# Chapter XVI

# SAFETY AND ENVIRONMENTAL IMPACT

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## 1. INTRODUCTION

As part of the INTOR Phase-One design, each of the participating countries has prepared a preliminary assessment of the principal safety and environmental concerns of the proposed reactor. The purpose of this assessment is to identify these concerns so that they can be dealt with and solved, where possible, early in the design of the reactor. This interaction between safety analysis and design is a crucial process towards the objective of ensuring that INTOR can be constructed, operated, and ultimately decommissioned without undue risk to operating personnel and the general public, and with a minimum environmental impact.

These assessments are not intended to be complete safety analyses of the reactor. In fact, there are a number of areas where additional research, development and analysis remain to be done to ensure a sound design as well as an accurate safety analysis. Nevertheless, the assessments have been done in sufficient depth to provide confidence that the key safety concerns have been identified and that there are no overriding safety problems that cannot be solved with prudent siting, design, construction and operating practices. Future analyses should be done in sufficient detail so that specific requirements for engineered safety features can be specified.

The primary safety concerns for INTOR involve the presence of radioactive inventories and potential radioactivity release mechanisms. The radioactive inventories include both tritium and activation products that are produced by the interaction of various materials with the high-energy neutrons produced by the D-T fusion reaction. Release of a portion of these radioactive inventories could occur by leakage during normal operation or maintenance activities, or as a result of an accident or failure of various components, systems or structures.

This chapter provides a summary of the INTOR safety and environmental analysis that has been done to date. Included in the chapter are discussions of: (1) the various radioactive and toxic materials, and energy sources of concern,

(2) INTOR containment philosophy and requirements, (3) radioactive release under normal and accident conditions, and (4) consequences of radioactive release and waste management considerations.


## 2. SOURCE TERMS

### 2.1. Radioactive inventories

The radioactive inventories for INTOR include tritium, which is used as fuel, and activation products that are produced by the interaction of materials used in INTOR with high-energy neutrons from the fusion process. The purpose of this section is to discuss those radioactive inventories in order to evaluate their effects on the safety assessment in later sections.

### 2.1.1. Tritium inventory

The tritium inventory has been estimated to be in the range of from approximately 3.4 to 3.9 kg. Of this inventory, about 2.3 kg is in the storage facilities in a non-vulnerable form. The major uncertainty regarding tritium inventory is the quantity contained in the solid tritium breeder material, ranging from 0.5 to 1 kg. Research data are needed to make a better estimation of this inventory, especially for irradiated breeder material. The most vulnerable tritium inventories for release during an accident are those in the plasma chamber, the vacuum system and the fuel processing system. The estimated value of this total inventory is approximately 0.6 kg. The inventory contained in storage is in a separate vault and is therefore considered relatively invulnerable regarding accidental release.

Based on these data, the tritium inventories for the INTOR conceptual design have been estimated as follows:

| | |
|---|---|
| Tritium handling systems | 0.6 kg |
| Breeding blanket systems | 0.5 to 1.0 kg |
| Storage blanket systems | 2.3 kg |
| Total | 3.4 to 3.9 kg |

Estimates were also made of the concentration of tritium in the coolant water. The estimated leakage rates of tritium into the primary coolant range from 3 Ci/d to 10 Ci/d. When lithium hydroxide is used as corrosion inhibitor, there is a production of less than 1 Ci/d of tritium in the primary water by neutron activation. The designers have estimated that the primary-coolant tritium activity could be kept below 30 $Ci/m^3$, a value lower than that for the primary coolant

of a current heavy water reactor, by operating a tritiated water recovery unit installed in a bypass flow-stream from the primary coolant.

It has been pointed out by the participating countries that there can be a significant difference in the estimates of tritium permeation rates into the primary coolant, depending on whether one uses a conventional thermal diffusion model of tritium through a metal wall or different models which assume implantation of tritium particles from the plasma into the metal wall first before the diffusion process takes place. Further investigation on the applicability of such models is needed for a better understanding of the tritium permeation mechanism.

### 2.1.2. Activation product inventory

#### 2.1.2.1. Inventory in structures

Activation products will be present in the solid structures and equipment surrounding the plasma. The most significant activation product inventory is in the first wall, blanket and shield structures. Estimates of these inventories range from approximately $3 \times 10^8$ Ci to $1.3 \times 10^9$ Ci at shut-down. More than half of the inventory is in the first wall, which contains 40 to 100 MCi/m$^3$. The most significant radioactive isotopes at shut-down are isotopes of $^{55}$Fe (40 to 50% in radioactivity), $^{56}$Mn (15 to 20%), $^{51}$Cr ($\sim$ 10%), $^{58}$Co ($\sim$ 7%), and $^{54}$Mn (4%), which are activation products of constituents of stainless steel. Of these isotopes, $^{56}$Mn decays quickly after shut-down because of its short half-life.

#### 2.1.2.2. Inventory in coolant streams

A small portion of the activation products produced in the structures is circulating in the primary coolant water through corrosion or sputtering. Estimates of this inventory range from 24 to 70 Ci. The radioactive materials contained in the water in significant amounts are again iron, manganese and cobalt. The water coolant will also contain a large quantity of $^{16}$N, approximately $10^6$ Ci/m$^3$. However, because of the short half-life of this isotope, it is not considered a significant radioactive material from the point of view of safety.

#### 2.1.2.3. Inventory in the reactor building atmosphere

Estimates were also made of the concentration of $^{41}$Ar that would be present in the reactor room during operation. These estimates were of the order of $10^{-7}$ Ci/m$^3$, if one assumes a reasonable ventilation rate of the reactor building atmosphere. $^{41}$Ar decays quickly after reactor shut-down because of its short half-life. The production rate of $^{14}$C has been estimated to be 0.07 Ci/day. However, because of the large volume of the reactor building, and with proper operation

TABLE XVI-1. ENERGY SOURCES IN INTOR

|                                                              | Total energy (GJ) |
| ------------------------------------------------------------ | ----------------- |
| TF coils, magnetic field energy                              | 40                |
| Energy due to tritium and deuterium combustion in air        | 0.30              |
| Plasma thermal energy                                        | 0.23              |
| Plasma disruption energy                                     | 0.22              |
| Plasma magnetic energy                                       | 0.06              |
| First wall and blanket coolants, energy above boiling point  | 0                 |

of the ventilation system, the concentration of $^{14}C$ in the reactor building would be kept to an insignificant level.

## 2.2. Toxic materials

At present, no large inventories of toxic materials are foreseen in INTOR. Quantities of lead and boron may be contained in the blanket and shield regions; however, the use of these materials in industry is common and no special problems are envisioned. Small amounts of metals from the first wall could conceivably be released as aerosols after an accident. If this were to occur, concern about the toxicity of the aerosols would be far outweighed by concern about their radioactivity. Other toxic compounds may be present in instrumentation and in experimental equipment, but these have not been identified as yet.

## 2.3. Energy sources

INTOR will have energy sources which conceivably could cause releases of radionuclides to the containment. The magnitudes of these sources are summarized in Table XVI-1. The purpose of this section is to discuss the various energy sources in order to estimate their possible effects on INTOR in later sections. The principal energy sources are described in the following paragraphs. Other energy sources such as lithium metal, to be used in a test module, may exist in INTOR, but they have not been identified yet and should be considered as they become incorporated into the design.

### 2.3.1. Energy stored in the large superconducting magnets

The superconducting magnets of INTOR represent a possible safety hazard inasmuch as the release of their stored energy (about 40 GJ) could conceivably

serve as the initiating energy for an accident sequence that would release toxic or radioactive materials. The energy of the magnets could be coupled to other systems of the reactor through: (1) a major structural failure and the breaking of adjacent electrical cables and piping, (2) a quench of the magnets in the common cryostat and possible overpressurization of the containment, and (3) the interaction of fringe fields and circuitry in the control room during an accident sequence.

### 2.3.2. Energy released if the deuterium and/or tritium in the fuelling systems were to be released into the containment and ignited

Chapter XI indicates that INTOR will have 3.4 to 3.9 kg of tritium present in the various tritium systems of the reactor. An estimated amount of up to 1 kg will be present, mostly as $T_2O$, in the blanket. If the containment had an air atmosphere, the tritium could burn if released in gaseous form into the containment. However, the total "vulnerable" inventory of tritium is estimated to be less than one kilogram. This amount, if mixed with an equal amount of deuterium, would occupy about 7 m³ at standard temperature and pressure.

### 2.3.3. Energy contained in the plasma and energy to be deposited during plasma disruption

The INTOR plasma, with a total beta value of 5.6%, has a thermal energy of 230 MJ and an energy of 60 MJ contained in its poloidal magnetic field. Of those energies, 220 MJ are estimated to be deposited on the various parts of the first wall during a plasma disruption of 20 ms duration. The energy, if deposited on a limited area of the first wall, could cause melting of a portion of the first wall.

### 2.3.4. Energy due to nuclear heating of the first-wall divertor and blanket

Nuclear heating could cause thermal transients in reactor components if cooling is compromised while the reactor remains on. Total losses of cooling are potentially a problem, even in the case where the plasma is immediately extinguished, because of afterheat in the first wall and blanket.

