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Health Physics Aspects Associated with Magnetic Confinement Fusion Power Facilities

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Abstract

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Health physics aspects associated with magnetic confinement fusion are addressed. These aspects are investigated within the context of the operation of the International Thermonuclear Experimental Reactor currently called ITER. ITER has health physics elements in common with existing fission facilities as well as unique fusion related 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. Numerous open issues are required to complete the design of ITER and resolution of these issues will affect the design as well as the health physics implications for occupationally exposed workers.

Key Words: Fusion Power Facility, ITER, Magnetic Confinement, Radiological Characteristics

1.0 Introduction

Commercial fission power facilities have a well defined design basis and thousands of years of operating experience that provides a specific envelope to assess their health physics characteristics and associated hazards. The accidents at Three Mile Island Unit 2, Chernobyl Unit 4, and Fukushima Daiichi Units 1 – 4 provided data to further define the design basis hazards, and advance the ability of the health physics profession to assess the radiological aspects of severe accidents. Proposed fusion power facilities have no operational experience, and health physics assessments are currently based on models with projected operational characteristics.

This paper reviews the current technological basis for fusion power with a focus on ITER's magnetic confinement fusion processes. Although much progress has been achieved in fusion technology, technological gaps remain. These gaps include development of credible plasma operating scenarios, defining operational transients, establishing design characteristics and requirements for plasma exhaust systems, sustaining the burning plasma regime, developing materials to facilitate sustained operations, verifying engineering designs, establishing credible models for plasma stability and sustained operations, establishing accurate construction and operating costs, establishing operating characteristics and radiation profiles, and determining radioactive waste profiles. These and other issues are examined in this paper.

2.0 ITER

ITER previously known as 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]. The ITER is based on magnetic confunement, and 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.

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 likely be the first near full scale facility to demonstrate fusion power operations [2,5,8,9,11-13].

This paper investigates the health physics aspects of the ITER facility. Since 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 [1,2,14-32]. However, a comprehensive health physics treatment of ITER in combination with addressing the emerging technological uncertainties and their impact on the design basis and operational characteristics have yet to be published .

The current design basis also permits a comparison of ITER to commercial fission power facilities. This paper compares the radionuclides of interest, operational characteristics, and design basis accidents 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 and anticipated sources of radiation exposure from this facility 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. More importantly, the uncertainties in resolving the open technical issues have the potential to impact the current ITER design. These uncertainties also impact the health physics characteristics of ITER, and this uncertainty must be recognized in the subsequent discussion.

3.0 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 high-level waste generation, storage of spent reactor fuel, vulnerability to terrorism, and nuclear proliferation [14,25,29-32]. These factors offer considerable motivation for replacing fission reactors with fusion reactors once a viable fusion power design is defined, tested, licensed, and successfully operated. ITER offers the potential for verifying a scalable fusion power facility design.

At ITER, the fusion reaction or process occurs within a plasma composed of deuterium and tritium nuclei. ITER uses magnetic confinement to facilitate the fusion process. Magnetic confinement has been considered to be 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 [6,33-36]. The fusion confinement method influences the fuel materials construction, and the resulting radiation types that must be controlled by the health physicist.

The ITER D-T fusion processes differ substantially from those encountered in fission reactors. Fusion does not produce actinides or fission products (e.g., radioactive isotopes of iodine, cesium, strontium or noble gases) [29-32]. ITER does produce a variety of fusion and activation products. 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.

3.1 Fusion Process Candidates

Fusion involves the interaction of lighter 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 nuclei, 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 [2,8,9,25,26,29-32]:

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

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

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

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 [14,25,28-32,37-39].

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].

An initial assessment of fusion technology was provided in Ref. 32 with an emphasis on ITER. Many of the issues noted in that paper remain unresolved. At the present time, ITER represents the most advanced reactor concept capable of answering many of the open issues associated with the development of fusion power facilities. For this reason, this paper focuses on magnetic confinement fusion and its baseline $^2H + {}^3H$ or DT reaction.

3.2 Overview of the ITER

The ITER 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 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 by fusion neutrons. 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. In addition, material striking the confining walls introduces additional damage that increases maintenance requirements with associated worker doses.

