage, fusion involving only deuterium nuclei may become more important. However, the D-D process is somewhat more difficult to achieve because its inclusive cross sections are smaller in magnitude and higher densities are required to initiate and sustain the D-D fusion process [14,25,28-31,33,34].

In the D-D fusion process, the tritium nucleus formed in Eq. 1 subsequently fuses through the D-T process, Eq. 3. An advantage of the D-D process is that it avoids the need for a tritium fuel source, which eliminates a significant health physics hazard. However, it is unlikely that the conditions for D-D fusion will be realized on a practical scale before D-T fusion. Therefore, the subsequent discussion focuses on the D-T process at the ITER facility.

The D-T fusion energy output is about 94 x 10^{6} kWh per kilogram of a mixture of deuterium (0.4 kg) and tritium (0.6 kg). On a per mass basis, this is more than four times the energy released from fission [14].

The D-T fusion process occurs within a state of matter known as plasma. Before proceeding further, it is necessary to define the forces governing the plasma as well as the characteristics and properties of plasmas.

2.2 Plasma Properties and Characteristics

Plasma is often referred to as the fourth state of matter because it has unique properties. A plasma consists of a collection of atoms, ions, and electrons in which a large fraction of the atoms are ionized so that the electrons and ions are essentially free. Ionization occurs when the temperature or energy of the plasma reaches a threshold value characteristic of the plasma's initial atomic constituents.

The energy (E) of an ion or electron in the plasma is related to its absolute temperature (T):

$$\mathbf{E = kT} \qquad (4)$$

where k is Boltzmann's constant ($1.38\text{x}10^{-23}$ J/K). An ion with an energy of 1 eV corresponds to a temperature of:

$$\mathbf{T = \frac{E}{k} = \frac{(1eV)(1.6x10^{-19}\,J/eV)}{1.38x10^{-23}\,J/K} = 1.16x10^{4}\,K} \qquad (5)$$

Ionization is not the only process that occurs in a plasma. The ions and electrons also recombine. However, ionization dominates recombination for practical fusion plasmas. Plasmas are governed by the Maxwell equations [35]. Their electromagnetic and nuclear interactions generate radiation types that produce the relevant health physics hazards.

2.3 Plasma Confinement

Once the fusion process is selected, it will be necessary to confine the resulting plasma with a suitable physical mechanism. Two primary approaches are available for confining the charged particles forming the fusion plasma. These are magnetic confinement and inertial confinement [7,8,25,29-31]. ITER utilizes magnetic confinement.

Magnetic confinement is based on the force that a charged particle experiences in a magnetic field [35]. The magnetic force ($\vec{\mathbf{F}}_{\mathbf{mag}}$) is a component of the total electromagnetic force that serves to confine the charged particles comprising the plasma, and has the form:

$$\vec{\mathbf{F}}_{\mathbf{mag}} = \mathbf{q}\,\vec{\mathbf{v}} \times \vec{\mathbf{B}} \qquad (6)$$

The cross product is rewritten in terms of the angle (θ) between $\vec{\mathbf{v}}$ and $\vec{\mathbf{B}}$:

$$\vec{\mathbf{F}}_{\mathbf{mag}} = \mathbf{q}|\mathbf{v}||\mathbf{B}|\mathbf{Sin\theta} \qquad (7)$$

where q is the particle's electric charge, $\vec{\mathbf{v}}$ is its velocity, and $\vec{\mathbf{B}}$ is the magnetic induction [35]. Since the fusion plasma is composed of charged particles, it is possible to confine plasma using a magnetic field with the requisite topological characteristics.

In magnetic confinement D-T fusion, quantities of deuterium and tritium gas are maintained to initiate and sustain the process. The tritium gas (T_2) must be carefully monitored and controlled since it presents an internal intake concern. In addition, T_2 is readily converted into HTO that is considerably more radiotoxic [29-31].

Tritium gas fuel and HTO present a greater internal hazard than the solid D-T pellet used in an inertial confinement device. Therefore, unfused tritium fuel requires a cleanup/recovery system to minimize its health physics impact.

In magnetic confinement, the D-T plasma has a density (ρ) on the order of 10^{15} ions/cm^3 with a confinement time (τ) on the order of 0.1 s. For D-T fusion, the product of the density and confinement time must be at least 10^{14} ion-s/cm^3 to satisfy the Lawson criterion for energy break-even [14]. Break-even occurs when the energy input to establish and maintain the plasma equals the fusion energy output.

For the IC and MC fusion situations noted above, the effective dose (E) is proportional to the product of the plasma density and confinement time [20,24,29]:

$$\mathbf{E \propto \rho\tau} \qquad (8)$$

For MC fusion,

$$\begin{aligned}\mathbf{E_{MC}} &\propto \mathbf{(10^{15}\ ions/cm^3)(10^{-1}\ s)} \\ &\propto \mathbf{10^{14}\ ions\text{-}s/cm^3}\end{aligned} \qquad (9)$$

MC fusion results in nearly steady state operation in a production scale device.

3.0 Overview of the ITER

The International Thermonuclear Experimental Reactor is based on the tokamak design concept that utilizes D-T fusion [1-13]. The tokamak principle of magnetic confinement in a torus was developed in the former Soviet Union in the 1960s. The name tokamak is derived from the initial letters of the Russian words meaning: “toroidal”, “chamber”, and “magnetic”, respectively [29].

The goal of ITER is to achieve a self-sustaining reaction that relies on fusion heat without the need for external sources. When this occurs, the fusion process is controlled only by the rate of fuel addition to the torus.

The heart of the ITER’s magnetic confinement system is the torus that is a large vacuum vessel surrounded by devices to produce the confining magnetic field. Other major components of the ITER are the superconducting toroidal and poloidal magnetic field coils that confine, shape, and control the plasma inside the vacuum vessel [1-13]. The magnet system includes toroidal field (TF) coils, a central solenoid, external poloidal field coils, and correction coils. The vacuum vessel is a double-walled structure. Associated with the vacuum vessel are systems supporting plasma generation and control. These systems include the divertor system and blanket shield system.

The fusion process occurs within the vacuum vessel. Radiofrequency energy and ion beams heat the plasma in order to reach the fusion ignition temperature. In addition to its confinement function, the magnetic field is also designed to prevent plasma from striking the inner wall of the vacuum vessel. If the plasma strikes the vacuum vessel wall, material is removed and forms a particulate dispersed into the plasma and is activated. As a particulate, this activated material presents both an internal and external radiation hazard. From an ALARA perspective, the quantity of this material should be minimized.

Associated with the vacuum vessel is the divertor system. The major functions of this system are plasma power, particle exhaust, and impurity control. A secondary function of the divertor system is to provide vacuum vessel and field coil shielding. The plasma particle exhaust removes ^{4}He and other nuclei formed in the fusion process and through nuclear reactions.

A second system supporting vacuum vessel operations is the blanket shield system, and its major components are the first wall and blanket shield. The structure facing the plasma is referred to collectively as the first wall, and it is subdivided into a primary wall, limiters, and baffles.

The primary wall establishes the initial protection of components located beyond it. Limiters provide specific protection at distributed locations around the vacuum vessel. Baffles preserve the lower area of the machine close to the divertor from high thermal loads and other conditions created by the plasma.

The blanket shield supports the first wall by providing neutron shielding for the vacuum vessel. This shielding is a combination of stainless steel and water. The blanket also provides the capability for testing tritium breeding blanket modules and for tritium production blankets. ITER has not yet specified the specific requirements for these blankets.

Production scale facilities build upon the ITER experience, are physically larger, and have a higher power output. The ITER is a formidable structure with the main plasma parameters and dimensions provided in Table 1.

Table 1. ITER Plasma Parameters and Dimensions[a]

Total Fusion Power	500 – 700 MW
Plasma Major Radius	6.2 m
Plasma Minor Radius	2.0 m
Plasma Current	15 MA
Toroidal Field @ 6.2 m	5.3 T
Plasma Volume	837 m^3
Plasma Surface Area	678 m^2

[a] Ref. 2.

There are internal, replaceable components that reside inside the vacuum vessel. The components include blanket modules, port plugs such as the heating antennae, test blanket modules, and diagnostic modules. These components absorb heat as well as most of the plasma neutrons and protect the vacuum vessel and magnet coils from excessive radiation damage. The shielding blanket design does not preclude its replacement by a tritium-breeding blanket in subsequent ITER enhancements. A decision on incorporating a tritium-breeding blanket will be based on the availability of tritium fuel, its cost, the results of breeding-blanket testing, and acquired experience with plasma and machine performance.

The heat deposited in the internal components and in the vacuum vessel is removed using a tokamak cooling water system designed to minimize releases of tritium and activated corrosion products to the environment. The entire vacuum vessel is enclosed in a cryostat, with thermal shields located between hot components and the cryogenically cooled magnets.

The vacuum vessel fueling system is designed to inject gas and solid hydrogen pellets. During plasma start-up, low-density gaseous fuel is introduced into the vacuum vessel chamber by the gas injection system. The plasma is generated using electron-cyclotron-heating [35], and this phenomenon increases the plasma current. Once the operating current is reached, subsequent plasma fueling (gas or pellets) leads to a D-T process at the design power rating.

From a safety perspective, the design focuses on confinement with successive barriers provided for the control of tritium and activated material. These barriers include the vacuum vessel, the cryostat, air conditioning systems with detritiation capability, and filtering capability of the containment building. Effluents are filtered and detritiated such that releases of radioactive material to the environment are minimized.

Worker radiation safety and environmental protection are enhanced by the structure housing the vacuum vessel. For worker protection, a biological shield of borated concrete surrounds the cryostat and concrete walls provide additional neutron and gamma-ray shielding.

Accidental releases of tritium and activated material are minimized by engineered systems that maintain pressure differences to minimize any release of radioactive material. These systems are designed such that air only flows from lower to higher contamination areas. These differential pressure and airflow characteristics are maintained by the air conditioning system.

4.0 ITER Project Phases

The estimated cost and schedule for the ITER Project has grown since the initial estimate and the schedule has expanded [12]. In 2008, the first plasma was predicted for 2016 and that date has slipped to the 2020 – 2023 timeframe [12]. This schedule slippage is not unexpected for a project with significant scope, but it illustrates the uncertainties associated with the ITER Project. This project currently envisions four generic phases.

The first stage begins with a three-year period using only hydrogen fuel (^{1}H) at DT ignition temperatures. The hydrogen plasma permits testing of tokamak systems in a non-nuclear environment. Stage two is a one-year period of operation with deuterium. The power output from DD fusion is expected to be low. Nuclear operations with DD fuel test additional systems including the heat transport, tritium processing, and particle control systems. The third stage includes DT plasma operations with output power at or below 500 MW. At the end of this three-year phase, testing of Demonstration Fusion Reactor blanket assemblies is planned. In the fourth phase, DT operations focus upon improving DT fusion performance.