### 2.3.5. Energy potentially released by expansion of the liquid cryogens, if these are released into the containment

INTOR may contain about seventy cubic metres of liquid helium in the superconducting coil cases and their storage reservoirs. Approximately thirty cubic metres of liquid nitrogen will also be present. Smaller amounts of other cryogens may be present in instrumentation systems and in experimental facilities.

These cryogens, if released, could cause pressurization of the containment building as well as structural problems in those systems with which they come in contact.

### 2.3.6. *Energy stored in the coolant for the first wall, blanket and blanket test modules*

The design rule being employed for INTOR is that (for water coolant systems) the system pressures must not exceed 1 MPa and the outlet temperatures must not exceed 100°C. Under these conditions, a loss-of-coolant accident (LOCA) type of event would not cause significant pressurization of the containment as long as the volume of water added to the containment is not significant compared with the containment volume.

## 3. CONTAINMENT PHILOSOPHY AND REQUIREMENTS

### 3.1. General discussion of layout

INTOR is an experimental installation containing 3.4−3.9 kg of tritium and approximately $10^9$ Ci of activation products; therefore, it is reasonable to surround it with a containment. Such a containment must serve as a last protective barrier, preventing radioactive and toxic chemical releases into the atmosphere in both normal and accident conditions.

Since tritium presents the major radiation risk potential in INTOR, the requirements for the containment are mainly determined by the task of preventing tritium releases in accidents. Containments are usually costly; therefore, an adequate choice of their concept and structure is important.

The containment is an anti-accident system; therefore, its design should be based on requirements obtained from accident analysis. Estimates of tritium release to the building during low-probability accidents range from 10 to 100 g; such releases must be localized and collected with the help of the containment and active tritium clean-up systems.

Placing all the potentially radiation-hazardous equipment, e.g. the reactor, tritium handling and fuelling systems, cooling circuits of the first wall, blanket and blanket modules, and coolant cleaning system, in a single containment would require building a containment with an area of approximately $100 \times 100$ m². Such a containment would substantially increase the INTOR capital cost; therefore, it is preferable to construct a containment system consisting of a central containment of smaller size and separate vacuum-tight boxes as auxiliary containments.

The central containment, having the shape of a cylinder, about 60 m in diameter and about 50 m high, with a lid, will contain the reactor and five neutral

beam injectors (NBI), while the vacuum-tight boxes will contain other potentially hazardous equipment. Both containments may have passive protective systems, e.g. underground water tanks for dilution of radioactive effluents or holding tanks for radioactive gases.

Such an arrangement is preferable from the viewpoint of maintenance because failure of any single system does not result in radioactive contamination of others.

## 3.2. Number of layers

A reasonable design approach is to provide a two-layer containment, for example consisting of

— an outer shell of prestressed concrete, lined with
    3—5 mm of stainless steel
— an inner stainless steel shell, 5 mm thick.

Between these shells a negative pressure with respect to the containment atmosphere of about 10 torr difference should be maintained. This will serve as an additional barrier for uncontrolled releases of tritium and other radioactive gases generated inside the containment.

## 3.3. Vacuum boundaries

A series of separate vacuum volumes, divided by appropriate barriers inside the main reactor building, will help to decrease hazardous releases into the atmosphere and will separate adjacent systems, thereby avoiding mutual interactions in case of an accident.

One such vacuum barrier configuration is as follows (Fig.XVI-1):

(a) The primary or inner vacuum toroidal boundary comprises the first wall and blanket/shield components requiring replacement or maintenance. The vacuum level inside it is determined by the plasma operating conditions and can attain $10^{-7}$ torr.

(b) As this primary vacuum barrier includes bellows and thin welded joints that cannot be radiographed, secondary or outer vacuum-tight boundaries are placed around the primary vacuum to give the possibility of detecting leakages.

(c) The separate cryostat vacuum boundary is placed around the TF and PF coils. The vacuum level inside it is about $10^{-5}$ torr.

Inside the vacuum-tight boxes for the tritium systems, such as the fuel gas circulation system, breeding-blanket tritium recovery system and tritium waste treatment
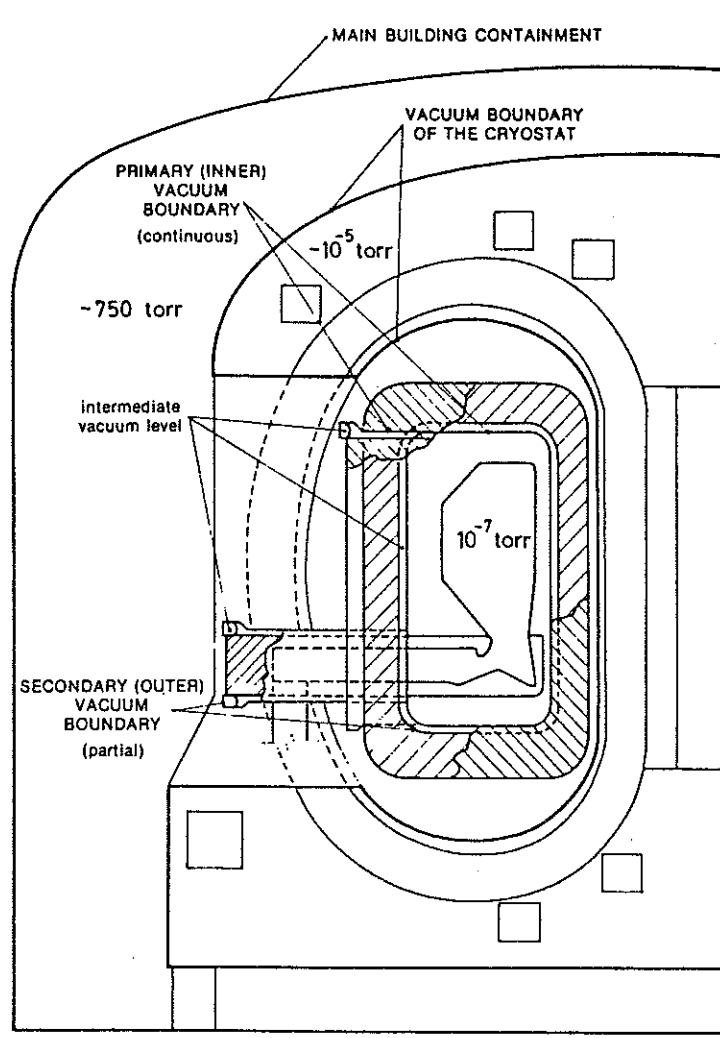
*FIG.XVI-1. Vacuum boundary topology of INTOR.*

system, and for other equipment, e.g. the primary cooling system, it is desirable to have primary and secondary tritium barriers (Figs XVI-2, XVI-3).

## 3.4. Internal missile protection

In case of an accident inside the containment it is possible that some equipment fragments will have a high speed, e.g. when a high-pressure helium circuit pipe ruptures. Therefore, it is necessary to provide the following measures to protect the internal surface of the containment:

—  local protective screens and covers
—  force supports and bindings.

When placing equipment and constructing internal partitions, it is advisable to take into consideration the possibility of local high-pressure transient zones occurring in "semi-confined" volumes which may result in damage to concrete walls as well as falling and accelerated fragments. Therefore, in some cases, one should avoid using pre-cast concrete structures because of their lower strength compared with monolithic structures.

### 3.5. External missile protection

The containment should be designed with proper consideration of the probability of an aircraft crash. The design practices applied for nuclear power plants may be used.

### 3.6. Natural disasters

When determining the requirements of the containment, the problem of natural disasters should be considered for the actual conditions of the INTOR site. For example, the containment must be designed to withstand seismic events, according to the relevant regulations and standards.

### 3.7. Containment atmosphere

At present, three possible options are suggested for the containment atmosphere: inert atmosphere, vacuum, and air at subatmospheric pressure. The first two options avoid potential hazards connected with the appearance of explosive mixtures of hydrogen and oxygen as well as lithium fires. Also the use of a vacuum atmosphere eliminates the problem of air activation within the containment. However, since these options substantially complicate maintenance, the use of air under subatmospheric pressure seems to be the best option.

The ventilation systems should follow the usual rules: providing for air flow from lower risk areas to higher risk areas, with proper monitoring and provision of lower pressures as the contamination risk increases. For example, the ventilation system of the reactor hall may be built so that it provides air from the outside to the workshop, then to the airlock (of the reactor hall), to the hot cells, to the waste store and finally to the reactor hall, with pressure within each unit along the ventilation "line" being maintained at a slightly lower level.