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. Without a successful tritium breeding blanket design, the ITER concept will not be viable. The current world supply of tritium is insufficient to support ITER operations, and this limitation places an emphasis on tritium breeding by the fusion device.

Production scale facilities build upon the ITER experience, are physically larger, and have a higher power output. Their viability depends on the success of ITER. 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] [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 [39], 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 ITER 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.

Section 6 and 7 address design limitations and open issues associated with these basic systems, structures, and components. Successfully resolving these issues is crucial to the establishment of a viable magnetic confinement fusion power facility.

3.3 General Radiological Characteristics

The ITER radiological hazards are representative of those occurring in a production scale fusion facility. These hazards include tritium; alpha, beta, gamma, and neutron radiation; activation products; and particulates generated by plasma collisions with containment structures [25,29-32]. These hazards and their relative consequences may change as the ITER design evolves.

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 including the emission of alpha, beta, and gamma radiation types [1 – 13, 29-32]. At the current design stage of 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 isotopes of 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]. These remote systems have yet to be demonstrated for their feasibility, operability, and long-term performance.

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 an accident or operational transient.

Tritium, activation products, and toxic materials could be released during an accident or off-normal event. A number of ITER energy sources, including the inherent energy residing in the fusion plasma and magnet systems, facilitate the dispersal of radioactive and toxic material.

3.4 Accident Scenarios/Design Basis Events

The safety envelope, 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 including quench events, loss of cryogen, tritium plant events, and auxiliary system faults. Given the current ITER design, these scenarios form the foundation for the 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.

3.4.1 Loss of Coolant Accidents

LOCAs are serious events in a fission reactor because the coolant removes heat from the fuel core [29-32,40]. In a fission reactor (e.g., pressurized water reactors (PWRs) and boiling water reactors (BWRs)), a LOCA has the potential to damage the fuel/cladding fission product barrier. Any loss of coolant, increases the temperature of the fuel, increases the likelihood for fuel damage, and reduces the margin for protection of the fuel from melting. From a health physics perspective, a LOCA results in the release of fission and activation 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 have the potential to disperse radioactive material including tritium and activation products. The extent of the dispersal area and quantity of radioactive material released 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 include 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 projected to be 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 assumed 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.

3.4.2 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 [29-32]. 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.

3.4.3 Loss of Vacuum Accidents

In a fission reactor, a loss of vacuum event typically involves the turbine's condenser. Fission reactor 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 [30,32,40].

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].

3.4.4 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 [29-32,40].

In a fusion power reactor, plasma transients include overpower events and plasma disruptions [29]. 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 results [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.

3.4.5 Magnet Fault Transients

Electrical faults occur in power supplies, switchgear, turbine-generators, and other components. These faults include arcing and other electrical discharge mechanisms that 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 generate arcs that produce 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.

3.4.6 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 of the superconductor without an electrical discharge can cause helium leakage. Cryogenic 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 [29-32].

3.4.7 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 [29-32]. Tritium is monitored and controlled to minimize the potential for internal dose impacts [29-32].