5.0 ITER Safety Characteristics

Although ITER is a prototype reactor, it exhibits the essential characteristics of a fusion power production facility using magnetic confinement. ITER also has favorable nuclear safety characteristics when compared to fission reactors [1-13,25,29-31]. As an example, a criticality accident cannot occur through the fusion process or through the interaction of any fusion products. In addition, fissile and fertile materials are neither utilized nor produced in the D-T fusion process. For comparable power ratings, the total energy inventory in the D-T fuel and D-T plasma are several orders of magnitude lower than in a commercial fission reactor core. This lower energy inventory inherently limits the extent of any offsite release of radioactive material.

Another positive benefit of ITER is the fact that the total D-T fuel inventory within the plasma containment vessel is small. If the inventory is not replenished, fusion is only sustained for about one minute. In addition, the reaction products of D-T fusion are a neutron and ^{4}He.

Fusion facility radioactivity inventory is minimized using low-activation materials. These materials reduce the overall radiological source term and minimize the quantity of radioactive waste resulting from fusion operations.

ITER also has positive operational safety characteristics. A low fusion power density and positive thermal characteristics facilitate a wide safety margin for response to a loss of fusion reactor cooling. The low fusion power density and large heat transfer area permit passive cooling of plasma facing components and breeder blankets if active reactor cooling is interrupted. However, the magnetic field energy associated with ITER has the potential to distort the tokamak structure and lead to a release of radioactive material.

6.0 General Radiological Characteristics

The ITER radiological hazards are representative of those occurring in a production scale fusion facility. These hazards include tritium, neutron radiation, activation products, and particulates generated by plasma collisions with containment structures [25,29-31].

Tritium in gaseous form (T_2) and as oxides (HTO, DTO, and T_2O) will be present at ITER. The particular chemical form depends on the location within the tritium processing system and the physical conditions encountered during a tritium release scenario.

Neutron radiation is produced in the D-T fusion process. The 14.1 MeV neutrons pose a direct radiation hazard, and have a significant potential for activating and damaging reactor components. The radiation damage increases maintenance requirements, radioactive waste generation, and occupational radiation doses.

Activation products of stainless steel and copper are large contributors to the radiological source term. At ITER, the most significant activation products of stainless steel are isotopes of Mn, Fe, Co, Ni, and Mo and the most significant activation products of copper are Cu, Co, and Zn. During ITER's Extended Performance Phase, a reactor inventory of approximately 10^{14} MBq is anticipated [1-13]. Smaller activation product inventories reside in structures outside the shield blanket or circulating as suspended corrosion products in the first wall, blanket, and divertor coolant streams.

These activation products and their activities present high radiation fields inside the cryostat and vacuum vessel. The radiation fields are sufficiently high to require remote maintenance for systems, structures, and components within the cryostat and vacuum vessel [1-13,20].

Fine particles are produced as a result of ion impacts with plasma facing components. These particles form a fine radioactive dust that could be released during maintenance inside the plasma chamber or during a severe accident.

Tritium, activation products, and toxic materials could be released during an accident or off-normal event. There are a number of ITER energy sources that facilitate the dispersal of radioactive and toxic material. These energy sources, possible consequences of their discharge, and control measures are summarized in Table 2.

7.0 Accident Scenarios/Design Basis Events

The safety characteristics, radiological characteristics, and energy sources form the basis for deriving ITER's accident scenarios. These accident scenarios are summarized in subsequent discussion and include loss of coolant accidents (LOCAs), loss of flow accidents (LOFAs), loss of vacuum accidents (LOVAs), plasma transients, magnet fault transients, loss of cryogen, tritium plant events, and auxiliary system faults. These scenarios form the foundation for ITER's design basis events. Although the physical processes differ, the fusion design basis events have similarities to fission power reactor events. Each of these events has health physics implications because their occurrence can lead to a release of radioactive material to the environment

7.1 Loss of Coolant Accidents

LOCAs are serious events in a fission reactor because the coolant removes heat from the fuel core [29-31,36]. In a fission reactor, a LOCA has the potential to damage the fuel/cladding fission product barrier. Any loss of coolant, increases the temperature of the fuel, increasesthe likelihood for fuel damage, and reduces the margin for protection of the fuel from

Table 2. ITER Energy Inventories[a]

Energy Source	Power or Energy	Release Time	Potential Consequences of Energy Discharge	Control Measures
Fusion Power	1.5 GW	1000 s (pulse duration)	Melting of plasma facing components In-vessel loss of coolant accidents Mobilization and release of tokamak, plasma facing component, and activation products.	Normal coolant systems operations Active power shutdown systems Passive shutdown for large disturbances
Plasma	2.3 GJ	< 1 s	Disruption/vertical displacement event Limited evaporation and release of plasma facing components	Plasma control systems Disruption/ vertical displacement event mitigation systems
Magnetic	120 GJ	s to min	Electric arcs Localized magnet melting Mechanical damage Release of radionuclide inventory near the arc	Normal operation of coolant systems Insulation design Rapid quench detection and discharge system
Decay Heat	260 GJ (In the first day) 910 GJ (In the first week)	min to y depending on the concern	Heating near plasma components and materials Maintenance concerns Waste management concerns Driving force to mobilize activation products	Minimization of decay heat production Defense in depth design features Normal cooling system operations Active decay heat removal systems Passive heat removal using radiative heat transfer and natural circulation cooling
Chemical (following a reaction)	800 GJ	s to h	Overheating of plasma facing components Hydrogen fires or explosions Overpressurization or damage of the plasma confinement structure Driving force to mobilize activation products	Using passive means, limit temperatures to about 500 ^{o}C on plasma facing components to minimize hydrogen production Design measures to prevent ozone accumulation
Coolant	300 GJ	s to min	Pressurization of vacuum vessel, cryostat, or heat transport system vault Pressurization creates a driving force for activation product mobilization	Overpressure suppression systems

[a] Ref. 1.

melting. From a health physics perspective, a LOCA results in the release of fission products to the facility and potentially to the environment.

At the ITER, LOCAs involve actively cooled components (e.g., blanket, shield, vacuum vessel, and divertor cooling system) that remove fusion energy [1-13]. Cooling media include water and helium. LOCAs at the ITER are divided into two broad categories (in-vessel and ex-vessel) [1-13].

An in-vessel LOCA diverts coolant into the vacuum vessel leading to pressurization or chemical reactions with hot plasma facing components (PFCs). Coolant entering the plasma chamber during plasma operations disrupts and extinguishes the plasma. However, pressure or chemically initiated events disperse radioactive material including tritium and activation products. The extent of the dispersal area and quantity of radioactive material dispersed depend on the specific fusion reactor design, its operational history and operating characteristics, and details of the accident sequence. Parameters that determine the severity of the LOCA include the type and quantity of fluid leaked into the vacuum vessel, the vacuum vessel volume, the internal energy of the fluid, and for water LOCAs, the presence of condensation surfaces.

Ex-vessel LOCAs involve piping runs to heat removal systems such as steam generators or heat exchangers. Since the ex-vessel piping has a larger bore than in-vessel piping, ex-vessel LOCAs involve larger volumes of coolant than in-vessel events. Rapid detection of an ex-vessel event is required to protect the divertor and first wall from overheating when coolant is lost. The time required for detection of the ex-vessel LOCA and for shutdown of the plasma reaction depends on the plasma facing component's heat load. For ITER, the time is on the order of seconds.

The probability of an ex-vessel LOCA is judged to be much lower than that of an in-vessel LOCA. This reduced probability is associated with the regularity of scheduled inspections of heat removal systems and associated piping [1,2,5].

LOCAs have very similar release consequences in fission power reactors and at ITER. Although the internal mechanisms generating the LOCA differ at these facilities, a resulting release of radioactive material occurs. The ITER LOCA has a significantly different source term since no fission products are released. Tritium and vessel activation products dominate ITER's release source term.

7.2 Loss of Flow Accidents

In a fission reactor, loss of flow events are less severe than LOCAs, and result from transients that limit the flow of cooling water to the core [29-31,36]. Pump failures or shutdowns, valve mispositioning, loss of motive force (steam or off-site power) or instrument failures initiate these events. Their severity depends on the duration of the event and integrity of the reactor coolant system piping.

At the ITER, LOFAs are predominantly caused by a loss of off-site power, which results in the decrease, or loss of coolant pump output [1,2,5]. LOFAs often lead to LOCAs because a loss of cooling flow can lead to tube overheating and subsequent tube failure if plasma shutdown is not achieved rapidly.

In-vessel LOFAs are induced by tube plugging or coolant system blockage. Since in-vessel components usually involve small diameter piping, an in-vessel LOFA leads to overheating and subsequent failure of the tube or channel and results in an in-vessel LOCA. After tube or channel failure, coolant is released into the plasma chamber with disruption and termination of the plasma. Following plasma termination, component cooling is required to prevent further damage that could result in a release of radioactive material.

The consequences of a LOFA depend on fusion process heat loads and the design of cooling systems to manage these heat loads. Therefore, an ITER LOFA's significance depends on the particular phase of the operating cycle. Key parameters that affect a LOFA are the coolant material, divertor heat load, first wall heat load, and the heat transport system design. Therefore, an ITER LOFA is most severe during the period of full power operations.

7.3 Loss of Vacuum Accidents

In a fission reactor, a loss of vacuum event typically involves the turbine's condenser. Fission turbines operate under vacuum conditions to facilitate steam transport. A loss of vacuum leads to a turbine trip and subsequent reactor trip. The safety significance of loss of condenser vacuum is significantly less that the hazard associated with a loss of coolant event [30,31,36].

In a fusion reactor, the plasma chamber is operated under vacuum conditions. When plasma chamber vacuum is lost, a loss of vacuum event occurs. Vacuum disruption is realized when a gas including air leaks into the plasma chamber. Disruption follows a component failure such as a diagnostic window, port, or seal, caused by a defect, vessel erosion, component wear, radiation damage, excessive load or overpressurization of the plasma chamber following an in-vessel LOCA [1,2,5]. In addition to allowing fluid ingress into the vessel, the component failure allows radioactive material (tritium or activated material) to escape from the vessel. If air enters the vacuum vessel, it reacts chemically with the hot plasma facing components. This interaction produces thermal energy that can volatilize additional radioactive material. The severity of a LOVA depends on the operating period of ITER and will be most severe during full power operation [1,2,5].

7.4 Plasma Transients

Over power transients occur in a fission reactor and are triggered by changes in reactor coolant temperature, secondary system transients, and secondary system failures including valve malfunctions. These transients are potentially severe and can lead to core damage with a subsequent release of fission products [29-31,36].