Estimates have shown that the tritium concentration level inside the containment boxes under normal operating conditions may be about $5 \times 10^{-6}$ Ci/m$^3$. The ventilated volume is about $3 \times 10^5$ m$^3$, and the ventilation rate provides 5 to 10 volume changes per day. Neutron activation of the containment air atmosphere would add $1.5 \times 10^{-7}$ Ci/m$^3$ of $^{41}$Ar and $3 \times 10^{-8}$ Ci/m$^3$ of $^{14}$C. Proper operation of the ventilation system will help to maintain the airborne activity within acceptably low levels.
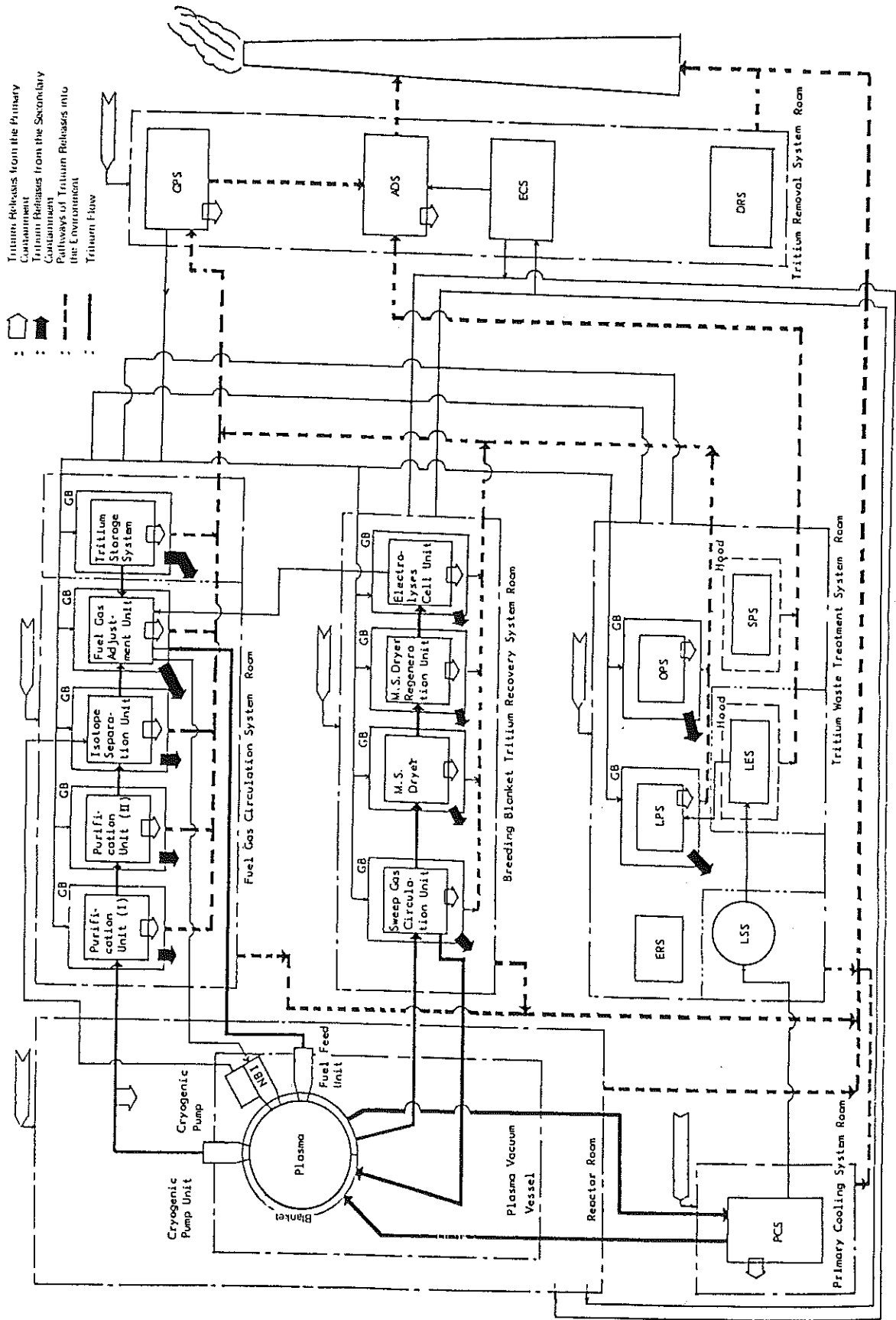
## 3.8. Containment atmosphere clean-up systems

No unique treatment is required for the air released from the containment through the stack during normal operation. However, this air should be monitored for contamination and, if the radionuclide concentration exceeds the permissible level, the emergency air-cleaning system (ECS) must be used. Within the tritium barriers around radiation-hazardous equipment, tritium may be accumulated as a result of mechanical equipment failures or operational errors; therefore, the air within the tritium barriers must be monitored for radionuclide concentration and cleaned by the tritium removal system if required.

The schematic pathway of tritium releases during normal operation and one of the suggested schemes of the clean-up system are illustrated in Fig.XVI-2.

The most important component of the clean-up system is the tritium removal system, which consists of the following systems:

(a)  *Inert gas purification system (GPS)*, which includes a combination of catalytic oxidizers, molecular sieve (MS) dryers and cold traps to remove impurities and tritiated species from the glove-box (GB) atmospheric inert gas in order to minimize tritium leakage to the environment.

(b)  *Effluent air detritiation system (ADS)*, which should remove tritium in a once-through mode from air effluents of tritium systems.

(c)  *Emergency air-cleaning system (ECS)*, which should provide detritiation of the reactor containment atmosphere following an accidental tritium release in order to reduce the activity level in the contaminated room. This system includes catalytic oxidizers and molecular sieve dryers. It is actuated and operated automatically.

(d)  *Dryer regeneration system (DRS)*, which should regenerate the dryers of the tritium removal system and effluent tritium removal system (ERS) belonging to the tritium waste treatment system. The molecular sieve dryers should be regenerated by means of recirculated heated gas, and outgassed water should be collected by condensation.

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*FIG.XVI-2.  Schematic pathway of tritium releases during normal operation.*
*GPS = gas purification system, MS = molecular sieve, GB = glove-box,*
*ADS = air detritiation system, ECS = emergency air-cleaning system,*
*DRS = dryer regeneration system, ERS = effluent tritium removal system,*
*LPS = liquid waste packaging system, OPS = oil packaging system,*
*LSS = liquid waste storage system, LES = liquid waste enrichment system,*
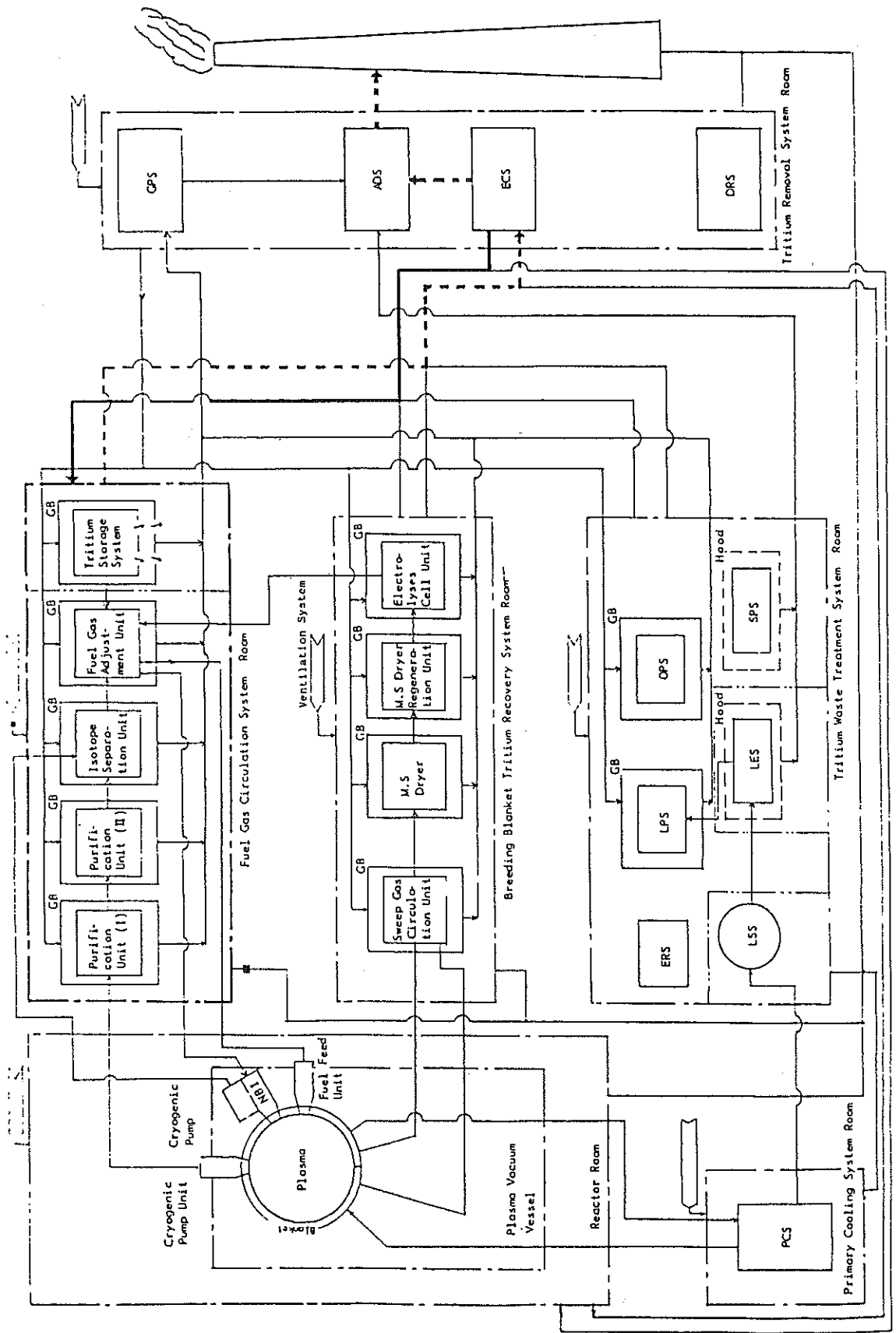*SPS = solid waste packaging system.*

FIG.XVI-3. Schematic pathway of accidental tritium releases.