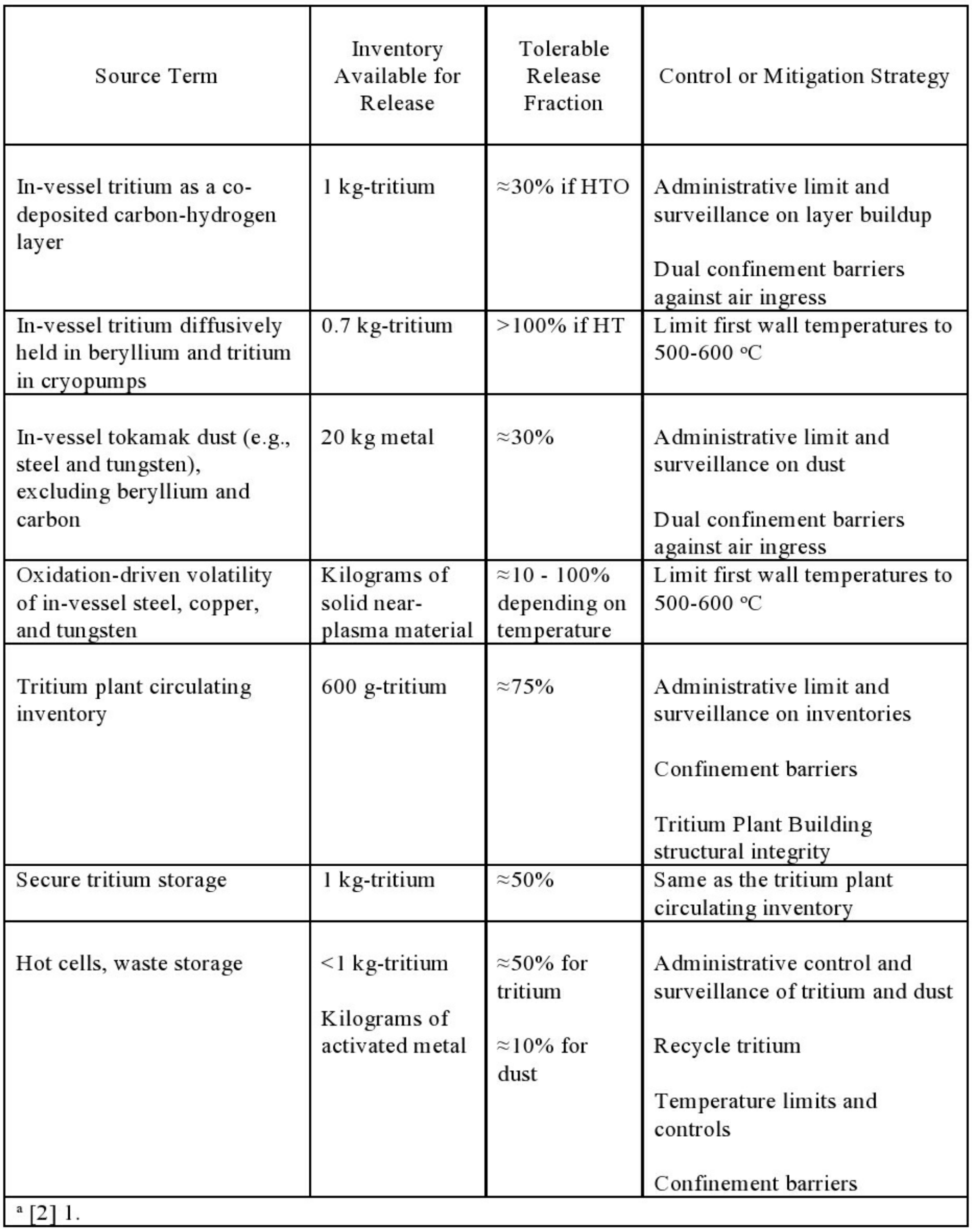
Hydrogen is used in fission reactors for primary system oxygen control and for its thermal properties in turbine generator systems. Any leakage of hydrogen gas creates the potential for an explosive mixture, and hydrogen monitors are included in the design to minimize these events. As demonstrated by the Fukushima Daiichi accident, hydrogen explosions have the potential to disperse fission products [40].

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 [1-13,41,42].

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 [30,31].

3.5 Radioactive Source Term

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



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%	Same as the tritium plant circulating inventory
Hot cells, waste storage	<1 kg-tritium Kilograms of activated metal	≈50% for tritium ≈10% for dust	Administrative control and surveillance of tritium and dust Recycle tritium Temperature limits and controls Confinement barriers

[a] [2] 1.

Table 2 Radioactive Material Inventories in Postulated ITER Accident Events [a]

The tolerable release fraction listed in Table 2 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 2 and its associated data are unique to the ITER design and could change as operational experience is incorporated into a production scale facility.

3.6 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].

The BDBE events require periodic review as the ITER design evolves. As noted in subsequent discussion, regulatory authorities have raised concerns regarding the capability of the structural capability of ITER to withstand the additional weight of required shielding to achieve the design dose rates.