Plasma transients include overpower events and plasma disruptions. Overpower conditions occur in a plasma when the balance between fusion energy generation and energy loss is disrupted. When generation exceeds loss, an increase in temperature results until the accumulation of ^{4}He and depletion of D-T fuel occurs. After about 2 – 10 s, a disruption and plasma shutdown occurs [1,2,5]. Plasma disruptions include a variety of instability transients.

During a disruption, confinement of the plasma is lost, the fusion process terminates, and energy is rapidly transferred to the surrounding structures. This energy transfer induces PFC ablation and possibly melting. During this energy transfer, the plasma current quenches within about a second, and magnetic forces are exerted on the vessel and support structures.

Disruption can be induced by thermal excursions, impurities injected into the vacuum vessel, and loss of plasma control. These conditions are expected to occur during ITER power operations. In addition, plasma disruptions generate high-energy electrons that damage PFCs and initiate failure of first wall/blanket modules or segments. These failures liberate activation products and enhance the possibility of their release from the vacuum vessel.

7.5 Magnet Fault Transients

Electrical faults occur in power supplies, switchgear, turbine-generators, and other components. These faults include arcing and other electrical discharge mechanisms, which release energy. Severe transients in a fusion reactor have the potential to disrupt electrical power supplies. If the loss of power is extended, the event has the potential to mobilize radioactive material [1-13].

Magnetic field transients induce forces that can damage structural integrity and induce faults in other machine components. Off-normal forces yield large magnet coil displacements that affect other systems (e.g. the vacuum vessel and plasma heat transfer system piping) and produce arcs that induce localized component damage. At ITER, magnetic field transients could damage the vacuum vessel and its associated ducts and piping and the cryostat. This damage facilitates the release of radioactive material.

Electromagnetic forces also result from equipment and operational transients that lead to electrical shorts in coils, faults in the discharge system, or power supply faults. Electrical arcs between coils, to ground, and at open leads facilitate localized component melting. Arcs also arise from insulation faults, gas ingress, or overvoltage transients. The degree to which arcs or magnetic faults occur depends on the facility design and its operational characteristics. However, arcs damage structures and increase the potential for the release of radioactive material.

7.6 Loss of Cryogen

An extensive loss of helium or nitrogen cryogen is a radiological safety issue because the pressures developed following the leak are sufficient to breach confinement barriers [1,2,5]. The released helium and nitrogen also displace air and present a suffocation hazard.

Releases of helium and nitrogen result from component failures or transient conditions. For superconducting magnets, quenching the superconductor without electrical discharge causes helium leakage. Cryogen plant failures release nitrogen gas following volatilization of the liquid phase. These gas releases also provide a mode of force to mobilize radioactive material.

There are no comparable fission reactor events involving cryogen. Some radiation detection instrumentation requires cooling, but the loss of cryogen does not lead to the release of radioactive material [29-31].

7.7 Tritium Plant Events

Although tritium is not a material used in a fission reactor, it is produced during power operations from activation of the reactor coolant via $^{2}H(n, \gamma)^{3}H$, chemical agents including boric acid ($^{10}B(n, 2\alpha)^{3}H$) and lithium hydroxide ($^{6}Li(n, \alpha)^{3}H$), and tertiary fission. Tritium is monitored and controlled to minimize the potential for internal impacts [29-31].

Hydrogen is used in fission reactors for primary system oxygen control and for its thermal properties in turbine generator systems. Any leakage of hydrogen gas creates the potential for an explosive mixture and hydrogen monitors are included in the design to minimize these events. As demonstrated by the Fukushima Daiichi accident, hydrogen explosions have the potential to disperse fission products [36].

Breaching confinement barriers of the tritium processing and fueling system releases a variety of chemical forms (e.g., T_2 and HTO). Tritium release events should also consider the potential for hydrogen explosions. However, tritium design standards normally require double or triple containment for systems containing hydrogen. These standards should reduce the frequency of large release and explosion events [1-13,37,38].

An explosion provides a potent force to disperse radioactive material. The specific plant location of the explosion and its magnitude govern the quantity of radioactive material dispersed and the severity of the event. However, tritium releases lead to potential internal intakes that become more severe as tritium gas is converted to the HTO form.

8.0 Radioactive Source Term

The aforementioned ITER accident scenarios have the potential to release radioactive material within plant structures and the environment [1,2,5]. The extent of the release depends on the available radioactive material and the plant conditions. Table 3 summarizes the inventories and the release assumptions currently used in evaluating the consequences of the postulated ITER events. A more complete discussion of the assumptions associated with ITER radioactive material dispersal events is provided in subsequent discussion.

The tolerable release fraction listed in Table 3 is based on a 50 mSv effective dose during the release period plus 7 days considering no evacuation, average meteorology, and ground level release conditions [1]. Table 3 and its associated data are unique to the ITER design and could change as operational experience is incorporated into a production scale facility.

9.0 Beyond Design Basis Events

ITER's beyond design basis events (BDBEs) have frequencies of $<10^{-6}$/y. BDBEs include vacuum vessel collapse, magnet structure collapse or movement, and building structural failure. Collapse of the vacuum vessel, collapse of magnet structural supports, or movement of magnet structural supports sever tokamak coolant lines and damage one or more of the tokamak confinement barriers. Gross building failure also damages tokamak coolant lines and structural barriers and leads to fire related events. All of these events have the potential for a significant release of radioactive material [1].

10.0 Assumptions for Evaluating the Consequences of Postulated ITER Events

The ITER adopted a standard set of assumptions for evaluating the consequences of the postulated design basis events [1]. This set provides common assumptions for comparing

Table 3. Radioactive Material Inventories in Postulated ITER Accident Events[a]

Source Term	Inventory Available for Release	Tolerable Release Fraction	Control or Mitigation Strategy
In-vessel tritium as a co-deposited carbon-hydrogen layer	1 kg-tritium	≈30% if HTO	Administrative limit and surveillance on layer buildup Dual confinement barriers against air ingress
In-vessel tritium diffusively held in beryllium and tritium in cryopumps	0.7 kg-tritium	>100% if HT	Limit first wall temperatures to 500-600 ^oC
In-vessel tokamak dust (e.g., steel and tungsten), excluding beryllium and carbon	20 kg metal	≈30%	Administrative limit and surveillance on dust Dual confinement barriers against air ingress
Oxidation-driven volatility of in-vessel steel, copper, and tungsten	Kilograms of solid near-plasma material	≈10 - 100% depending on temperature	Limit first wall temperatures to 500-600 ^oC
Tritium plant circulating inventory	600 g-tritium	≈75%	Administrative limit and surveillance on inventories Confinement barriers Tritium Plant Building structural integrity
Secure tritium storage	1 kg-tritium	≈50%	Administrative limit and surveillance on inventories Confinement barriers Tritium Storage Building structural integrity
Hot cells, waste storage	<1 kg-tritium Kilograms of activated metal	≈50% for tritium ≈10% for dust	Administrative control and surveillance on tritium and dust Recycle tritium Temperature limits and controls Confinement barriers

[a] Ref. 1.

Table 4. Public Dose Criteria for Postulated ITER Events[a]

Accident Consideration or Parameter	Operational Events	Likely Events	Unlikely Events	Extremely Unlikely Events	Hypothetical Events
Annual Expected Frequency (f)	Expected to occur	>0.01/y	$0.01/y \geq f > 10^{-4}/y$	$10^{-4}/y \geq f > 10^{-6}/y$	$\leq 10^{-6}/y$
ITER Objective	Apply ALARA principles	Avoid releases	Avoid the need for any public countermeasures	Avoid the potential for public evacuation	Limit risk [e]
Dose criteria to meet the design basis [b]	0.1 mSv/y chronic dose (all pathways integrated over all operational event categories)[c]	0.1 mSv/y chronic dose (all pathways integrated over all likely event categories)[c] 0.1 mSv/y chronic dose (without ingestion) [c]	5 mSv/event chronic dose (without ingestion)	5 – 50 mSv/event [d]	Limit risk [e]

a Ref. 1.
b When dose criteria are "per year", average annual meteorology is assumed; when "per event", baseline worst-case meteorology is used for design basis events and average meteorology is used for beyond design basis events.
c The summation of the operational event dose and likely event doses must be ≤0.2 mSv/y.
d The range for extremely unlikely events results from the variation among various national dose criteria. For design purposes, a value of 10 mSv during the release period plus 7 days is utilized by ITER.
e The goal is to limit risk. In addition to meeting the extremely unlikely event dose criteria, the no evacuation goal implies the need to limit doses to the local population to approximately 50 mSv/event during the release period plus 7 days

the relative severity of the postulated events, and permits event dose limits and release limits to be calculated in a consistent manner.

Design basis events use a 100 m elevated release and incorporate conservative meteorology and precipitation. Beyond design basis events incorporate a ground level release that includes building wake effects. Operational events and beyond design basis accidents use average meteorology conditions, but precipitation is not considered.

All dose calculations are based on a release duration of 1 hour and a 1 km distance from the release point to the nearest member of the public. The dispersion factors (χ/Q) used for accident releases with worst-case meteorology and a ground-level release (2-4x10^{-4} s/m^3), accident releases with worst-case meteorology and an elevated release (1.4-2.7x10^{-5} s/m^3), and average annual meteorology (1.0x10^{-6} s/m^3) are given in parenthesis. These dispersion factors do not credit ground deposition and washout effects.

Public dose criteria, summarized in Table 4, are used to evaluate the acceptability of the postulated events and the need for modification of the ITER design. In Table 4, events are categorized as operational events, likely events, unlikely events, extremely unlikely events, and hypothetical events. Operational events are events that occur during routine operations including some faults and conditions that occur because of ITER's experimental nature. Likely events are not considered to be operational events but occur one or more times during the lifetime of the facility. Unlikely events are events that are not likely to occur during ITER's operational lifetime. Extremely unlikely events are events that are not likely to occur by a very wide margin during ITER's operational lifetime. ITER's design basis is derived from the extremely unlikely events. Hypothetical events have an extremely low frequency. These events are postulated with the goal of limiting ITER's risk, and they form the basis for the beyond design basis events.

11.0 Caveats Regarding the ITER Technical Basis

The ITER project [1-13] concept assumes a success-oriented design, construction, and operation that presumes the basic science is essentially resolved. A similar approach is adopted by the US Department of Energy Fusion Program [37,38]. This type of approach must be view with caution in view of recent magnet shutdown issues associated with another large, cutting-edge, international project, namely the Large Hadron Collider (LHC). Magnet issues limited operations at the LHC and delayed the machine from reaching the design beam energy [39].

Issues have been raised regarding possible gaps in the scientific foundation of the ITER [40-42]. These issues arise because large projects often focus upon the political and financial aspects of new technologies. This focus often leads to decision makers assuming that the important scientific questions are resolved and only engineering details remain. It is possible that there are scientific gaps in the conceptual foundations of ITER and in the magnetic confinement devices that merit a more thorough review [40,42]. These gaps are not unexpected because fission reactor development encountered unanticipated events as demonstrated by the Three Mile Island, Chernobyl, and Fukushima Daiichi accidents [29-31,36].