In case of a failure of the vacuum or tritium barriers, tritium may be released into the main or secondary containment. Then the following steps should be taken automatically:

(a) The corresponding containment volume should be connected to the ECS
(b) The air supply and exhaust duct damper of the normal ventilation system should be closed
(c) The ECS should be actuated and operated until the activity level in the contaminated premises is reduced to an acceptable value
(d) The contaminated room should always be maintained at slightly sub-atmospheric pressure to prevent tritium leakage
(e) During ECS operation, a part of the air flow should be exhausted to the ADS to balance infiltration air flow; the remainder of the air flow should be recirculated to the contaminated (cleaned) premises.

Figure XVI-3 illustrates the schematic pathways of accidental tritium releases.

During maintenance the tritium accumulated in structural materials may be released into the containment through access ports. In this case, ventilated air may be released to the environment directly through the stack or via the ADS, depending on the tritium concentration level.

## 4. SAFETY ANALYSIS

The purpose of this section is to assess the safety acceptability of INTOR. The emphasis is on assessment rather than analysis, because the design is not yet completed in sufficient detail to allow a complete safety analysis. The regulatory basis has not been established, and there is no other large D-T burning device that can be used as a precedent for INTOR. Thus, the objective of the work done to date is to assist in guiding the Phase-One design and to point out problems that must be solved as the design progresses to greater detail. This interaction between safety assessment and design provides for improvement in the inherent safety of the reactor, which has the effect of (1) decreasing the consequences in terms of reactor damage or radioactive release of various accident-initiating events, and/or (2) decreasing the probability of an accident. Thus, the overall safety risk of operation of the facility is reduced.

Included in this section are discussions of representative accidents that could potentially lead to radioactive release. Discussion of the consequences of radioactive release under normal operation and as a result of an accident is presented in Section 5.

### 4.1. Accident analysis

Section 2 of this chapter discusses the radioactive and energy sources that will be present in the INTOR reactor. The focus of this section is to provide a

preliminary assessment of the potential for the various energy sources to cause accidental release of a portion of the radioactive inventory.

### 4.1.1. Superconducting magnet accidents

The energy stored in the toroidal magnetic field of the INTOR conceptual design, approximately 40 GJ, is the largest potentially damaging energy source in the reactor. Even a part of this energy released during an accident could damage the magnets and other reactor systems, in turn releasing some of the radioactive materials in the plant.

A number of potential accidents have been postulated in which the stored energy can lead to damage to the magnets or to related equipment. A quench, i.e. the complete loss of the magnet's superconductivity, is the most common abnormal operating condition that can lead to an accident. During a quench, at least part of the magnet's stored energy is dissipated in the copper stabilizer of the winding. The magnet design includes a reliable quench detector and provides for an energy dump outside the cryogenic system. Calculations have shown that even if the entire 40 GJ were uniformly distributed in the magnets by resistive heating during a quench, the maximum temperature would be approximately 200°C. No gross damage would be expected to result from this event, although some local yielding of the copper could occur.

The formation of an arc, either within a superconducting magnet or across two of the leads, could be the most damaging of the postulated magnet accidents. The Euratom group [2] has developed a model for an arc in the magnet circuit and a corresponding computer code to analyse the consequences of an arcing accident. The model uses the Ayrton formula for the voltage-current characteristic of the arc. The constants used in the Ayrton formula and reported in the literature generally apply to currents below 1000 A. More experimental information on the voltage-current characteristic of the arc in realistic fusion environments will be necessary to obtain suitable accuracy for final INTOR magnet design.

A first estimation of arcing consequences on the toroidal field coils of INTOR has been performed, assuming that the initiating event is fracturing of the conductor, that the conductor current is 20 kA to 40 kA, and that the arc voltage is 150 V. With these assumptions, the consequent evaporation rate for copper was calculated to be 480 g/s to 960 g/s and the helium evaporation rate 14 kg/s to 28 kg/s. These values refer to adiabatic conditions, without any decoupling of energy by the intervention of appropriate safety systems. The time scales involved are such that the formation of missiles may be possible. Additional analysis and/or data on the potential formation of missiles will have to be taken into consideration when the final design is developed.

Routine operation of the INTOR toroidal field magnets imposes a set of strict requirements on their structural elements: the strain and displacement of

the superconductors within the coil jacket must be strictly limited; the insulation must maintain high dielectric strength under irradiation; and very little hydrogen can be released owing to insulation decomposition. Although most super-conducting magnets now in use are small, laboratory-scale coils, experience shows that the time between failures is only about 1.5 to 2 years. To achieve acceptable failure frequencies for the superconducting magnet and to contain the severity of those accidents that do occur, the INTOR design should develop a special independent emergency protection system which includes at least the following three subsystems:

(1) A control and failure detection system arranged for the necessary redundancy
(2) A temperature stabilization and equalization system
(3) An energy dump system.

The presence of large quantities of cryogens poses a potential hazard due to their thermal expansion when heated. The calculation of the possible over-pressurization within the containment is given elsewhere in this report. Perhaps the worst effect of a cryogen release would be to lower the temperature of the magnet support structures and piping to below the nil ductility transition (NDT) temperature while the magnets are still operating. Brittle failure of the support structures under full load could ensue. To avoid a total release of the cryogens, the dewars should be divided into compartments. In addition, vents can be provided to relieve dewar pressure to the outside atmosphere under quench conditions.

### 4.1.2. Primary coolant system accidents

The total thermal power to be generated by INTOR will be 620 MW. The heat load resulting from this power will be removed from the first-wall and blanket regions by a low-temperature and low-pressure water coolant system. A failure of this coolant system could cause overheating of the first wall and blanket, and potentially lead to release of activation products. Therefore, analysis of coolant system failures is an important part of INTOR safety analysis.

The design rule being employed for INTOR is that the system pressures should not exceed 1 MPa and the outlet temperatures should not exceed 100°C. Under these conditions, the coolant system would not have a large amount of stored energy for initiating a loss of coolant, and a loss-of-coolant accident (LOCA) type of event would not cause significant pressurization of the containment since the volume of water added to the containment would be small compared with the containment volume.

Nuclear heating, however, could cause thermal transients in reactor structures and components if the cooling is compromised while the plasma remains burning.

Total losses of cooling are potentially a problem, even in the case where the plasma is immediately extinguished, because of the decay power of the activation products in the first wall and blanket. Many permutations and combinations of coolant system accident conditions are possible. For this study, three cases were selected for the graphite-moderated blanket option as representing bounding cases for this option and to provide input for future INTOR design considerations:

(1)  Total loss of coolant from the first wall and blanket -- plasma burn continuing
(2)  Same as (1), with immediate shut-down of the plasma
(3)  Coolant leakage from first wall/blanket into the vacuum vessel.

The first two cases were calculated using an analytical model which included the first wall, the multiplier, and the blanket out to the blanket jacket. Afterheat values were calculated for 6 $MW \cdot a/m^2$ integrated loading, using a one-dimensional (1-D) cylindrical approximation. The blanket jacket was assumed to be an adiabatic boundary. Coolant loss was assumed to take place in 10 seconds.

For the loss-of-coolant accident with the plasma burn continuing (case 1), the analyses performed showed that 200–300 s are required for the first wall to reach the melting temperature. The response of the system is rather slow, even for this extreme condition. Thus, significant time is available for safety system response or emergency operator action in case of any loss-of-fluid accident (LOFA) or of a LOCA. Of course, the plasma burn would be terminated automatically, or it would be extinguished because of the evolution of impurities from the first wall at elevated temperatures, before the first wall would melt. Melting of the lead multiplier, however, occurs in approximately 30–40 s because of its relatively low melting temperature (600 K).

For the loss of coolant with the plasma burn terminated (case 2), the analysis showed that neither the first wall nor the blanket would melt owing to decay heat if a heat sink were available. The lead moderator would start to melt at about 3000 s. It was assumed that the lead would remain in place even though molten. In case 2, a heat sink was provided by the tritium-removal helium purge flow which was assumed to continue. The calculation indicated that heat conduction from the first wall to the blanket and thereby to the purge flow would be sufficient to prevent the system from overheating. This indicates that it should be possible to design INTOR such that melting of the first wall from afterheat is prevented, even for very severe events such as total loss of off-site power or severe LOCAs in the plant. It is important to note that if the conductance path from the first wall to the blanket is impaired, for example by moderator fracturing, active cooling of the first or second wall may be required to prevent melting. For example, at 24 hours after plasma shut-down, 60% of the total decay heat is generated in the first wall and 30% in the multiplier region. Analysis of cases 1 and 2 showed a high probability for lead melting; thus, provision should also be made for the expansion of the multiplier in order to prevent structural damage.