4.0 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 [29-32]. 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 PWR and BWR light water reactors (e.g., 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 and their locations 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 overall worker safety profile and operational 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.

4.1 D-T Systematics

The various low energy rearrangement or break-up channels in the ^{5}He 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, $^{3}H + ^{2}H \rightarrow ^{3}H + p + n$ 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, α, β, and γ), activation processes, and radionuclides define the fusion power facility source term, which is discussed in subsequent sections of this paper.

4.2 Direct Fusion Radiation Sources

Knowledge of the D-T fusion process and ITER plasma characteristics permits an amplification of previous radiation protection overviews [1,2,14-32]. 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, ^{3}He)) 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.

4.3 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, α)^{3}H reaction. Tritium is also a concern in fission reactors [29-32], 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 [29-32].

^{24}Na is a significant part of the concrete source term. Its generation is through a variety of reactions including ^{23}Na(n, γ)^{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:

$$A_{sat} = N \sigma \varphi \quad (4)$$

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 rate or flux 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.

4.4 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 addressed in subsequent discussion.

4.4.1 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 the 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 [30-32].

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 daughters that are part of the materials of construction (e.g., concrete) or dissolved in the facility's water supplies.

4.4.2 Beta Particles

Beta radiation primarily results from the decay of activation products, ^{238}U and its daughter products, 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 (2.29 MeV) [30-32]. The higher energy beta particles also produce bremsstrahlung that is an external dose concern.

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 [30-32].

4.4.3 Photons

Photons are produced from the decay of activation products, bremsstrahlung, and from nuclear reactions that occur within the ITER plasma. The photons emitted from activation products vary considerable in energy. 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 ^{2}H(n, γ)^{3}H, ^{3}H(p, γ)^{4}He, ^{3}He(n, γ)^{4}He, and ^{2}H(^{2}H, γ)^{4}He [41,42]. High-voltage equipment supporting plasma heating in magnetic confinement fusion and laser support equipment in IC fusion are an additional source of x-ray photons. The primary shielding surrounding the vacuum vessel mitigates the photon radiation.

4.4.4 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 values at fission reactors [30-32].

Following D-T fusion, some neutrons escape the ITER vacuum vessel. The expected 14 MeV neutron flux impinging upon the reaction chamber wall and total neutron flux striking the reaction chamber wall are $> 10^{13}$ n $/\mathrm{cm}^{-2}$-s and $> 10^{14}$ n $/\mathrm{cm}^{-2}$-s respectively [21,44,45]. Although these values are comparable to values at a fission power reactor [29-32], 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 design 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 and associated damage. 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 ^{6}Li(n, ^{3}H)^{4}He. The design blanket change-out frequency ensures sufficient time to permit breeding the required quantities of tritium to reach self-sufficiency. Radiation damage is an important consideration in determining this frequency. A viable blanket design is a significant open issue and is required for a successful fusion power facility.

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. The resulting radionuclides 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) [30-32]. These activation products are common to both the fission and fusion processes.

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 available for the activation of two components (i.e., the vacuum vessel liner and vacuum vessel structural material) [1,2,5,23,46].

4.4.5 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.

4.5 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.

4.5.1 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 [30-32]. Multiple nucleon transfer reactions are possible because the D-T fusion neutron spectrum imparts sufficient energy to facilitate these reactions.

4.5.2 Activation of Stainless Steel – Vacuum Vessel Structural Material

At 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 Refs. 45 and 46.

4.5.3 Comparison of Fission and Fusion Radionuclides

The previous sections addressed various accidents, radiation types of interest, and radionuclides of interest. Maintenance activities are addressed in the next section, but these activities are affected by the radionuclides of interest. Table 3 compares the general classes of radionuclides that could be encountered at fission and fusion power facilities. These radionuclides impact maintenance and other activities at these power facilities.