An example of ITER's physics gaps is the lack of a manageable mathematical framework that reproduces the observed experimental results or is sufficiently evolved to predict planned experiments. A situation in which the theory is decoupled from predicting and focusing the experimental program is not necessarily a fatal flaw, since a number of important discoveries have occurred in the absence of a rigorous, predictive theoretical framework [40-42].

The aforementioned issues are important from a health physics perspective. Any design shortcomings have a health physics impact that could lead to potentially larger releases of radioactive material, higher effective doses, and uncertainties in the focus of the radiation protection program. However, innovative projects like ITER have inherent uncertainty and design iterations should utilize sound ALARA principles. Considering these issues, specific aspects of fusion radiation protection are addressed.

12.0 Overview of Fusion Energy Radiation Protection

The D-T reaction of Eq. 3 provides 17.6 MeV for transfer to alpha particles (3.50 MeV) and neutrons (14.1 MeV). The neutrons and alpha particles initiate other nuclear reaction including activation.

The fusion power facility has radiological hazards that are also present in contemporary facilities. For example, the tritium/HTO hazard is similar to that encountered in a Canadian deuterium (CANDU) reactor that uses D_2O as the coolant and moderator [29-31]. The 14.1 MeV neutrons resulting from D-T fusion are similar to the neutron hazard encountered in a low-energy accelerator facility. Therefore, health physics experience with CANDU reactors and accelerators provide insight into a portion of the radiological hazards encountered in a fusion power facility.

A fusion power facility utilizes systems not found in contemporary light water reactors (e.g., the tritium fueling, cleanup, breeding, and recovery; vacuum pumping; plasma heating; water tritium removal; and isotope separation systems). The assessment of the occupational effective dose associated with each of these ITER systems requires detailed design knowledge and related system design details such as the nature and configuration of penetrations in the vacuum vessel, activation of structural materials, water chemistry, and the leak tightness of tritium removal systems. An analysis of the radiation protection consequences of these systems is only possible once specific information regarding the occupancy factors, fusion specific effective dose rates, frequency of operations, and number of workers involved in the operations are known. Given the current stage of ITER design, this information is not yet available.

Although these details should evolve as the design and operational concepts are finalized, considerable health physics information is obtained by considering the individual source terms at a fusion power facility. These source terms directly influence the facility's collective dose.

The collective dose from fusion power plants is one of the criteria for judging their overall success. The current and anticipated fission facility and anticipated fusion facilities' annual collective doses for boiling water, pressurized water, Canadian deuterium, gas cooled, and Generation IV fission reactors, and initial fusion plant are 2.21, 1.20, 0.63, 0.26, 0.70, and 1 – 2 person-Sv, respectively [22]. Therefore, it appears that the collective effective doses at fission and fusion power facilities are comparable. It is likely that fusion facility doses will decrease as operating experience accumulates.

12.1 D-T Systematics

The various low energy rearrangement or break-up channels in the ^5He system govern the systematics of D-T fusion [43]. For example, without added energy, D-T fusion via Eq. 3 occurs with the liberation of 17.6 MeV. No other D-T reactions are likely unless several MeV of excitation energy is provided. For example, $^3H + ^2H \rightarrow ^3H + p + n$ will only occur if at least 2.2 MeV is available.

The various reactions produce nuclides and radiation types that directly influence the radiation characteristics of the facility. These various radiation types (e.g., n, p, and γ), activation processes, and radionuclides define the fusion source term, which is discussed in subsequent sections of this paper.

12.2 Direct Fusion Radiation Sources

Knowledge of the D-T fusion process and ITER plasma characteristics permits an amplification of previous radiation protection overviews [1,2,14-31]. In particular, the sources of occupational radiation exposure arise from the fusion process and activation of associated confinement materials.

The dominant ionizing radiation types include gamma rays from the fusion process and activation sources, beta particles from activation sources and tritium, and neutrons from the fusion process. The external effective dose predominantly receives contributions from beta, gamma, and neutron radiation. Internal intakes of tritium and activation products are also a concern.

In the lower energy fission neutron spectrum, the (n, γ) and (n, p) reactions predominate. The higher energy D-T fusion neutron spectrum opens additional reaction channels. In addition to (n, γ) and (n, p) reactions, more complex reactions (e.g., (n, 4n), (n, 2n α), and (n, ^3He)) occur [43] and contribute to the activation product source term. Additional discussion regarding fusion specific activation products and their production mechanisms are discussed in subsequent sections of this paper.

12.3 Activation Sources

Activation of reactor components will be an important source of ITER radiation exposure. Expected activation products include ^{3}H, ^{16}N, ^{24}Na, and ^{60}Co.

^{3}H is produced in cooling water systems through $^{2}H(n, \gamma)^{3}H$ and spallation reactions in soil surrounding the facility, and in blanket assemblies through the $^{6}Li(n, \alpha)^{3}H$ reaction. Tritium is also a concern in fission reactors [30,31], but the inventory is much less than present at ITER.

^{16}N is produced in cooling water systems from the $^{16}O(n, p)^{16}N$ reaction and has the potential to be a significant source of radiation exposure at ITER. Fission reactor shielding inside containment is dominated by neutron and ^{16}N radiation [30,31].

^{24}Na is a significant part of the concrete source term. Its generation is through a variety of reactions including $^{23}Na(n, \gamma)^{24}Na$. ^{24}Na is not a significant radionuclide at light water fission reactors.

The reactions that significantly contribute to the worker's effective dose depend on the radionuclide produced and its activity. This activity is determined by the contribution from the terms comprising the activation equation. For simplicity, consider the saturation activity (A_{sat}) applicable for sustained, steady state fusion reactor operation:

$$\mathbf{A_{sat} = N \sigma \phi} \qquad (10)$$

where N is the number of target atoms, σ is the energy-dependent microscopic cross section for the reaction of interest, and ϕ is the energy dependent fluence of the particle initiating the reaction of interest.

The number of atoms of a particular target is determined by the mass of the materials of construction for the component being activated; the cross section is determined by the specific reaction and the neutron energy; and the energy-dependent neutron fluence is governed by the fusion process, the fusion reactor configuration, and the materials of construction for the vacuum vessel and its support components. The reactor configuration and the materials of construction govern the neutron interactions, and these interactions degrade the neutron energy. Therefore, the importance of a specific reaction depends on the details of the reactor design and the fusion process utilized to produce power.

12.4 General Ionizing Radiation Hazards

D-T fusion produces a variety of radiation types including alpha particles, beta particles, photons, and neutrons. Heavy ions are also produced, but they deposit the bulk of their energy within the plasma and vacuum vessel. Each of these radiation types and their health physics importance are discussed.

12.4.1 Alpha Particles

In the D-T fusion process of Eq. 3, alpha particles are directly produced, and their energy is deposited within the plasma or in the lining of the ITER vacuum vessel. Alpha particles are also produced by activation of vacuum vessel and plasma support components. The radiation damage induced by alpha particles contributes to increased maintenance requirements and the need for component replacement. ITER operating experience will provide an indication of the required frequency of component replacement, particularly the vacuum vessel and internal tokamak components.

Alpha particles, produced through activation or nuclear reactions with materials of construction, present an internal hazard if they are dispersible. The fusion alpha hazard is not as severe as the alpha hazard associated with transuranic elements (e.g., plutonium and americium) produced in a fission power reactor or recovered in a fuel reprocessing facility [30,31].

Other alpha particle generation results from the unique materials utilized in the facility. Depleted uranium (^{238}U) containers may be used to store the ITER's tritium fuel. Alpha particles arise from the uranium series daughter's that are part of the materials of construction (e.g., concrete) or dissolved in the facility's water supplies.

12.4.2 Beta Particles

Beta radiation primarily results from the decay of activation products, ^{238}U, and tritium, and it presents a skin, eye, and whole body hazard. Given equivalent power ratings, it is expected that the fusion power reactor beta hazard will be similar to that encountered in a fission power reactor.

Beta radiation is also associated with the tritium fuel material and associated depleted uranium storage containers. The tritium beta particles represent an internal hazard while the beta radiation from the ^{238}U series is both an internal and external radiation hazard. Since an equilibrium thickness of ^{238}U metal leads to an absorbed dose rate of 2.33 mSv/h at 7 mg/cm^2, ALARA measures are required near the depleted uranium storage containers to minimize the beta effective dose. The major contributor to the ^{238}U beta absorbed dose is its daughter ^{234m}Pa ($E_{\beta}^{\max} = 2.29$ MeV) [30,31].

Beta radiation is a health physics issue during routine ITER operations and maintenance activities, fueling and defueling activities, and waste processing operations. Appropriate health physics measures are required to minimize the beta radiation hazard [30,31].

12.4.3 Photons

Photons are produced from the decay of activation products and from nuclear reactions that occur within the ITER plasma. The photons emitted from activation products vary considerable in energy and half-life. As noted in subsequent discussions, shielding requirements are influenced by activation gammas including the ^{16}N and ^{24}Na photons.

Photon radiation also occurs from a variety of reactions associated with the D-T fusion process. Sources of photons include bremsstrahlung and nuclear reactions including $^2H(n, \gamma)^3H$, $^3H(p, \gamma)^4He$, $^3He(n, \gamma)^4He$, and $^2H(^2H, \gamma)^4He$ [33,34]. High-voltage equipment associated with plasma heating in MC fusion and laser support equipment in IC fusion is an additional source of x-ray photons. The primary shielding surrounding the vacuum vessel mitigates the photon radiation.

12.4.4 Neutrons

The fusion process produces fast neutrons (e.g., ≥14.1 MeV in D-T plasmas) and lower energy neutrons, including thermal neutrons, as the 14.1 MeV neutrons interact and scatter in the various reactor components. These neutrons activate structural materials, coolant, instrumentation, and devices used to sustain the plasma (e.g., radiofrequency coils and the D-T injection system). One result of activation is the creation of high dose rate components that require remote handling during maintenance operations. The dose rates in these components will be comparable to refueling and maintenance outage dose rates at fission reactors [30,31].

Following D-T fusion, some neutrons escape the ITER vacuum vessel. The expected 14 MeV neutron flux on the reaction chamber wall and total neutron flux on the reaction chamber wall are > 10^{13} n /cm^{-2}-s and > 10^{14} n /cm^{-2}-s respectively [21,44,45]. Although these values are comparable to values at a fission power reactor [29-31], they are about a factor of 10 lower than the projected value at a commercial fusion power reactor [1,2,5].