For case 3, the pressure response in the vacuum vessel was calculated, to examine its response following a breach of the first-wall/blanket system with coolant entering the plasma chamber. The primary cooling system is pressurized by surge tanks. The pressure and maximum temperature are 1 MPa and 100°C. If the surge tanks are not working during the accident, water flowing into the vacuum vessel will evaporate by residual heat and the pressure in the vacuum vessel will rise on account of the generated steam. The pressure in the vacuum vessel would rise to about 0.51 MPa in the case of rupture of a blanket pipe, and to about 0.61 MPa in the case of rupture of a first-wall cooling tube. If the pressure containment capability under internal pressure is considered to be approximately 0.2 to 0.3 MPa, the vacuum vessel may be damaged. Therefore, it may be necessary to install a device such as a rupture disk to relieve the pressure and to mitigate the potential consequences of such an accident.

### 4.1.3. Plasma disruptions

Another type of nuclear heating that is of safety interest is plasma disruption. The analysis given in Chapter VII shows that a "hard" plasma disruption will create a significant amount of first-wall melting and ablation. The analysis shows that a melt-layer of approximately 0.14 mm is formed at the hottest point during a disruption with an energy deposition of 289 $J/cm^2$ and a disruption time of 20 ms. The extent of the ablation is dependent on both the area and duration of plasma energy deposition. Analyses have been performed to determine the area and disruption time that could cause a hole in the first wall. It was calculated that an energy of $2.8 \times 10^4$ $J/cm^2$ would be required to ablate 4.0 mm of stainless steel (the end-of-life thickness) if it were deposited in 18.5 ms. This energy exceeds the design reference plasma disruption energy (289 $J/cm^2$) by approximately two orders of magnitude and would require a spatial peaking factor of about 210. This indicates that it is not likely that a single plasma disruption could penetrate the first wall. The ablation model used for this study was that described in Chapter VII.

### 4.1.4. Cryogenic system failure

INTOR will contain approximately 70 $m^3$ of liquid helium in the super-conducting coil cases and their storage reservoirs. About 30 $m^3$ of liquid nitrogen will also be present. Smaller amounts of other cryogens may be present in instrumentation systems and in experimental facilities. These cryogens, if released, could cause pressurization of the containment building as well as structural problems in systems the cryogens come in contact with.

Release of the 30 $m^3$ of liquid helium could cause an increase in the reactor building internal pressure. Estimates of the reactor building volume have ranged

from (1.0 to 3.5) $\times 10^5$ m$^3$. Thus, a total release of the liquid helium inventory could result in reactor building overpressures ranging from approximately 0.015 to 0.05 MPa. Large containment buildings can be designed and built to withstand such overpressures.

This large quantity of liquid helium could also have an effect on load-bearing structures. For example, the 70 m$^3$ of liquid helium could cool approximately 5.5 $\times 10^6$ kg of steel — the approximate weight of the tokamak structure — by 5 K. Thus, an additional concern about a cryogenic system failure is the effect on load-bearing structures resulting from nil ductility transition effects.

Design solutions that could be effective in reducing the consequences of a cryogenic failure would be:

(1)  compartmentalization of the cryogen inventory to reduce the quantity of cryogen spilled, and/or

(2)  provision for venting to reduce reactor building pressure build-up.

Liquid-liquid explosive interactions, e.g. between liquid helium and water, presumably will not be a problem because their occurrence could only take place in a large volume, i.e. the containment.

### 4.1.5. Hydrogen fires

Tritium and deuterium are both isotopes of hydrogen. Since there are numerous ignition sources in a fusion plant, the potential for a hydrogen fire or explosion exists if significant concentrations are released to the air atmosphere of the reactor building.

It is estimated that INTOR will have an inventory of 3.4 — 3.9 kg of tritium. However, the total "vulnerable" inventory of tritium is expected to be less than 0.6 kg. This amount, if mixed with an equal amount of deuterium, would occupy only about 4 m$^3$ at standard temperature and pressure. If the containment volume is $10^5$ m$^3$, then the concentration of hydrogen isotopes is well below the amount (4%) required to sustain ignition in air. A concentration of 9% is necessary for downward flame propagation, and approximately 18% concentration is required for detonation. The entire inventory of 3.4 — 3.9 kg, combined with an equal amount of deuterium, would provide a concentration of only about 0.03%. Thus, it is not expected that hydrogen fires or explosions have the potential to violate the INTOR containment integrity; however, caution in design must be exercised to prevent the potential for accumulation of explosive mixtures of deuterium and tritium in local areas. As an example, the fuel storage vault will have an inert atmosphere to prevent an explosion hazard there.

No mechanism is currently identified whereby significant quantities of hydrogen could be evolved in INTOR by reduction of metals or dissociation of water; therefore, hydrogen explosions do not appear to be a probable energy source for radioactive releases from INTOR.

## 6. Events external to the facility

There are a number of external events with the potential for causing radio-
ive release that INTOR designers will need to consider in future design efforts.
:se are: severe natural phenomena, airplane crashes and sabotage.

No analysis of the effects of severe natural phenomena has been performed
date for INTOR, primarily because no site has been selected. Such analyses are
cessarily site-specific; however, some issues are generic to devices such as INTOR.
ır example, certain natural phenomena such as floods, landslides, lightning
ɔrms, and winds and tornadoes will affect the outer envelope of the buildings
ther than the tokamak itself. For these kinds of events, techniques are available
ıch that the buildings can be designed to withstand occurrences of whatever
:verity is appropriate for the site that is chosen.

Seismic events differ from other kinds of severe natural phenomena because
ıey could affect the tokamak and its supporting systems. The appropriate
ɔismic magnitude and vibrational signature to be considered for INTOR will be
ite-specific. However, it has been necessary in the conceptual design to select
.ome event for purposes of demonstrating that INTOR can withstand events of
·easonable magnitude. This is particularly important in the design of the magnet
ɔupport system since the seismic loads should be additive to the large dynamic
magnetic loads that occur during pulsed operation. In the structural design, it
was assumed that a free-field acceleration of 0.25 g could occur in a Most Intense
Event. The value of 0.25 g does not seem to be unduly restrictive for siting and so
far it appears that INTOR can be designed to withstand this magnitude of event.

It is expected that INTOR will be shut down as soon as any severe natural
event affecting the site is detected. This policy should reduce the probability of
radioactive releases, because machine shut-down will reduce heat sources and
bring about a more safe configuration. Again, seismic events differ from other
kinds in that a severe seismic event could occur while the machine is operating.

The crash of a large aircraft into the INTOR containment structure is one
accident that could both damage the containment building and lead to release of
tritium or activation products from the reactor systems. This could lead to a
severe accident situation, but adequate analytical and design techniques exist for
calculating the probability of aircraft crashes into the site and for protection
against them.

Safeguards and security aspects, together with consideration of the conse-
quences of sabotage or terrorist attack, have been taken to be beyond the scope
of the INTOR project at this time.

### 4.1.7. Lithium blanket test module failure

The INTOR blanket is to have lithium silicate, $Li_2SiO_3$, as a breeder material.
Safety experiments have indicated that this material does not react with water on

a time-scale such that it could contribute to an accident. However, it is expected that a lithium metal test module may be present in the blanket. The lithium would likely be cooled by helium gas and would be in an independent module. Such a module might have roughly 0.5 m³ of space containing lithium and perhaps several hundred kilograms of lithium could be present. This amount is equivalent to roughly 10 GJ of energy if burned in air.

Lithium safety tests have indicated that the presence of small amounts of lithium metal may be an acceptable risk. Energetic reactions may be expected if lithium contacts water, or concrete, which contains water. However, if lithium is spilled on steel in an air atmosphere, a few minutes are required before the reaction of lithium with the nitrogen in the air becomes severe. The nitrogen reaction with lithium is much less energetic than the oxygen-lithium reaction and apparently blocks or slows down this reaction. The pool temperatures could reach on the order of 1000°C if nothing is done to mitigate the spill. The atmospheric pressure above the lithium does not increase, because the lithium consumes the oxygen and nitrogen in the air. Any hydrogen evolved from water is burned owing to the lithium ignition source. The reactions can be stopped by flooding with an inert gas such as argon.

A successful safety practice for use of a lithium blanket module may be to ensure that any spill will be to a steel sump or catch basin that can be flooded with inert gas. Any concrete surface that could be contacted by the lithium must be lined with steel, and lithium piping can be double-walled to reduce the chances of a leak. Because of the small amounts of lithium present, the separation of the lithium from water, and the use of the above safety practices, it does not appear that a lithium spill will necessarily cause a release of radioactive materials, other than those contained in the lithium, into the containment.

### 4.1.8. Human error

Investigation into the causes and consequences of accidents shows that human error is a primary or contributing factor in the large majority of accident cases. The errors may arise during the design, construction or operational phases of the reactor.

The potential for human errors should be considered to be relatively high for a first-of-a-kind device like INTOR. In many cases, both the designer and the operator are performing a task for the first time without a background of previous relevant experience.

Thus, a disciplined design approach should be followed for INTOR, together with a rigorous quality assurance programme, to minimize the potential for design and construction errors. To aid in reducing errors in the operational phase, careful consideration must be given to the design of instrumentation and control

systems, reactor safety systems, and the man-machine interface. These factors are taken to be beyond the scope of the INTOR Phase-One design effort but must receive full attention in future design.