Table 3 Radionuclides, Waste, and Effluents Affecting Maintenance Activities at Fission and Fusion Power Facilities

Radionuclide, Waste, and Effluent Types	Fission Reactor	Fusion Reactor
Activation Products	The lower neutron energy spectrum produces a variety of systems dominated by (n, γ) and (n,p) reactions.	The 14.1 MeV neutrons allow a greater variety of reactions, but the actual radionuclides depend on the specific facility design.
Fission products	A wide variety of radionuclides are produced depending on the fuel (e.g., Th, U, and Pu) characteristics.	None produced.
Radioiodine	Major component of the release source term.	None produced.
Noble Fission Gasses	Major component of the release source term.	None produced.
Tritium	Produced primarily in pressurized water reactors from $^{10}B(n, 2\alpha)$, $^{6}Li(n, \alpha)$, $^{2}H(n, \gamma)$, and tertiary fission.	Significantly greater activities of tritium are available because the facility is based on the DT reaction.
Actinides	Produced by neutron absorption by the Th, U, and Pu fuel.	None produced.
High Level Waste	Actinides and long lived fission products.	None expected depending on the final selection of component materials.
Accident Releases	Noble gasses, iodine, and particulates.	Tritium and particulates.
Effluents	Primarily activation products, tritium, and selected fission products (e.g., radioiodine and noble gasses).	Tritium and activation products.

5.0 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 4. 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.

Table 4 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

Table 4 Health Physics Concerns Associated with Anticipated Maintenance Activities at a Fusion Power Reactor (continued)

Maintenance Activity	Health Physics Hazards	Health Physics Concerns
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 the source for tritium addition to the vacuum vessel is performed at locations well separated from the vacuum vessel, and any uranium components storing tritium are shielded.

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. In particular, the facility design will be impacted by the resolution of numerous engineering and licensing issues that are addressed in subsequent discussion.

6.0 Engineering Obstacles to a Magnetic Fusion Power Facility

Fusion is well understood from a fundamental physics viewpoint. The $^{2}H + {}^{3}H$ baseline fusion reaction has known cross sections as well as reaction products. The challenge is translating this well known reaction into a viable fusion facility that is capable of providing a sustained output of electrical energy. Accomplishing this task presents a significant engineering challenge. The most significant engineering challenges are noted in subsequent discussion [47,48]. These include: (1) plasma stability, (2) plasma exhaust control, (3) plasma transients, (4) burning plasma conditions, and (5) technology challenges.

6.1 Plasma Stability

Although Maxwell's equations [39] provide a basic physics envelope for a fusion plasma, the engineering aspects are significantly more complex. A magnetized plasma is an extremely complicated system. There are limited external control options available at ITER, and the associated defining parameters and spatial profiles are difficult to determine without extensive experimental data. Numerical simulations provide important guidelines, but experiments are required to benchmark these calculations. However, it is currently only possible to create experimental conditions that facilitate a particular configuration or plasma scenario. A general simulation to model the complete plasma process is beyond current capabilities. Large scale devices such as ITER will provide essential data for defining a viable commercial fusion facility. However, the ITER facility is not yet operational and numerous open issues exist.

In a fusion power facility, the plasma must reach sufficiently large pressures and temperatures, and these conditions must be stable with respect to localized and macroscopic perturbations. The plasma should also be characterized by low turbulence levels. Low turbulence enhances both energy and particle confinement with a limited accumulation of impurities produced by interactions of the plasma with the various confining structures. The peripheral plasma regions must also be stable to avoid interactions with the surrounding wall material.

Plasma stability affects the generation of particulate material and the lifetime of system components. Maintenance requirements increase as component lifetimes decrease which increases worker radiation doses. The generation of particulate material also affects the offsite release source term. Stable plasma operation is essential to a viable fusion facility as well as limiting worker radiation exposures.

6.2 Plasma Exhaust Control

Following the DT reaction, hundreds of megawatts of power must be exhausted from the peripheral plasma region before the next reaction can occur. During this process, plasma-facing materials should be subjected to minimal erosion to limit the production of impurities that could contaminate the plasma and degrade subsequent fusion reactions. Moreover, materials of construction should minimize retention of tritium, reaction products, and erosion materials as well as facilitate the removal of the fusion-reaction helium.