The expected neutron irradiation of inner fusion reactor components, including the blanket and shield, dictate their required material properties (i.e., capability of withstanding operating temperatures and pressures as well as meeting the radiation damage limits). In addition, reactor components should have low activation properties in order to facilitate operations and maintenance activities in an ALARA manner. A limited set of structural materials has the desired activation properties including those based on ferritic martensitic steel, SiC/SiC ceramic composites, and vanadium alloys. These materials are currently incorporated into the ITER design [1-13].

Neutron radiation damage affects facility equipment lifetimes. Major components require periodic replacement due to the high-energy neutron bombardment. These components incorporate remote handling and processing to minimize worker doses. As an example, consider the blanket assemblies surrounding the vacuum vessel.

Blanket assemblies produce tritium through reactions including $^6Li(n, ^3H)^4He$. The blanket change-out frequency ensures sufficient time to permit breeding of the required quantities of tritium to reach self-sufficiency. Radiation damage is an important consideration in determining this frequency.

Fusion neutrons also present an external radiation hazard. The 14.1 MeV neutrons are considerably more energetic than fission neutrons. Neutrons, escaping the vacuum vessel and not captured by the blanket assembly or other components, lead to occupational doses during surveillance and maintenance activities. These neutrons require shielding, and particular attention must be paid to leakage pathways. At ITER these leakage paths will only be determined following construction and documentation of the as-built configuration.

Neutrons also activate fusion reactor structures and components. Activation products are produced by the neutron fluence impinging on the components of the fusion reactor including the vacuum vessel. At ITER, the candidate component materials include stainless steel, vanadium, and ceramic materials such as Al_2O_3. Activation products include isotopes of Na, Fe, Co, Ni, Mn, and Nb that decay by beta emission, positron emission, and electron capture with associated gamma emission. A key ALARA feature is the optimization of materials that produce minimal activation products or activation products with short half-lives.

Typical neutron activation products of structural materials include ^{55}Fe, ^{58}Co, ^{60}Co, ^{54}Mn, ^{56}Mn, ^{59}Ni, and ^{63}Ni. The variety of materials used in a fusion facility and their associated trace constituents increase the diversity of activation products.

Activation products are primarily solid materials. Excluding the activation products of argon, noble gases are not produced in a fusion machine. Unlike a fission reactor, significant quantities of radioactive krypton and xenon are not expected in a fusion power facility.

Expected fusion activation products also include those resulting from air (e.g., ^{11}C, ^{13}N, ^{15}O, and ^{41}Ar), water (e.g., ^{3}H, ^{7}Be, ^{11}C, ^{13}N, ^{15}O, and ^{16}N), and soil (^{3}H, ^{22}Na, and ^{24}Na) [30,31]. These activation products are common to both the fission and fusion processes. Subsequent discussion explores the additional complexity introduced by the higher energy D-T fusion neutron spectrum.

In addition to the expected activation products, fusion specific activation products are produced. Since all materials used in the ITER reactor are not completely specified, specific examples are only provided for the activation of two components (i.e., the vacuum vessel liner and vacuum vessel structural material) [1,2,5,23,46].

12.4.5 Heavy Ions

Most of the heavy ions remain confined within the vacuum vessel and deposit their energy in the plasma or vessel wall. Therefore, it is not likely that heavy ions present a significant health physics concern in a D-T fusion facility. However, heavy ions contribute to vacuum vessel radiation damage, and increase maintenance requirements and the associated worker doses.

12.5 Specific Radiation Hazards

Likely candidate materials for the vacuum vessel liner include vanadium, a vanadium alloy, and a vanadium composite material. Stainless steel is a likely candidate material for vacuum vessel structural material. The activation of vanadium and stainless steel are addressed in the next two sections of this paper. Tritium is another specific hazard that is addressed following the discussion of vessel activation.

12.5.1 Vanadium Activation – Vacuum Vessel Liner

In view of the previous discussion regarding uncertainty in the selection of materials, it is reasonable to consider natural vanadium as the vacuum vessel liner material. The dominant vanadium activation products resulting from ITER operations are listed in terms of the isotope produced (half-life) and associated production reaction [threshold energy]: ^{47}Ca(4.5 d): ^{50}V(n, p ^{3}He)^{47}Ca[21.5 MeV] and ^{51}V(n, p α)^{47}Ca[11.7 MeV], ^{46}Sc(84 d): ^{50}V(n, n α)^{46}Sc[10.1 MeV] and ^{51}V(n, 2n α)^{46}Sc[21.3 MeV], ^{47}Sc(3.4 d): ^{50}V(n, n ^{3}He)^{47}Sc [20.2 MeV] and ^{51}V(n, n α)^{47}Sc[10.5 MeV], ^{48}Sc(43.7 h): ^{50}V(n, ^{3}He)^{48}Sc[11.8 MeV] and ^{51}V(n, α)^{48}Sc[2.1 MeV], ^{48}V(16 d): ^{50}V(n, 3n)^{48}V [21.3 MeV] and ^{51}V(n, 4n)^{48}V [32.6 MeV], ^{51}Cr(27.2 d): ^{54}Fe(n, α)^{51}Cr[0.0 MeV] and ^{56}Fe(n, 2n α)^{51}Cr[20.0 MeV], and ^{92m}Nb(10.1 d): ^{92}Mo(n, p)^{92m}Nb[0.0 MeV] and ^{93}Nb (n, 2n) ^{92m}Nb[8.9 MeV] [23]. Some of the threshold energies are beyond those encountered in the fission process. For example, the ^{51}V(n, 4n)^{48}V reaction has a threshold energy of 32.6 MeV. In addition, the higher energy D-T fusion neutron spectrum leads to activation reactions that are more complex than the fission activation product production mechanisms that are typically dominated by (n, γ) and (n, α) reactions [30,31]. Multiple nucleon transfer reactions are possible because the D-T fusion neutron spectrum imparts sufficient energy to facilitate these reactions.

12.5.2 Activation of Stainless Steel – Vacuum Vessel Structural Material

In the ITER, the vessel structural material and shielding material candidates are composed of stainless steel (SS-316) [1,45,46]. There are 12 major radionuclides produced from SS-316 activation that dominate the effective dose rate and shielding considerations after 1 day post irradiation and during the subsequent 30 day period [46]. The activation products and their relative contribution to the post-shutdown effective dose rate are provided in Table 5.

Table 5. Fraction of the ITER Effective Dose Rate from an Activated SS-316 Shield[a]

Nuclide	Time post shut-down			
	1 day	*7 day*	*15 day*	*1 month*
^{56}Mn	0.11	<0.0001	<0.0001	<0.0001
^{57}Ni	0.43	0.075	0.0026	<0.0001
^{58}Co	0.22	0.60	0.70	0.70
^{99}Mo	0.085	0.053	0.0089	0.0002
^{64}Cu	0.014	<0.0001	<0.0001	<0.0001
^{99m}Tc	0.024	0.015	0.0025	0.0001
^{54}Mn	0.029	0.080	0.099	0.11
^{51}Cr	0.022	0.055	0.056	0.045
^{60}Co	0.022	0.063	0.079	0.092
^{48}Sc	0.0088	0.0026	0.0002	<0.0001
^{59}Fe	0.013	0.035	0.038	0.035
^{92m}Nb	0.005	0.0095	0.0069	0.0029

[a] Ref. 46.

The results of Table 5 are strikingly similar to fission reactor experience because it indicates that ^{58}Co and ^{60}Co are significant activation sources in stainless steel [30,31]. These isotopes will likely dominate shutdown and outage radiation fields in a manner that is similar to existing fission power facilities. Therefore, maintenance activities involving steel components will be similar to fission reactor activities, and the health physics controls, practices, and lessons learned from fission component maintenance is directly applicable to ITER operations.

12.5.3 Tritium

ITER will maintain kilogram inventories of tritium and deuterium. The tritium inventories present an internal intake concern. Tritium in either molecular or chemical forms diffuses through the vacuum vessel at high operating temperatures. In addition, tritium leakage from the vacuum vessel's coolant, through seals, valves, and piping requires health physics attention. Some tritium also diffuses into the steam system and is released to the environment.

A portion of the tritium resides in routine work areas where it presents a skin absorption, ingestion, and inhalation hazard. Tritium appears as surface contamination which can be resuspended into the air or directly contaminate personnel.

Tritium also resides in a variety of fusion reactor systems. For example, the tritium injection systems require careful operational control and maintenance in order to preclude leakage. In addition, systems transporting tritium or systems involved with tritium recovery merit special health physics attention. A number of health physics challenges are associated with operating and maintaining tritium transport and delivery systems including (1) monitoring sealing systems having very low leakage requirements, (2) performing periodic radiation and contamination surveys of large, complex surfaces, (3) controlling health physics access into facility areas having a variety of radiological conditions, and (4) providing methods for the temporary containment of tritium.

For HTO, the tritium activity absorbed through the skin (s) is proportional to the inhaled (i) tritium activity:

$$\mathbf{I_s = f\ I_i} \qquad (11)$$

where I_s is the HTO activity absorbed through the skin, f is a skin absorption factor, and I_i is the inhaled HTO activity. Values of f range from 0.5 to 1.0 [30,31,47,48].

Facility operations are complicated by the presence of tritium outside the vacuum vessel. The problem is more complex than encountered in CANDU reactors [48], because tritium diffuses through the vacuum vessel. As in a CANDU reactor, minimizing the leakage of systems contaminated with tritium is an essential element of the facility's contamination control program [48].

In CANDU reactors, 30 - 40% of a worker's effective dose is due to tritium intakes. It is expected that a fraction of the effective dose in a fusion facility will also arise from tritium intakes. Therefore, the measurement of the tritium source term, sound contamination control practices, and an active bioassay program are essential elements of the radiological controls program at the ITER [30,31,48].

In CANDU reactors, urinalysis is the preferred method of bioassay for HTO [48]. However, given the various possible forms of tritium (e.g., HTO, HT, HD, DT, T_2, and HDO) that may be encountered in a fusion facility, other bioassay techniques may be required. It is likely these techniques will be developed as warranted by the operational configurations encountered at the ITER.

The activity and diversity of the tritium compounds that may be encountered at the ITER requires a variety of measurement techniques. These techniques are noted in subsequent sections.

13.0 Uncertainties in Health Physics Assessments Associated with External Ionizing Radiation

A number of uncertainties exist at the current stage of ITER fusion power development. These uncertainties are fewer in number than those associated with other fusion power concepts and include the (1) achieving the design density and confinement time, (2) obtaining the design operating lifetimes of the materials of construction when subjected to the D-T fusion neutron spectrum, (3) accurately determining the magnitude and angular dependence of the cross-sections for some of the reactions induced by the D-T fusion neutron spectrum, and (4) determining the final ITER operating staffing levels including their distribution by discipline. Each of these uncertainties has the potential to affect the health physics practices and worker doses at the ITER.