Of the above sources of energy and of various events, the only ones that seem to be able to cause major releases of radioactivity from the facility are human error and seismic events beyond the design basis. It is believed that the technologies for seismic risk assessment and structural design for seismic loadings are sufficiently advanced so that the facility can be protected to any desired probability or risk level. The same is true for the design of man-machine operational and protective systems to prevent exceeding the safety criteria through operational errors or lack of foresight. Additional analysis in the area of magnet safety seems appropriate. Major structural failures due to large normal or off-normal forces seem very improbable, but are of concern because of the great damage that might follow.

No accidents were identified that would lead to a release of large quantities of activation products to the environment; estimates made of routine releases of activated reactor building gases and fluids to the environment showed that the doses from routine activation product releases were only about 1% or less of the routine tritium releases. Future INTOR studies should include a more detailed analysis of various accident scenarios that may have the potential for causing large radioactive releases. In particular, more analysis and experimental data are needed on the potential for superconducting magnet accidents to generate missiles that could cause radioactive release and containment building damage.

## 5. ENVIRONMENTAL ANALYSIS

### 5.1. Introduction

The purpose of this analysis is to provide a preliminary assessment of potential environmental impacts from INTOR.

Unique environmental features of INTOR that require special consideration include the presence of magnetic fields and the large inventory of tritium. Even if the accident analysis performed until now has not identified accident paths resulting in large tritium releases, it is clear that the major safety issue will be the containment of tritium, in both normal and accident conditions. In particular, the tritium routine releases and the corresponding radiation exposures indicated in the following paragraphs require a high capability for tritium containment, which must be assessed by theoretical evaluations and experimental results from tritium test facilities. The effects of magnetic fields must be further investigated, but interim guidelines on exposures from specialized laboratories are already available

and no fundamental changes in exposure limits are expected. Other environmental items, such as

> release of activated materials
> long-term risk of the activated waste
> amount and effect of non-radioactive effluents

require supplementary investigations, but it is not likely that they will change the actual scenario where the main environmental issue of INTOR is the tritium containment.


## 5.2. Releases in normal operating and maintenance conditions

### 5.2.1. Tritium release

The sources of tritium releases have been evaluated [4] and are reported in Table XI-6 of Chapter XI. They can be summarized for the various plant components as follows:

> Coolant: 3–10 Ci/d
> Buildings: $\leqslant 1$ Ci/d
> Ventilation: 8–16 Ci/d
> Solid waste: $\leqslant 1$ Ci/d
> Total tritium release: 11–20 Ci/d.

These data are within the design goal of a release of less than 10000 Ci/a. However, to ensure that this goal is not exceeded, the fulfilment of several conditions is required. The principal conditions are given below:

(a) The tritium activity in the primary coolant must be limited to $< 30$ Ci/m$^3$ and the coolant leak rate to $< 1$ m$^3$/d. To keep the tritium concentration of the primary coolant acceptably low, a permeation barrier inside the coolant tubes and a tritium recovery system from the primary coolant should be used.

(b) The tritium level in the building must be maintained at low concentrations, $\leqslant 5 \times 10^{-6}$ Ci/m$^3$, and the leak rate must be low. Double containment must be used throughout and triple containment should be used where possible. Tritium areas must be kept at reduced pressure and each of them must have independent tritium recovery units. Tritium containment walls must be lined or coated with tritium barriers. Materials and equipments in the tritium areas must be selected to minimize surface absorption.

(c) The ventilation and air conditioning systems must have appropriate tritium recovery units.

(d) Processes must be selected which produce a minimum of solid waste; the use of organics must be minimized; valves, pumps, etc. should have metal seals; tritiated waste must be reprocessed, where feasible.

Further evaluations of tritium airborne releases based on calculations for TSTA and of tritium aquatic releases based on HWR experience have been made [4]. They give a total release of < 10 Ci/d, which agrees well with the former values.

To illustrate potential impacts from tritium release during normal operation of INTOR, an evaluation of the dose commitment[1] due to a tritium release of 5000 Ci/a was performed. All the released tritium was considered to be oxidized, which is a very conservative assumption. Moreover, from an accidental tritium gas release of about $4.8 \times 10^5$ Ci at the Savannah River Plant (1974) [1], it was found that less than 1% of the tritium was in the oxidized form. The conversion rate from gaseous tritium to oxide was also very low. A release of about $3 \times 10^5$ Ci at Lawrence Livermore Laboratory produced similar results. When developing maintenance procedures, special consideration must be given to tritium release mechanisms in order to prevent exceeding the guide release values.

### 5.2.2. Activation product release

The primary coolant will be activated by the fast neutron flux and large amounts of $^{16}N$ will be produced, $\approx 10^6$ Ci/m$^3$. However, this isotope is not considered to cause significant problems because of its short half-life. The radioactivity level in the primary coolant due to activated corrosion products has been evaluated [3, 4]. Specific activity is estimated to be in the range of from $3 \times 10^{-2}$ to $1.1 \times 10^{-1}$ Ci/m$^3$. Activated corrosion products may leak through the heat exchanger into the secondary coolant or through the pipes into the reactor containment, and can then be released to the environment from the cooling tower and the ventilation system.

An estimation of the activation product release is given in Table XVI-2.

Transformation of primary coolant concentrations to air and airborne release was done with computer codes developed for PWR analysis. Annual air and aquatic releases were taken to be equivalent to the radioactivity contained in 0.13 m$^3$ and 0.25 m$^3$ of coolant respectively.

### 5.3. Exposure to the magnetic field

Information on the biological effects of magnetic fields is not conclusive. Limiting values for personnel are given by interim guidelines (see Chapter XXI). Typical values are:

(a) *For extended exposure periods:* whole body or head: 100 gauss;
    arms and hands: 1000 gauss

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[1] The time integral of the average dose rate to a specific population, resulting from a practice.

TABLE XVI-2. ESTIMATED ANNUAL ACTIVATION PRODUCT RELEASES FROM INTOR

| Nuclide | Primary coolant specific activity (Ci/m$^3$) | Aquatic release (Ci/a) | Atmospheric release (Ci/a) |
|---------|-----------|----------|-------------|
| Co-57 | $4.6 \times 10^{-5}$ | $1.2 \times 10^{-5}$ | $5.7 \times 10^{-6}$ |
| Co-58 | $1.9 \times 10^{-3}$ | $4.9 \times 10^{-4}$ | $2.4 \times 10^{-4}$ |
| Co-60 | $7.0 \times 10^{-4}$ | $1.8 \times 10^{-4}$ | $8.6 \times 10^{-5}$ |
| Cr-51 | $2.7 \times 10^{-3}$ | $6.8 \times 10^{-4}$ | $3.2 \times 10^{-4}$ |
| Fe-55 | $1.7 \times 10^{-2}$ | $4.3 \times 10^{-3}$ | $2.1 \times 10^{-3}$ |
| Fe-59 | $4.3 \times 10^{-5}$ | $1.1 \times 10^{-5}$ | $5.4 \times 10^{-6}$ |
| Mn-54 | $1.5 \times 10^{-3}$ | $3.8 \times 10^{-4}$ | $1.8 \times 10^{-4}$ |
| Mn-56 | $6.5 \times 10^{-3}$ | $1.6 \times 10^{-3}$ | $8.1 \times 10^{-4}$ |

(b) *For short exposure periods:* whole body or head: 2000 gauss; arms and hands: 20000 gauss.
(Exposure to fields above those specified should be avoided and it is recommended that people do not linger unnecessarily in magnetic fields, regardless of their intensity.)

For the general public, values on the order of the earth's normal field, $\approx 0.5$ gauss, could be accepted.

Magnetic field values are strongly dependent on design. For INTOR, magnetic fields of about 500 gauss at 30 m from the centre of the reactor and of about 0.6 gauss at 250 m have been evaluated [3]. It is shown that the site boundary should be several hundred metres in radius to protect the population against accidental release of tritium. Thus, no harm to the public should be expected from magnetic field exposure; however, magnetic shielding may be required for personnel and for instrumentation.