The exhaust system must also facilitate stable boundary conditions for fusion performance without producing extreme transient events. Fusion exhaust incorporates a divertor that is a magnetic field component that directs the plasma to flow long distances along magnetic field lines before striking a surface. These trajectories cause the plasma to lose energy through collisions and radiation emission. Optimizing the plasma flow limits erosion and impurity production that promotes a sustainable plasma, and limits the release source term. Unfortunately, mechanisms to achieve the desired performance are not fully resolved.

The mechanisms to achieve a stable plasma and control its dynamics are currently open issues. Resolution requires determination of the desired geometry of the associated magnetic field. Predicting and optimizing the width of the unconfined plasma layer through which power is channeled is also unresolved. The resolution of these issues requires experiments in subscale devices that test various configurations, validation of the subscale results in larger systems that create reactor conditions, and integrated peripheral plasma models to utilize the experiments to optimize reactor designs. None of these requirements have been achieved.

6.3 Plasma Transients

Transients occur due to instabilities at the plasma edge, in the core or internal volume, or across the whole plasma volume. At the edge, a tokamak plasma tends to exhibit a steep pressure gradient that naturally creates instabilities. These instabilities are nonlinear and have the potential to produce periodic, short-lived transients. The transients generate outgoing pulses of energy and associated particles. These pulses create a localized thermal load that can impact the integrity of the plasma-facing components. Research efforts are ongoing to mitigate these edge effects, but a final resolution of this issue has yet to be resolved.

An associated challenge is plasma disruptions including plasma volume changes that lead to an energy deposition in the confining wall structure. At ITER, plasma disruptions are projected to produce peak divertor depositions on the order of 10 MJ over a few milliseconds. A tungsten divertor is estimated to only survive only a few hundred of these events. Divertor material lifetimes need to be better defined, and methodologies developed to mitigate the consequences of plasma disruptions. These transients impact the sustainability of the plasma, and must be resolved before ITER type fusion devices can lead to a commercial fusion power facility.

6.4 Burning Plasma Conditions

A commercial fusion power facility must have the capability for sustained operations in the burning plasma regime. In this operational configuration, plasma heating by the fusion produced alpha particles and reaction byproducts of the DT fusion reactions dominate over the external heating sources.

The burning plasma regime is entered when fusion power gains exceed 5, but these conditions have yet to be achieved in a magnetic fusion device. As a consequence of alpha particle heating, couplings of plasma parameters characterize this state. However, these parameter couplings are not fully understood. Burning plasmas conditions are partly inaccessible in existing experimental devices, and alpha heating is an open issue. An experimental evaluation of this regime is only possible in an actual burning plasma. Resolving the alpha heating question is one of the prime motivations for constructing ITER. Unfortunately, much of ITER will be constructed before this issue is resolved, and there is no guarantee that a successful design will emerge. This uncertainty impacts the health physics aspects of the facility. The operational characteristics of a fusion power device affect its radiation output, activation products produced, effluent release pathways, and design basis accidents. These factors will have a strong influence on worker radiation exposures and measures to minimize these doses.

6.5 Technology Challenges

There are significant technology challenges associated with establishing a commercial fusion power facility incorporating magnetic confinement. These include, but are not limited to (1) developing a viable and sustainable tritium breeding blanket, (2) obtaining materials consistent with the fusion plasma, (3) producing magnets capable of generating the requisite confining fields, and (4) instituting an integrated system to confine and control the plasma.

6.5.1 Tritium Breeding Blanket

Electricity production in any power facility requires the generation of heat to boil a working fluid to spin a turbine-generation. In a fusion power facility, heat is generated by the fusion reaction and its reaction products. Conversion of fusion power into heat occurs in the blanket. The blanket is the structure, surrounding the plasma, capturing the fusion neutrons, after multiplication and moderation, where the $^{6}Li(n, \alpha)^{3}H$ reaction occurs to produce tritium. The blanket provides a self-sufficient tritium production mechanism, and shielding to surrounding reactor components.

Currently, there is no viable design for this system. A viable tritium breeding blanket is required for a feasible fusion power facility, because the available world tritium inventory is insufficient to support commercial fusion power facilities.