As an illustration of the impact of these uncertainties, consider the activation of water by fast fusion neutrons. In particular, ^{16}O is activated via fast neutron capture to produce ^{16}N via the $^{16}O(n, p)^{16}N$ reaction.

The ITER shielding considerations differ from fission reactor shielding in that fusion shielding must protect equipment as well as personnel. Although a fission reactor is not shielded from the neutron and gamma radiation, the ITER vessel is shielded to maximize its operating lifetime.

ITER shield thickness is determined from radiation constraints for in-vessel components, particularly the superconducting toroidal (TF) and poloidal (PF) field coils. Shielding requirements are governed by blanket performance, critical component replacement, remote maintenance criteria, and operations to expedite replacement of critical components or perform routine maintenance. The limiting shielding constraints at the ITER, involve the shielding in and around the TF and PF vessel penetrations. ITER design requires that this shielding (1) minimize the nuclear heating in the TF and PF coils and cryogenic temperature components near the penetrations, and (2) reduce the shutdown dose rates at specific points in the cryostat to levels that would permit personnel access at two weeks post shutdown.

These requirements place significant constraints on TF and PF penetration shield design, and illustrate the uncertainty in the final design. For example, the total nuclear heating in the PF coils due to neutrons and prompt gamma rays is calculated to be 480 W [1-13]. An additional 100 W is estimated for ^{16}N gamma radiation which adds an additional 20 % to the PF coil heat load, but this value has not been finalized [1-13]. The ITER design philosophy is to first determine the vessel shielding requirements. When the protection and performance of the in-vessel components have been determined, then the ex-vessel shielding will be finalized.

14.0 Measurement of Ionizing Radiation

The measurement of ionizing radiation in a fusion power facility utilizes a variety of techniques. These techniques facilitate the detection of alpha, beta, x-ray, gamma, and neutron radiation, and are similar to the methods encountered at conventional fission power facilities and accelerators [30,31]. Therefore, instrumentation utilized at fission power reactors and accelerators is applicable to ITER. These techniques are summarized in Table 6.

Table 6. General Techniques for Detecting Ionizing Radiation at ITER[a]

Radiation Type	Detector Type	Energy Range	Efficiency
Alpha Particles	Ionization Chamber	All energies for counting and spectroscopy	High
	Proportional Counter	All energies including spec-troscopy applications	High, but dependent on window thickness
	Geiger Mueller Counter	All energies	Moderate
	Inorganic Scintillation Detector (ZnS)	All energies	High
	Organic Scintillation Detector (anthracene)	All energies	Moderate
	Semi-conductor Detector[b] (surface barrier and diffused junc-tion)	All energies	Low
Beta Particles	Ionization Chamber	All energies	Moderate
	Proportional Counter	All energies, and spectros-copy at low energies (< 200 keV)	Moderate
	Geiger Mueller Counter	< 3 MeV	Moderate
	Inorganic Scintillation Detector [CsI(Tl)]	Low energies	Moderate
	Organic Scintillation Detector (anthracene, stilbene, and plastics)	All energies	Moderate
	Semi-conductor Detector[b] (surface barrier, diffused junction, and lithium drifted silicon)	< 2 MeV	Low
X-Rays	Ionization Chamber	All energies encountered in typical applications	Dependent on window thickness particularly at low energies
	Proportional Counter	All energies encountered in typical applications	Moderate
	Geiger Mueller Counter	All energies	Dependent on window thickness
	Inorganic Scintillation Detector [NaI(Tl) with thin window and BGO]	All energies	High
	Semi-conductor Detector[b] (surface barrier, diffused junction, CdTe, HPGe, and lithium drifted germanium)	All energies	High
Gamma Rays	Ionization Chamber	All energies	Low
	Proportional Counter	All energies	Low
	Geiger Mueller Counter	All energies	Low
	Inorganic Scintillation Detector [CsI(Tl), NaI(Tl), and BGO]	All energies	Moderate
	Organic Scintillation Detector (plastics)	All energies	Low
	Semi-conductor Detectorb (surface barrier, diffused junction, CdTe, HPGe, and lithium drifted germanium)	All energies	Moderate

Table 6. (continue)

Neutrons	Ionization Chamber	Thermal neutron detection with BF_3 gas, boron lining, or fissionable material	Moderate
		Fast neutron detection with proton recoil from hydrogenous material	Moderate
	Proportional Counter	Thermal with BF_3 gas or boron lining	Moderate
	Geiger Mueller Counter	All energies via the (n, p) or (n, α) reactions	Moderate
	Inorganic Scintillation Detector [LiI(Eu)]	Thermal	Moderate
	Organic Scintillation Detector (plastics and liquids)	All energies depending on scintillation material	Low

[a] Ref. 29.

[b] Energy resolutions are at least a factor of 10 better than scintillation detectors and this permits spectroscopic applications.

Table 7. Health Physics Concerns Associated with Anticipated Maintenance Activities at a Fusion Power Reactor

Maintenance Activity	Health Physics Hazards	Health Physics Concerns
Vacuum vessel support component maintenance during an outage	Activation products Hot particles Tritium	External radiation Internal deposition
Vacuum vessel support component maintenance during power operations	Activation products Hot particles Tritium Fusion neutrons Fusion gammas	External radiation Internal deposition
Vacuum vessel maintenance during outages	Activation products Hot particles Tritium	External radiation Internal deposition
Routine maintenance and surveillance activities during power operations	Activation products Hot particles Tritium Fusion neutrons Fusion gammas	External radiation Internal deposition
Waste processing [a]	Activation products Hot particles Tritium	External radiation Internal deposition
Defueling and plasma cleanup operations	Activation products Hot particles Tritium Fusion neutrons Fusion gammas	External radiation Internal deposition
Tritium addition to the vacuum vessel [b]	Tritium	Internal deposition

[a] Assumes waste processing is performed at locations well separated from the vacuum vessel.

[b] Assumes tritium addition to the vacuum vessel is performed at locations well separated from the vacuum vessel, and any uranium components storing tritium are shielded

One of the dominant source terms at a fusion reactor is the external radiation derived from fusion neutron and gamma radiation, activation products, and other direct radiation from the fusion process. Tritium is a source of beta radiation and has the potential to significantly impact worker doses. Tritium measurement is well established and includes ion chamber tritium air monitors, tritium bubblers, composition measurements, and thermal methods [37,38,48].

Table 6 summarizes general techniques for detecting alpha particles, beta particles, x-rays, gamma-rays, and neutrons. Relevant detector types, the applicable energy range, and a qualitative efficiency description are provided.

15.0 Maintenance

Maintenance of activated ITER structural components presents both an external as well as an internal radiation hazard. In particular, maintenance activities generate particles of a respirable size during cutting, grinding, welding, and other repair activities. The health physics measures to mitigate these hazards are similar to those utilized at a commercial fission reactor.

Anticipated maintenance activities at a fusion power reactor and associated health physics concerns are summarized in Table 7. Expected maintenance activities include vacuum vessel support component maintenance during outages and power operations, vacuum vessel maintenance during outages, routine maintenance and surveillance activities, waste processing, defueling and plasma cleanup operations, and tritium addition to the vacuum vessel.

Until the design of a fusion power facility is more complete, only a qualitative description of the health physics implications of maintenance operations is possible. However, general health physics considerations for vacuum vessel maintenance, vacuum vessel cooling water system maintenance, and routine maintenance are presented.

15.1 Vacuum Vessel Maintenance

Over time, the inner vacuum vessel wall suffers radiation and physical damage from neutron and heavy ion interactions. It will be necessary to replace the damaged vacuum vessel surfaces every few years. Studies suggest that maintenance involves shutdown radiation fields in the vacuum vessel that are on the order of 3×10^4 Sv/h requiring mechanical or remote handling equipment [1-13]. Vacuum vessel surface repair/replacement operations generate particulates that are respirable and present an internal radiation concern. Hot particle production from activated material is also possible.

The activated vacuum vessel structure and the associated support components produce a radiation hazard that is best addressed with shielding. The majority of the structural activation products are fixed and essentially immobile. However, residual tritium contamination represents an internal concern.

15.2 Vacuum Vessel Cooling Water System Maintenance

The type and design of the vacuum vessel cooling water system impacts maintenance effective dose values associated with these systems. The vacuum vessel coolant and coolant piping will be extensively activated. External radiation levels from these components during maintenance are influenced by internal piping corrosion and subsequent deposition of radioactive material in piping, valves, pumps, and heat transfer systems. Studies suggest that inspection and maintenance activities lead to substantial occupational doses, but with appropriate chemistry control and design, collective effective dose values could be reduced to 2-3 person-Sv with the prospect of further reduction to 0.5 person-Sv [20-24,26,49,50].

15.3 Routine Maintenance

Routine maintenance activities generate particulate material that can become airborne. These airborne particulates enter the body through inhalation and ingestion.

Particles are also generated through the operation of systems that are in proximity to the vacuum vessel including the fuel system, coolant system, and waste extraction system. The nature of these particulate aerosols has not been fully characterized and depends on the specific operating characteristics, maintenance practices, and the actual neutron spectrum of the ITER.

Dust is created in the inner wall of the vacuum

Table 8. Selected Effective Dose Rates - One Day following ITER Shutdown[a]

ITER Location	Effective Dose Rate (mSv/h)	Dominant Nuclides[b]	Source of Nuclides[b]
Between TF Coils and Cryostat	0.0395	^{24}Na (25%)	Biological Shield (100%)
		^{60}Co (21%)	Cryostat (45%) TF[c] Inter-Coil Structures (45%) TF Coils (5%)
		^{58}Co (15%)	TF Inter-Coil Structures (69%) Cryostat (30%)
Between Cryostat and Biological Shield	0.0608	^{24}Na (68%)	Biological Shield (100%)
		^{60}Co (11%)	Cryostat (63%) Biological Shield (29%)

[a] Ref. 20.
[b] The percentage contribution is provided in parenthesis.
[c] Toroidal field

vessel due to surface erosion. This dust is of concern because it can be activated. It is estimated that ITER erosion accumulation is approximately a few hundred grams most of which will be collected by precipitators. Any dust released into accessible work areas presents an internal intake concern.

As noted in Table 7, occupational doses arise from a number of sources during routine maintenance activities. The external radiation hazard from routine maintenance activities is controlled primarily by the facility design.

Routine maintenance activities include the replacement and repair of instrumentation, motors, valves, and packing adjustments. In addition, maintenance support is required for a variety of operational activities including filter replacement, resin sluicing, resin addition, spill cleanup, decontamination activities, fueling operations, sampling activities, and defueling operations. Both internal and external effective dose must be considered in the health physics planning to support routine maintenance.

For example, maintenance support of refueling operations has the potential to encounter a variety of hazards including tritium, activated material, hot particles, and fusion neutron and gamma radiation. Refueling should be engineered such that it can be performed in a low dose rate area with the control of tritium receiving the major focus.