## 5.4. Toxic materials

The use of toxic materials is not expected to create special problems for INTOR. Both in normal operation and after an accident the chemical toxicity of the aerosols released to the atmosphere would be far outweighed by their radioactivity.
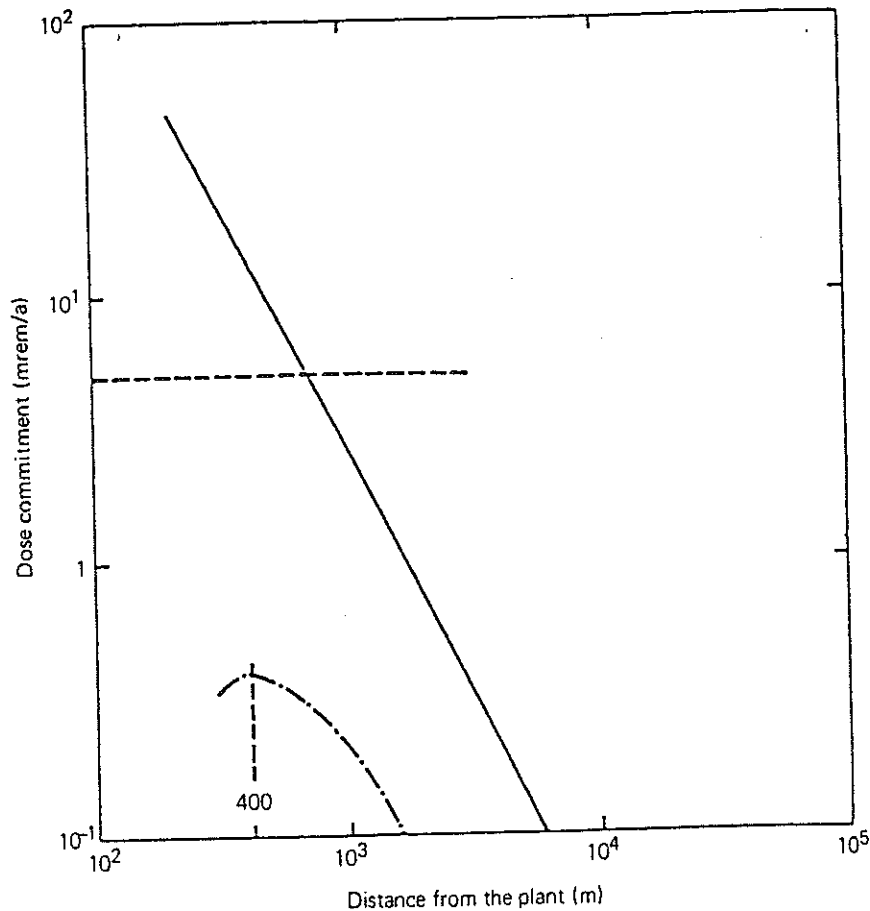
FIG.XVI-4. *Chronic exposure due to inhalation and skin exposure from a routine tritium release of 5000 Ci/a.*
—— *ground-level release,* — — — *current guidelines,* — · — *100 m stack release.*

## 5.5. Accidental releases

The tritium inventories which are considered most vulnerable for release during an accident are those in the plasma chamber, vacuum system and fuel processing system. Values for this inventory of about 0.6 kg have been estimated. Even though these data must be more rigoroudly justified when the INTOR design aspects which have not been developed in detail, an accidental release in the reactor building of all the vulnerable inventory seems unlikely because of the separation and modularization of the tritium systems. The tritium amount released to the atmosphere would be further reduced by the multiple containment and the intervention of the safety systems.

Even if these data must be more rigorously justified when the INTOR design is developed in more detail, a release to the atmosphere of 10 g of tritium in the oxidized form is considered to be a conservative assumption for INTOR accident conditions.

TABLE XVI-3. COMPARISON OF TRITIUM DOSES WITH BACKGROUND RADIATION EXPOSURE

| | INTOR dose (air) | DOE-exposure standard | Background (tritium) | Background (tritium) (%) | Background (all sources) | Background (all sources) (%) |
|---|---|---|---|---|---|---|
| Most exposed individual | $1 \times 10^{-4}$ rem[a] | 0.5 rem/a | $1.2 \times 10^{-6}$ rem/a | 8300 | $8 \times 10^{-2}$ rem/a | 0.12 |
| Population within 80 km | 1 man·rem[a] | | 1.2 man·rem/a[b] | 83 | $8 \times 10^4$ man·rem/a[b] | 0.001 |
| Global population | 5 man·rem[c] | | $2.4 \times 10^{-3}$ man·rem/a[b] | 0.2 | $1.6 \times 10^8$ man·rem/a[b] | 0.000003 |

[a] Committed effective dose equivalent per year of practice.
[b] Collective dose per year.
[c] Dose commitment per year of practice due to airborne release of $2.2 \times 10^3$ Ci of tritium in oxidized form.

Realistic accident scenarios with a large release of activated material have not been identified in analyses performed to date.

## 5.6. Dose due to routine and accidental releases

Effects of a tritium release of 5000 Ci/a during normal operation are shown in Fig.XVI-4. It is assumed that the tritium is oxidized and released from a 100 m stack. The committed dose (time of integration = 50 years) per year of practice is at least one order of magnitude lower than the 5 mrem/a limit of the exposure guidelines. For a ground-level release, which is not a realistic hypothesis, values lower than this limit are obtained for all distances greater than 800 m.

A more detailed analysis for routine releases has also been performed [4]. In this evaluation, a local population of one million adults uniformly distributed within a radius of 80 km was assumed, with a projected 1990 global population of $2 \times 10^9$ living between 30 and 50°N latitude. Average meteorological data were used.

The radioactive sources were:

- 100 $m^3$/a of low-level waste, scaled for INTOR from LWR data
- Activation-product releases as indicated in Table XVI-2
- Tritium aquatic and atmospheric releases of $5.4 \times 10^2$ Ci/a and $2.2 \times 10^3$ Ci/a, respectively, both in oxidized form.

The results of the evaluation are as follows:

(a) The committed dose (time of integration = 50 years) for exposure to effluents from 1 year of operation is $1 \times 10^{-4}$ rem for the most exposed individual and 1 man·rem for the population within 80 m

(b) The committed dose from ingestion of contaminated water and food, and from swimming, for the maximally exposed individual is $2 \times 10^{-5}$ rem per year of practice

(c) The dose commitment for the global population is 5 rem per year of practice.

In all cases, tritium is the major contributing radionuclide. Table XVI-3 shows that the above doses are small fractions of the background radiation exposure. In another evaluation [5], a routine release of 10 Ci/d of oxidized tritium from a 60 m stack gives an exposure of $2 \times 10^{-5}$ rem/a at a distance of 4 km.

Figure XVI-5 shows the total acute exposure due to inhalation and skin exposure from an accidental release of $10^5$ Ci (10 g) of tritium in the oxidized form, from a 100 m stack. Such a release gives dose commitments which are two orders of magnitude lower than the 25 rem limit for a once-in-a-lifetime
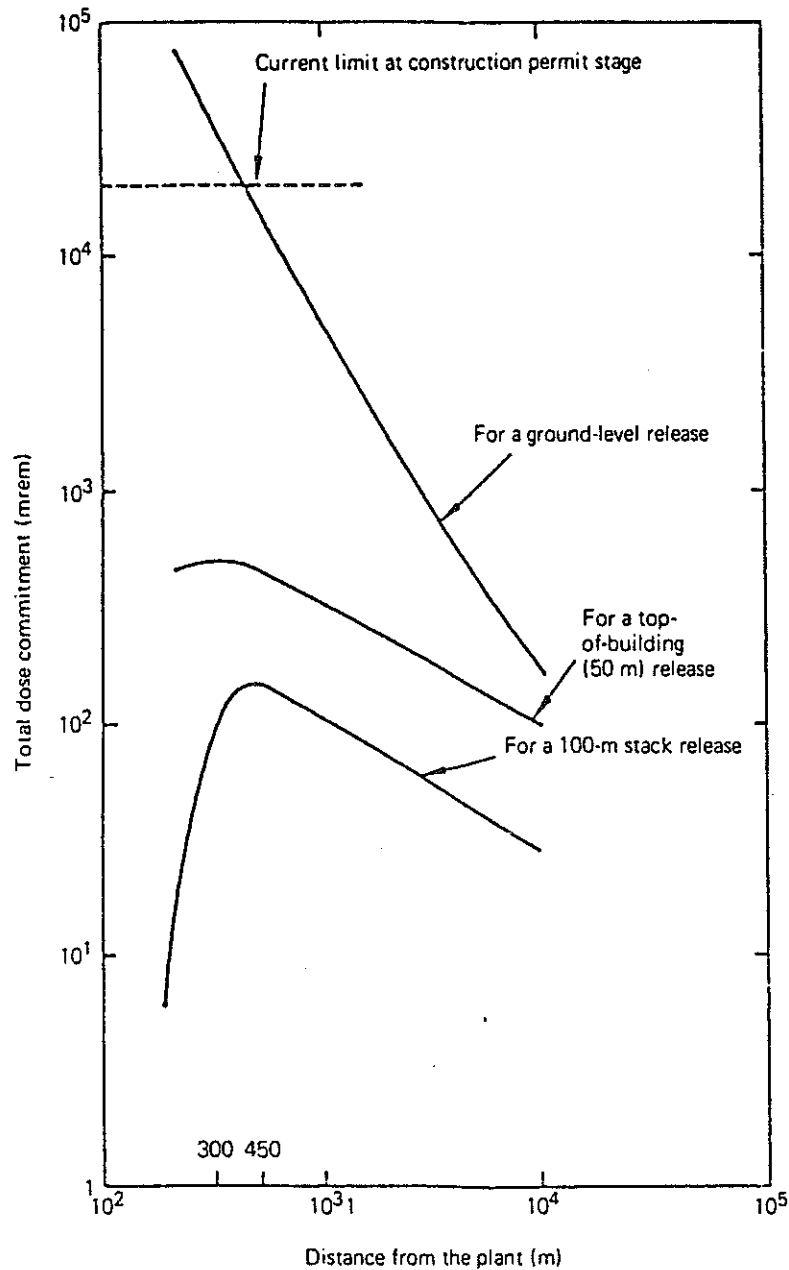
*FIG.XVI-5. Total acute exposure due to inhalation and skin exposure from an accidental tritium release of $10^5$ Ci.*

occurrence. This limit is only exceeded in case of ground releases for distances of less than 450 m.