Significant research and development are required to develop a viable blanket design. Since the blanket is required to provide a sustainable supply of tritium fuel, associated support systems are needed to extract tritium from the blanket. These support systems include remote handling systems to process the blanket and extract the produced tritium, and tritium processing systems to collect and handle the tritium for injection to sustain the fusion plasma.

Additional open issues associated with the blanket include the selection of an efficient transfer medium to extract heat. The choice of the transfer coolant (currently water or helium) is an unresolved design issue. Selection of a cooling medium is needed to finalize the design of the heat extraction system.

Tritium breeding is dependent on the blanket design that has yet to be defined. Specific breeding ratios and tritium processing times are significant design parameters that affect the available fuel supply. There are currently no dedicated facilities to test blanket designs under realistic plasma conditions. These facilities or a viable test program are required to remove this major obstacle to the development of commercial scale magnetic confinement fusion power facilities. A finalized tritium breeding design will impact worker doses as well as the release source term.

6.5.2 Materials

Radiation damage to structural components is a concern in fission power facilities, and an even greater concern in fusion facilities. The DT fusion reaction produces 14.1 MeV neutrons that will impinge upon structural materials and induce radiation damage. These structural materials must withstand the neutron fluxes that create atomic cascades and damage the material’s lattice structure. The net effect of the damage is the displacement of atoms and weakening of the structural materials. Materials are also affected by nuclear reactions that generate helium and hydrogen gas in the blanket and the vacuum vessel materials.

There is an incomplete experimental data set to support the assessment of the proposed structural materials and their behavior under fusion reactor conditions. Accurate numerical simulations are not yet able to model the complex spatial and temporal ranges encountered in a fusion power facility. Extrapolations from current data sets and test conditions are a significant challenge, because there is currently no method of testing and qualifying materials to meet the temperature and pressure conditions encountered in a fusion power facility.

6.5.3 Magnets

Fusion power facilities utilizing magnetic confinement require large magnetic fields over extended volumes to confine the plasma. Generating these fields entails large steady state current densities, and these currents should flow through a minimal resistance to reduce the power costs. The desired low power design goals demand utilization of superconducting materials. Challenges of utilizing superconducting magnets were illustrate at CERN when a massive magnetic quench event severely damaged the superconducting magnet systems [49].

Quench events and the subsequent dissipation of heat are exacerbated at a magnetic fusion power facility due to the large heat load generated by the fusion process. Avoiding quench events or other magnet failures demands stringent control of the superconducting manufacturing process and fabrication of magnets that produce the required confining field characteristics. These requirements demand that testing facilities qualify components of the magnet system at high operating field strengths and currents under operating system pressure and temperature conditions. Similar requirements are placed on the cryogenic systems supporting superconducting magnet operations.

6.5.4 Integration toward the Prototype Power Plant

These aforementioned technology issues must be addressed for a fusion power facility to be viable. However, these items represent significant open issues and cost drivers. In addition, the various systems of the facility must be integrated in a manner to support sustained operations. An integrated control system (ICS) must be capable of monitoring and stabilizing the plasma. This system maintains the heat balance between the high power microwave power sources and cryogenic system that supports the superconducting magnet systems. The ICS must also maintain plasma stability during heat extraction and tritium and deuterium injection. These are significant challenges that have yet to be tested. ITER provides an opportunity to ensure the development of an ICS required for a viable fusion power facility.

7.0 Current Status of ITER

ITER construction continues with the goal of achieving burning plasma via fusion in a large tokamak. This facility has encountered years of schedule delays and associated cost increases [50]. These issues have been compounded by the discovery of defective components, misalignment in welding surfaces in four vacuum chamber segments, and French regulatory agency concerns regarding the adequacy of radiological shielding. The aforementioned design deficiencies have the potential to impact the health physics characteristics of fusion power facilities and their viability.

These issues are complex and preclude ITER management from estimating the length of delays or the associated cost impacts. Ref. 50 estimates that ITER's completion could be delayed by about three years beyond the facility's current projected 2025 start date.