Since the external radiation fields influence maintenance activities, it is important to gain an understanding of the radiation fields that may be encountered during maintenance operations. Table 8 summarizes two-dimensional transport calculations that predict anticipated radiation levels at the ITER [20]. The two-dimensional model consists of a plasma region, toroidal field coils including inter-coil structures, the cryostat, and the biological shield.

At the ITER, a limit of 25 μSv/h is established for hands-on maintenance. This limit assumes that maintenance personnel work for 40 hours a week and 50 weeks a year. As initially evaluated, the effective dose rates, between the TF coils and the cryostat (39.5 μSv/h) and between the cryostat and the biological shield (60.8 μSv/h), exceed the 25 μSv/h ITER limit. An examination of Table 8 suggests that the dominant effective dose rate contribution arises from ^{24}Na [20].

^{24}Na is produced in the concrete biological shield, and its production modes include $^{23}Na(n, \gamma)^{24}Na$, $^{27}Al(n, \alpha)^{24}Na$, $^{24}Mg(n, p)^{24}Na$ [29]. The 25 μSv/h effective dose rate limit can be achieved by adding a 1-cm thick layer of boron to the front of the concrete biological shield that leads to the following reductions in the effective dose rate:

- A 65% reduction (due to thermal neutron capture in the concrete) in the effective dose rate at locations between the TF Coils and the cryostat. This reduction (0.35 x 39.5 μSv/h = 13.8 μSv/h) meets the 25 μSv/h hands-on maintenance limit.
- A reduction of a factor of 3 in the effective dose rate for locations between the cryostat and biological shield. This reduction (⅓ x 60.8 μSv/h = 20.3 μSv/h) also meets the 25 μSv/h hands-on maintenance limit.

These results suggest that the ITER design concept may require modification, but the effective dose rates can be managed through shielding modifications. The reader should note that the two-dimensional calculations are only scoping studies, and do not include the effects of streaming through ducts or penetrations that will exist in the vacuum vessel, toroidal field coils, inter-coil structures, cryostat, and the biological shield of the ITER [20].

16.0 Accident Scenarios

The unique scenarios of postulated fusion power reactor accidents present additional radiation hazards. Some initial fusion plant designs propose to use liquid metal coolant and heat exchange systems. In a severe accident the liquid metal coolant contacting air, water, or steam may lead to an explosive reaction that produces hydrogen gas. Such an event could lead to a loss of structural integrity with the subsequent transport and deposition of activation products, and tritium to offsite locations.

Accident releases differ significantly from those of a fission reactor, which involve primarily noble gases and radioiodine [30,31]. The final safety analysis report for a commercial fusion power reactor will address these and other fusion accident scenarios. A number of advisory groups recommend that fusion facilities be designed and operated such that no public evacuation is required even for a severe accident event [17-19,37,38].

17.0 Regulatory Requirements

In the United States, The Code of Federal Regulations Title 10, Parts 20 and 835, prescribes explicit requirements for worker protection, public protection, and ALARA [51,52]. Part 20 applies to U.S. Nuclear Regulatory Commission Licensees and Part 835 applies to U.S. Department of Energy Licensees. In terms of regulatory requirements, attention is focused on the ALARA aspects of a fusion power facility. In particular, 10CFR20.1101(b) states: "The licensee shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable." Specific ALARA considerations are discussed in the next two sections of this chapter. DOE design requirements are defined in Refs. 37 and 38. The International Commission on Radiological Protection [53] provides the most recent international radiation protection guidance.

17.1 ALARA-Confinement Methods and Fusion Process Types

Prior to reviewing specific design features, ALARA aspects of the fusion confinement method are presented in Table 9. Table 9 summarizes ALARA considerations for the selection of fuel type, reaction geometry, and plasma density for IC and MC fusion devices.

The impact of the inherent physics of the D-D and D-T fusion processes on selected facility design considerations is summarized in Table 10. Specifically, Table 10 considers vacuum vessel maintenance and changeout, production of ^{16}N, ^{3}H intakes, the Lawson criterion, and activation product generation.

Excluding all factors except radiation protection suggests the ideal fusion facility would not be based on D-T magnetic confinement. Tables 9 and 10 suggest that ALARA considerations alone would favor a D-D inertial confinement device.

17.2 ALARA - Design Features

The design and operating characteristics of a fusion power reactor are not yet fully defined. In spite of this uncertainty, the ALARA design features of a fusion power reactor should be developed in a manner that is analogous to existing fission power facilities. Examples of these features include [1-13,30,31]:

- Component and structure activation are minimized through the selection of appropriate low-activation materials
- Components are designed to minimize the accumulation of radioactive material and to facilitate decontamination. This design feature is accomplished by surface preparation (e.g., electropolishing or painting) or ease of disassembly to facilitate decontamination.
- Components are designed to facilitate removal and repair.
- Localized ventilation is provided to minimize airborne contamination. For example, air cleanup system components are located near sources of potential airborne contamination.
- Concrete surfaces are smooth and coated to facilitate decontamination
- Material substitution and purification are incorporated into the design. For example, the use of low-cobalt steel results in lower ^{60}Co activity.
- Shield design considers planned power upgrade and system modifications, and maintenance, surveillance, and operational activities.
- Mockups and full-scale component training aids are used to facilitate task completion.
- Quick disconnects and flanged connections facilitate the removal of components. These components must consider potential tritium leakage during power operations.
- Containment and isolation of liquid spills are facilitated using dikes, curbing, reserve tank capacity, and reserve sump capacity.
- The high-energy neutron spectrum is shielded to minimize the production of activation products and to limit radiation damage.
- Modular, separable confinement structures are used as contamination control barriers.
- Localized liquid transfer systems are used to isolate radioactive material and tritium bearing fluids.
- Fully drainable systems (e.g., piping and tanks) are utilized to facilitate their decontamination and to reduce worker doses. Flush connections are also a key system design feature.

18.0 Other Radiological Considerations

Before concluding the discussion of D-T fusion, a possible process enhancement is outlined. This enhancement is the use of negative muons (μ-) to catalyze the D-T fusion process [54,55].
A negative muon catalyzes the fusion of deuterium and tritium by forming a DTμ molecule. In a DTμ molecule, the muon binds the D-T system so tightly that fusion occurs very rapidly. After fusion of the DTμ system, the muon is released and catalyzes another fusion event. This process is repeated until the muon either decays or interacts with the various species in the fusion plasma.

A muon catalyzed fusion reaction has a profound impact on the health physics considerations at a D-T fusion facility. Muons would not only affect the size of the fusion device, but would also contribute to the facility's radiation signature.

Further discussion is deferred until the direct application of muon-catalyzed fusion in a prototypical device is achieved and sustained.

19.0 Other Hazards

Fusion power facilities have unique hazards as well as hazards common to fission power facilities. The unique hazards for MC fusion include low temperatures and cooling media associated with cryogenic systems, internal intakes related to operation of tritium feed and recovery systems, and strong magnetic fields [56]. Laser radiation and x-rays associated with their high-voltage power supplies are unique to an IC fusion power facility [30,31].

Electromagnetic fields are associated with magnetic confinement systems and plasma heating systems. These electromagnetic fields (EMF) do not have the same frequency and the

Table 9. ALARA Comparison of Fusion Confinement Methods

Consideration	Comment	ALARA Preference
Fuel Type	MC fusion uses T_2 and D_2 gas and HTO production is more likely than in IC fusion. IC fusion uses a solid D-T pellet	IC fusion - The solid fuel pellet minimizes the internal intake of tritium.
Reaction Geometry	MC fusion occurs within a toroidal geometry IC fusion occurs in the small D-T pellet (point source). For equivalent fusion powers and distances from the source, the point source geometry has a higher effective dose rate value. However, the effective dose rates are within about 1 % of each other when the distance from the MC source reaches 3 times the vacuum vessel diameter.	MC fusion – Near the vacuum vessel, higher effective dose rates occur with IC fusion for equivalent fusion powers. The MC fusion advantage disappears as the point of interest moves further from the reaction volume.
Plasma Density	IC fusion operates at a higher density that softens the fusion neutron and fusion gamma spectra. The MC fusion spectrum will be harder than the IC fusion spectrum.	IC fusion - The vacuum vessel receives less damage due to the softer neutron spectrum. Reduced neutron damage minimizes the associated maintenance requirements.

Table 10. ALARA Comparison of D-D and D-T Fusion Processes

Consideration	Comment	ALARA Preference
Vacuum vessel maintenance and change out	The threshold neutron energies from D-D and D-T fusion are 2.45 and 14.1 MeV, respectively.	D-D fusion - The vacuum vessel receives less neutron damage due to the lower energy D-D neutron spectrum. This reduces maintenance requirements and the need for high dose repair activities.
^{16}N activity	The D-D fusion neutron threshold energy lies below the $^{16}O(n, p)^{16}N$ activation reaction threshold. The higher energy D-T fusion neutron threshold lies above the $^{16}O(n, p)^{16}N$ activation reaction threshold.	D-D fusion - Compared to D-T fusion, the D-D fusion neutron spectrum minimizes the ^{16}N source term.
Internal Intake of ^3H	D-T fusion uses tritium and deuterium as the fuel source. Tritium and HTO are more hazardous than deuterium. D-D fusion uses deuterium as the fuel source. Tritium is produced inside the vacuum vessel via Eq. 1 and consumed via Eq. 3.	D-D fusion - Deuterium is less hazardous than tritium.
Lawson Criterion	A smaller value of $\rho\tau$ is required for D-T fusion	Uncertain - An ALARA decision depends on specific facility design requirements and operating parameters.
Activation products	The threshold neutron energies from D-D and D-T fusion are 2.45 and 14.1 MeV, respectively.	D-D fusion - Activation products with higher threshold energies are minimized by the lower energy D-D fusion neutron spectrum.

superposition of radiofrequency radiation with multiple frequencies occurs. The management of EMFs having multiple frequencies is similar to managing airborne contamination comprised of a set of radioactive materials. In both situations, a sum of fractions approach is utilized to establish compliance with regulatory requirements [30,31,57].

A number of hazardous materials will be utilized in a fusion power facility. These materials include metallic components that slowly erode during the fusion process, various gases, inorganic chemicals, and organic chemicals. Components in direct contact with the fusion plasma may contain beryllium, beryllium alloys, vanadium, or vanadium alloys. In the United States, limits for these materials are specified by Occupational Safety and Health Administration regulations and industrial standards [57].

20.0 Conclusions

Health physics considerations at a fusion power reactor have elements in common with existing facilities as well as some unique features. The neutron radiation component at a fusion power reactor has similarities to neutron radiation at an accelerator facility, and the tritium hazard is similar to that encountered at a CANDU reactor. When compared to a fission power reactor, a fusion power facility has unique activation products, unique materials of construction, a higher energy neutron spectrum, a broader spectrum of non-ionizing radiation, and unique components and systems that support the fusion process. The health physics characteristics of ITER depend on the final design, performance characteristics, and component lifetimes and associated maintenance requirements.