All these results have been obtained with different hypotheses; however, there is agreement on the following points:

(a) During normal operation, releases of the amounts indicated previously will always give exposures within the regulatory limit for routine exposure, with a site boundary of several hundred metres. It is stressed, however, that the

TABLE XVI-4. BIOLOGICAL HAZARD POTENTIAL (BHP) AND
POTENTIAL HAZARD INDEX (PHI) FOR THE FIRST WALL AND
NEUTRON MULTIPLIER ZONE OF INTOR
Operation time: 10 years

| Isotope | $T_{1/2}$ (years) | Activity $(Ci/cm^3)$ | BHP[a] $(m^3/kW(th))$ | PHI[b] $(kW(th))^{-1}$ |
|---|---|---|---|---|
| Fe-55 | 2.7 | $1.7 \times 10^1$ | $5.0 \times 10^4$ | $2.4 \times 10^4$ |
| Co-60 | 5.3 | 1.2 | $1.5 \times 10^5$ | $1.4 \times 10^5$ |
| Ni-63 | $10^2$ | $2.6 \times 10^{-2}$ | $2.6 \times 10^3$ | $4.6 \times 10^4$ |
| Mo-93 | $4 \times 10^3$ | $6.4 \times 10^{-6}$ | $5.7 \times 10^{-2}$ | $3.5 \times 10^1$ |
| C-14 | $6 \times 10^3$ | $4.8 \times 10^{-5}$ | $1.8 \times 10^{-1}$ | $1.8 \times 10^2$ |
| Nb-94 | $2 \times 10^4$ | $1.4 \times 10^{-6}$ | $3.1 \times 10^{-2}$ | $1.1 \times 10^2$ |
| Ni-59 | $8 \times 10^4$ | $2.8 \times 10^{-4}$ | 4.2 | $5.6 \times 10^4$ |
| Tc-99 | $2 \times 10^5$ | $9.0 \times 10^{-6}$ | $8.0 \times 10^{-2}$ | $3.0 \times 10^3$ |
| Mn-53 | $4 \times 10^6$ | $4.3 \times 10^{-7}$ | $1.9 \times 10^{-4}$ | $1.3 \times 10^2$ |
| Pb-205 | $10^7$ | $2.3 \times 10^{-6}$ | $1.6 \times 10^{-1}$ | $4.0 \times 10^5$ |

[a] The BHP is the total activity in curies per kW(th) divided by the maximum permissible
concentration (MPC) for exposure of the general public in curies per $km^3$ of air. The BHP is
equivalent to the volume of air required to dilute the activity per thermal kilowatt to MPC levels.
[b] The PHI is essentially the time integral of the BHP.

INTOR design must reduce the routine releases to as low as reasonably
achievable (ALARA).

(b) In case of accidents, large quantities of the vulnerable tritium inventory can
be released without giving exposures higher than the limit for a
once-in-a-lifetime occurrence.

## 5.7. Long-term risk of activated waste

Most of the activated material will remain in the INTOR plant until
decommissioning. However, waste disposal requires an assessment of the transport
problems, because of the large quantities of material involved, and an analysis of
the requirements for the waste repository. A preliminary analysis of the environ-
mental impact of activated waste has been performed [2]. The activity at shut-
down, the Biological Hazard Potential (BHP) and the Potential Hazard Index (PHI)
of the long-lived isotopes of the first wall and neutron multiplier zone are
compared with corresponding data for LWR fuel (Tables XVI-4 and XVI-5).

TABLE XVI-5.  BIOLOGICAL HAZARD POTENTIAL (BHP) AND POTENTIAL HAZARD INDEX (PHI) FOR LWR FUEL ELEMENTS
Burn-up:  30 000 MW(th) per tonne of uranium;  initial enrichment:  3.5%

| Isotope | $T_{1/2}$ (years) | Activity (Ci/kW(th)) | BHP[a] ($m^3$/kW(th)) | PHI[b] (kW(th))$^{-1}$ |
|---------|-------------------|----------------------|------------------------|-------------------------|
| Am-241 | $4 \times 10^2$ | $7.3 \times 10^{-2}$ | $4.3 \times 10^5$ | $3.5 \times 10^7$ |
| Am-243 | $7 \times 10^3$ | $5.4 \times 10^{-4}$ | $3.2 \times 10^3$ | $4.2 \times 10^6$ |
| Pu-239 | $2 \times 10^4$ | $7.6 \times 10^{-3}$ | $1.2 \times 10^4$ | $4.8 \times 10^7$ |
| Tc-99 | $2 \times 10^5$ | $3.2 \times 10^{-4}$ | $9.4 \times 10^{-1}$ | $3.6 \times 10^4$ |
| Np-237 | $2 \times 10^6$ | $2.4 \times 10^{-5}$ | $2.4 \times 10^3$ | $8.5 \times 10^8$ |

[a] The BHP is the total activity in curies per kW(th) divided by the maximum permissible concentration (MPC) for exposure of the general public in curies per $km^3$ of air.  The BHP is equivalent to the volume of air required to dilute the activity per thermal kilowatt to MPC levels.
[b] The PHI is essentially the time integral of the BHP.

In the PHI evaluation, the probability of interaction with the environment has been taken to be equal to unity in both cases.  Under these conditions, the risk of fission waste is several orders of magnitude higher than that of fusion waste.

The interaction with the environment is dependent on:

— the biological availability of the radionuclides in the food chain
— the environmental mobility of the radionuclides
— the reliability of the waste repository.

Further investigation is required to evaluate the mobility in the environment of the long-lived heavy metals in fusion waste.

## 5.8.  Natural phenomena

INTOR environmental considerations relative to natural phenomena do not differ from those of other nuclear plants.

## 6.  CONCLUSIONS

An assessment of the safety and environmental aspects of INTOR has been made and a consensus on what the key safety issues are has been reached at the INTOR Phase-One Workshop.  These issues involve the presence of radioactive inventories and potential energy sources and mechanisms that could cause a radioactive release.

SAFETY AND ENVIRONMENTAL IMPACT

The radioactive inventories consist of tritium and activation products. It is estimated that the tritium inventory will be in the range of from approximately 3.4 to 3.9 kg, less than $4 \times 10^7$ Ci. The major uncertainty for this value concerns the tritium inventory in the breeding blanket. Preliminary analysis has shown that releases from the plant can be limited to approximately 20 Ci/d. Releases at this level would not cause any significant dose to operating personnel and to the general public and would not have an environmental impact. Even the consequences of a relatively large (10 g) tritium release in the oxide form from the reactor building would not cause any significant radiological hazards.

The activation product inventory has been estimated to be up to $1.3 \times 10^9$ Ci, with most of the inventory bound up in solid structures of the first wall, blanket and shield. A small portion of this inventory, 24–70 Ci, would be circulated in the primary coolant water through corrosion or sputtering. Also, the reactor building air would contain $^{41}$Ar during operation; however, this activity will be maintained well within the maximum permissible concentration by operation of the ventilation system. A number of accidents have been investigated, including primary coolant system accidents, superconducting magnet accidents, cryogenic system failure, and hydrogen explosions. From these investigations, no credible mechanism has been identified that could cause release of gross quantities of radioactivity from the solid structures; however, additional analysis and research need to be done on the potential effects of magnetic arcs. The technology for handling the activation products in the reactor coolant and building air is well within current capability.

Also, an assessment has been made of potential problems that could be encountered with magnetic fields. From this assessment, it has been concluded that the magnetic fields would have no impact on the environment and the general public; however, local magnetic shielding may be required to keep the exposure of operating personnel within the proposed guideline values and to protect sensitive instrumentation.

On the basis of the safety and environmental assessment of INTOR that has been done to date, sufficient confidence has been gained in the overall safety of the reactor so that more detailed design could proceed. Considerable research, development and analysis remain to be done in order to provide a better quantification of potential radioactive releases. However, no safety concern has been identified for INTOR that should not be amenable to solution by prudent design, construction and operating practices.

## REFERENCES TO CHAPTER XVI

[1]   INTOR GROUP, International Tokamak Reactor: Zero Phase (Rep. Int. Tokamak Reactor Workshop Vienna, 1979), International Atomic Energy Agency, Vienna (1980) 650 pp. *See also:* Summary in Nucl. Fusion 20 3 (1980) 349.

[2]     Euratom Conceptual Design Contribution to the INTOR Phase-One Workshop,
        Rep. Commission of the European Communities, Brussels (1981).
[3]     Japanese Conceptual Design Contribution to the INTOR Phase-One Workshop,
        Rep. Japan Atomic Energy Research Institute, Tokai-mura (1981).
[4]     USA Conceptual Design Contribution to the INTOR Phase-One Workshop,
        Rep. INTOR/81-1, Georgia Institute of Technology, Atlanta, GA (1981).
[5]     USSR Conceptual Design Contribution to the INTOR Phase-One Workshop,
        Rep. Kurchatov Institute, Moscow (1981).

# INTERNATIONAL TOKAMAK REACTOR
# Phase One

REPORT OF THE
INTERNATIONAL TOKAMAK REACTOR WORKSHOP
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HELD IN SEVEN SESSIONS IN VIENNA
DURING 1980 AND 1981