In April 2022, cracks were detected in a portion of the thermal shield cooling water piping system. These pipes reside between the vacuum vessel and the surrounding superconducting magnetic field coils. Removal and rework are required to resolve this issue. These defects are not a positive indication of the construction quality. Additional defects will have negative cost and schedule implications, and jeopardize the completion and operation of ITER.

An additional significant defect involves misalignments in the welding surfaces of four vacuum chamber wall segments. The surfaces must be reworked to fill void areas and remove high points. The remaining five vacuum chamber segments are in the manufacturing process.

Repair work involving these two defects is expected to be completed in two years. ITER management suggests that the assembly process can proceed in parallel with the repairs.

ITER's cost and schedule is also affected by recent concerns of the French Nuclear Safety Authority (ISN). In February 2023, ISN issued a halt assembly order. ISN questioned the adequacy of ITER's radiological shielding, and expressed concerns that any shielding addition to the existing 3 m thick concrete structure surrounding the reactor assembly would increase the mass beyond the capacity of the reactor support system. These are nontrivial concerns and have the potential to alter the ITER design as well as its cost and schedule. The shielding issue and its resolution also have a direct impact on facility worker exposure during operations and outage activities.

Basic materials issues involving contact of surfaces with the fusion plasma are also unresolved. These issues affect the radiological characteristics of the device that impact worker exposures and accident release scenarios. The selected materials will determine the activation products produced as well as the lifetime of the affected components. These materials issues are not yet resolved. For example, the project originally planned to coat the vacuum chamber walls with beryllium. Worker exposure to the toxic characteristics of beryllium emerged as a concern as well as considerations for future commercial facilities. Tungsten is expected to enhance worker safety as well as ITER's capability to withstand radiation damage from high energy fusion neutrons.

In addition to these issues, there are significant unresolved items related to the adequacy of numerical simulations of ITER plasma behavior under the projected operating conditions. These issues are addressed in Ref. 51 and the interested reader should consult that reference for additional details.

8.0 DEMO

It is the author's view that each of the formidable engineering and technology issues will be individually solved. The challenge to achieving a viable magnetic confinement fusion power facility is not these individual issues, but successfully integrating the requisite systems into a reliable, sustainable operating, and cost effective plant. This integration may not be completely achieved at ITER. It may be delayed until the successor to ITER that is the demonstration fusion reactor (DEMO). DEMO is intended to prove that the desired integration is feasible.

It has a design goal of generating hundreds of MW of net electrical power. DEMO is intended to operate in a closed, self sufficient manner with an availability that is consistent with commercial applications. DEMO must also demonstrate that its waste stream that does not involve radionuclides that require a geologic repository.

The ITER burning plasma regime must be successful in order to ensure that this operating experience is incorporated into DEMO's operating envelope. DEMO faces an inherent challenge because there is an urgency to develop the demonstration reactor in a timely manner. This requires that DEMO proceed in parallel with ITER, rather than utilizing a sequential approach that depends on ITER successfully meeting its various milestones. This approach has inherent risk because ITER success is not guaranteed,

In spite of these considerable uncertainties, DEMO or a similar facility is required for fusion power systems to become a viable technology. In the interim, massive numerical simulations are required to support this venture. These calculations must be guided by evolving experimental results. Given the complexity of the numerical and experimental results, success is not guaranteed, and will require a sustained funding effort especially when progress slows due to the need to overcome evolving technical and regulatory issues.

Advanced computing systems, quantum computing, and artificial intelligence offer possible avenues for success. These approaches will guide future experiments and interpretation of results to improve numerical simulations. Fusion power technology advancements require a sustained multidisciplinary effort that requires integration of expertise in a number of disciplines including artificial intelligence; plasma, nuclear, solid state, and atomic physics; construction; regulatory affairs; public communications; robotics; material science; instrumentation and control; and mechanical, electrical, and structural engineering.

9.0 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 radiological characteristics of ITER depend on the final design, performance characteristics, and component lifetimes and associated maintenance requirements. Open issues in technology development and integration of the system components while sustaining the plasma impact the final design. Resolution of these issues as well as worker radiation estimates will depend on a final design configuration for a magnetic fusion power facility.

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