References

1. ITER EDA Documentation Series, No. 7, Technical Basis for the ITER Interim Design Report, Cost Review and Safety Analysis. International Atomic Energy Agency, Vienna (1996).

2. Summary of the ITER Final Design Report, G-A0 FDR 4 01-07-21 R.04, International Thermonuclear Fusion Experimental Reactor Project. Cadarache, France (2001).

3. Blake EM. France Named ITER Host Country, Japan to Provide Director-General. Nuclear News 48, No. 8, 130 (2005).

4. McLean A. The ITER Fusion Reactor and Its Role in the Development of a Fusion Power Plant. Radiation Protection Management **22**, No. 5, 27-36 (2005).

5. ITER Report ITER D 2X6K67 v1.0, Plant Description (PD), International Thermonuclear Fusion Experimental Reactor Project. Cadarache, France (2009). 6. Blake M E. Is Inertial Fusion Now in Place with Magnetic. Nuclear News **52** (1), 73-74 (2011).

6. Blake M E. Is Inertial Fusion Now in Place with Magnetic. Nuclear News **52** (1), 73-74 (2011).

7. IAEA STI/PUB/1562, Fusion Physics. International Atomic Energy Agency, Vienna (2012).

8. IAEA Fusion Energy Conference Report FEC-2012, Program Abstracts, Presentations, and Proceedings, 24th IAEA Fusion Energy Conference, October 8-12, 2012, San Diego, CA, International Atomic Energy Agency, Vienna (2012).

9. Motojima, O. Status Report of the ITER Project, Fusion Power Associates 34th Annual Meeting and Symposium, December 10-11, 2013, Washington, D.C. (2013).

10. IAEA TECDOC No. 1724, Applications of Research Reactors towards Research on Materials for Nuclear Fusion Technology, International Atomic Energy Agency, Vienna (2013).

11. US ITER Project Office, ITER Project, Oak Ridge National Laboratory, Oak Ridge, TN (2014), available at www.usiter.org.

12. US Government Accountability Office Report GAO-14-499, Fusion Energy-Actions Needed to Finalize Cost and Schedule Estimates for US Contributions for an International Experimental Reactor, Government Accountability Office, Washington, D.C. (2014).

13. Casper T, Gribov Y, Kavin A, Lukash V, Khayrutdinov R, Fujieda H, Kessel C. Development of the ITER Baseline Inductive Scenario. Nucl. Fusion **54**, 13005-1-13005-9 (2014).

14. Glasstone S. Energy Deskbook. United States Department of Energy, Technical Information Center, Oak Ridge, TN (1982).

15. Thompson HM. Occupational exposure control in the design of a major tritium handling facility, International Conference on Occupational Radiation Protection; Guernsey (United Kingdom); 29 Apr - 3 May 1991, British Nuclear Energy Society; London (United Kingdom) 117-122 (1991).

16. Thompson HM. Occupational problems with tritium and approaches for minimizing them. Proceedings of the IRPA Congress: Portsmouth 94, Nuclear Technology Publishing (1994).

17. Unnumbered UCLA Report, A Synopsis of Major Issues Discussed at the Third Meeting of the Utility Advisory Committee, Fusion Power Plant Studies Program. University of California Los Angeles, February 10, 1994.

18. Unnumbered UCSD Report, Report of the Sixth Joint Meeting of the Fusion Power Plant Studies Utilities Advisory Committee and EPRI Fusion Working Group, University of California San Diego, February 16-17, 1995.

19. Raeder J *et. al.* Safety and Environmental Assessment of Fusion Power (SEAFP). Report of the SEAFP Project, EURFUBRU XII-217/95, Brussels (1995).
20. Khater HY, Santora RT. Radiological Dose Rate Calculations for the International Thermonuclear Experimental Reactor (ITER). UWFDM-1024, Fusion Technology Institute, University of Wisconsin, Madison, WI (1996).
21. Aymar R, Chuyanov V, Huguet M, and Shimomura Y. ITER-FEAT – The Future International Burning Plasma Experimental Overview. Proceedings of the 18th IAEA Fusion Energy Conference, Sorrento, Italy, 4-10 October, 2000, IAEA (2001).
22. Eurajoki T, Frias MP, Orlandi S. Trends in radiation protection: Possible effects on fusion power plant design. Fusion Engineering and Design **69**, 621-624 (2003).
23. Fischer U, Simakov S, v. Möllendorff U, Pereslautsev P, Wilson P. Validation of Activation Calculations using the Intermediate Energy Activation File IEF-2001. Fusion Engineering and Design **69**, 485-489 (2003).
24. Perlado JM et al. Inertial Fusion Reactor Physics: Effect of Activation and Radiation Damage of Materials, and Tritium Emissions. 32nd EPS Conference on Plasma Phys. Tarragona, 27 June - 1 July 2005 (2005).
25. Bevelacqua JJ. An Overview of the Health Physics Considerations at a 21st Century Fusion Power Facility, Radiation Protection Management **22**, No. 2, 10-36 (2005).
26. IAEA-TECDOC-1440. Elements of Power Plant Design for Inertial Fusion Energy: Final Report of a Coordinated Research Project 2000-2004. International Atomic Energy Agency, Vienna, Austria (2005).
27. Pfalzner S. An Introduction to Inertial Confinement Fusion. Boca Raton, FL: Taylor and Francis, (2006).
28. Chen FF. Introduction to Plasma Physics and Controlled Fusion, 2nd ed. New York: Springer (2006).
29. Bevelacqua JJ. Health Physics in the 21st Century. Weinheim: Wiley-VCH (2008).
30. Bevelacqua JJ. Contemporary Health Physics: Problems and Solutions, 2nd ed. Weinheim: Wiley-VCH (2009).
31. Bevelacqua JJ. Basic Health Physics: Problems and Solutions, 2nd ed. Weinheim: Wiley-VCH (2010).
32. Hurricane OA et al. Fuel Gain Exceeding Unity in an Internally Confined Fusion Implosion. Nature **506**, 343-348 (2014).
33. Bevelacqua JJ. The Nuclear Four Body Problem Including Binary Breakup Channels, dissertation submitted in partial fulfillment of the requirements of the degree of Doctor of Philosophy in Physics, The Florida State University, Tallahassee, FL (1976).
34. Bevelacqua JJ, Philpott RJ. Microscopic Calculations for the ^{4}He Continuum, Nuclear Physics **A275**, 301-312 (1977).
35. Jackson JD. Classical Electrodynamics, 3rd ed. New York: John Wiley and Sons, Inc. (1999).
36. Bevelacqua JJ. Health Physics Aspects of Generation IV Reactors. International Nuclear Safety Journal **1** (No. 1), 1-26 (2014). Available from http://nuclearsafety.info/international-nuclear-safety-journal/index.php/INSJ/article/view/20.
37. DOE-STD-6002-96. DOE Standard, Safety of Magnetic Fusion Facilities: Requirements. U.S. Department of Energy, Washington, D.C. (1996).
38. DOE-STD-6003-96. DOE Standard, Safety of Magnetic Fusion Facilities: Guidance. U. S. Department of Energy, Washington, D. C. (1996).
39. Roy A. Science and the Large Hadron Collider: A Probe into Instrumentation, Periodization, and Classification, Dialect. Anthropl. **36**, 291-316 (2012).
40. Kamp LP, Montgomery DC. Toroidal flows in resistive magnetohydrodynamic steady states. Phys. Plasmas **10**, 157 - 167 (2003).
41. Kamp LPJ, Montgomery DC. Toroidal steady states in visco-resistive magneto-hydrodynamics. J. Plasma Phys. **70**, 113 - 142 (2004).
42. Montgomery D., Possible Gaps in ITER's Foundations. Physics Today **59**, No. 2, 10 (2006).
43. Tilley DR, Cheves CM, Godwin JL, Hale GM, Hoffman HM, Kelley JH, Shaw GG, Weller HR. Energy levels of light nuclei: A=5, 6, and 7. Nucl. Phys. **A708**, 1 - 164 (2002).
44. Ehrlich K, Bloom EE, Kondo T. International Strategy for Fusion Materials Development. Journal of Nucl. Mat.**79**, 283-287 (2000).
45. Batistoni P., Research in the Field of Neutronics and Nuclear Data for Fusion. International Conference, Nuclear Energy in Central Europe, Nuclear Society of Slovenia, PORT2001, Jamova 39, SI-1000 Ljubljana, Slovenia, 003.1 – 003.8 (2001).
46. Batistoni P, Rollet S, Chen Y, Fischer U, Petrizzi I, Morimoto Y. Analysis of dose rate experiment: comparison between FENDL, EFF/EAF, and JENDL nuclear data libraries. Fusion Engineering and Design **69**, 649-654 (2003).
47. ICRP Publication 30, Limits for Intakes of Radionuclides by Workers, Pergamon Press, Oxford, England (1979).
48. Bernham JU. Radiation Protection. New Brunswick Power Corporation, Point Lepreau Generating Station, St. John, NB (1992).
49. Information Systems on Occupational Exposure, Seventh Annual Report, Occupational Exposures at Nuclear Power Plants, 1997. Nuclear Energy Agency, Organization for Economic Co-Operation and Development, Paris (1999).
50. NEA-6164-ISOE, Occupational Exposures at Nuclear Power Plants, Fourteenth Annual Report of the ISOE Program, 2004. Nuclear Energy Agency, Organization for Economic Co-Operation and Development, Paris (2006).

51. 10CFR20, Standards for Protection Against Radiation, National Archives and Records Administration, U. S. Government Printing Office, Washington, D.C. (2014).
52. 10CFR835, Occupational Radiation Protection, National Archives and Records Administration, U. S. Government Printing Office, Washington, D.C. (2014).
53. ICRP Publication 103, The 2007 Recommendations of the International Commission on Radiological Protection, Elsevier, Amsterdam (2007).
54. Fujiwara MC *et. al*. Resonant Formation of dμt Molecules in Deuterium: An Atomic Beam Measurement of Muon Catalyzed dt Fusion. Phys. Rev. Lett. 85, 1642 - 1645 (2000).
55. Ishida K, Nagamine K, Matsuzaki T, Kawamura N. Muon Catalyzed Fusion, J. Phys. G: Nucl. Part. Phys. **29**, 2043 - 2045 (2003).
56. Thompson HM, Mustoe J. Field Hazard Management: An Approach to the Control of Magnetic Field Hazards in Fusion Plant Design, in Fusion Technology 1998: Proceedings of SOFT '98, EURATOM-CEA, Cadarache, France (1998).
57. 29CFR, Labor, National Archives and Records Administration, U. S. Government Printing Office, Washington, D.C. (2014).