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KEY FEATURES OF INTOR NUCLEAR SYSTEMS*

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1. INTRODUCTION

The International Tokamak Reactor (INTOR) Workshop is a collaborative effort among the USA, USSR, EURATOM, and Japan. The effort is conducted under the auspices of the International Atomic Energy Agency (IAEA). The purposes of the INTOR Workshop are to define the objectives of, assess the technical feasibility of, and develop a design for the appropriate next major experiment in the worldwide tokamak program.

The Zero-Phase (1,2) of the INTOR Workshop was conducted during 1979. The conclusion of this Zero-Phase is that the operation by the early 1990s of an ignited, deuterium-tritium burning tokamak experiment that could serve as an engineering test facility is technically feasible, provided that the supporting research and development activity is expanded immediately.

As a result of this positive conclusion, the INTOR Workshop was extended into Phase I, the definition phase in 1980. The objective of the Phase-I Workshop is to develop a conceptual design of INTOR. Phase I was completed in July 1981 (3,4).

The INTOR Workshop has played a major role in identifying and focusing the attention of the world fusion community upon the major problems that must be addressed before the next major experiment in the tokamak program can be undertaken. The Workshop has also made a major contribution in developing a consensus on the most likely solutions to these problems.

The International Fusion Research Council (IFRC) of the IAEA, which supervises the INTOR Workshop, has recommended that the Workshop be extended into Phase II-A Design, for the period July 1981 through June 1982.

The conceptual design effort for INTOR was broadly defined into three areas: (1) Plasma Physics, (2) Engineering, and (3) Nuclear Systems. This paper is devoted to a summary of the Nuclear Systems effort. The emphasis is placed on the First Wall, Breeding Blanket, and Divertor. References 5-11 should be consulted for additional technical details.

2. INTOR DESIGN SUMMARY

INTOR is conceived to be the maximum reasonable step beyond the next generation of large tokamaks (TFTR, JET, JT-60, and T-15) in the world fusion program. It should provide physics and engineering data relevant to the construction of a fusion demonstration power plant. The specific technical objectives of INTOR are summarized in Table 1. Those technical objectives will be achieved at different stages of INTOR operation. The staged operation schedule proposed for INTOR is shown in Table 2.

A conceptual design has been developed for a device that can fulfill the INTOR technical objectives. The major design parameters are given in Table 3. Ignition is predicted to be achievable with some margin for plasma physics uncertainties. Neutral beams are used for plasma supplementary heating. A single-null poloidal divertor, with the chamber at the bottom, has been selected for impurity control.

The mechanical configuration design was driven from the outset by the requirement to provide maximum access to facilitate maintenance and assembly/disassembly. A semipermanent inboard, upper and lower shield forms the primary vacuum boundary. Twelve blanket sectors fit within this semipermanent shield. [These blanket sectors are partially (outboard and upper) tritium-producing blanket and partially (inboard) heat-removal shield.] The final closure of the vacuum boundary on the outboard is at the outer boundary of the blanket, inside of the outboard bulk shield. Once the outboard bulk shield is removed and the vacuum boundary is cut, each blanket sector can be withdrawn horizontally with straight-line motion through a "window" between adjacent toroidal field coils. The divertor channel is broken up into 24 modules which are removable with straight-line horizontal motion between the toroidal field coils.

Semipermanent, superconducting toroidal and poloidal field coils will be enclosed in a common, semipermanent cryostat, thus completely separating the cold and warm structures. All poloidal field coils will be superconducting and external to the toroidal field coils, except for a set of resistive coils internal to the toroidal field coils.

Table 1. INTOR Technical Objectives

A. Reactor-relevant mode of operation:
   1. Ignited D-T plasma.
   2. Controlled >100 s burn pulse.
   3. Reactor-level particle and heat fluxes ($P_n \geq 1$ MW/m^2).
   4. Optimized plasma performance.
   5. Duty cycle ≥70%.
   6. Availability 25-50%.

B. Reactor-relevant technologies:
   1. Superconducting toroidal and poloidal coils.
   2. Plasma composition control (e.g., divertor).
   3. Plasma power balance control.
   4. Plasma heating and fueling.
   5. Blanket heat removal and tritium production.
   6. Tritium fuel cycle.
   7. Remote maintenance.
   8. Vacuum.
   9. Fusion power cycle.

C. Engineering test facility:
   1. Testing of tritium breeding and extraction.
   2. Testing of advanced blanket concepts.
   3. Materials testing.
   4. Plasma engineering testing.
   5. Electricity production, ~5-10 MWe.
   6. Fluence ~5 MW-yr/m^2 during Stage III for component reliability and materials irradiation testing.

Table 2. Staged Operation Schedule

Stage	No. Years	Emphasis	Availability (%)	Annual 14 MeV Neutron Fluence ($MW\text{-}yr/m^2$)[a]	Annual Tritium Consumption (kg)
IA	1	Hydrogen plasma operation, engineering checkout	10	——	——
IB	2	D-T plasma operation	15	0.16	3.6
II	4	Engineering testing	25	0.31	6.9
III	8	Ungraded engineering[b] testing	50	0.62	13.8

[a]At the outboard location of the test modules.

[b]The objective is to achieve $\sim$5 $MW\text{-}yr/m^2$ within $\leq$10 yr after the end of Stage II. This could be achieved in several ways; the case given here is only representative.

Table 3. INTOR Major Design Parameters

Chamber major radius, m	5.2
Plasma minor radius, m	1.2
Plasma elongation	1.6
Plasma volume, m^3	241
Plasma chamber area, m^2	380
Field on axis, T	5.5
Inner blanket/shield thickness, m	1.2
Outer blanket/shield thickness, m	1.5
Burn average beta, $\tilde{\beta}_t$ %	5.6
Plasma current, MA	6.4
Average neutron wall load, MW/m^2	1.3
Peak thermal power, MW	620
Plasma burn time, s	
Stage I	100
Stage II	200
No. of lifetime pulses	7×10^5
Maximum availability goal, %	50
No. of TF coils	12

3. FIRST WALL SYSTEM

A conceptual design of a first wall system that will survive the total reactor life has been developed for INTOR. The first wall system consists of (1) an outboard region that serves as the major fraction of the plasma chamber surface and receives particle and radiation heat fluxes from the plasma and radiative heating from the divertor; (2) an inboard region that receives radiative and particle fluxes during the plasma burn and the major fraction of the plasma energy during a disruption; (3) a limiter region on the outboard wall that serves to form the plasma edge during the early part of startup; (4) a beam shine-through region on the inboard wall that receives shine-through of the neutral beams at the beginning of neutral injection; and (4) a region on the outboard wall that receives enhanced particle fluxes caused by ripple effects during the late stages of neutral injection. Figure 1 is a poloidal view of the reactor showing the location of the various first wall regions. Table 4 summarizes the operating parameters for the first wall system.

The reference concept for all first wall regions is a water-cooled stainless steel panel (see Fig. 1). The wall thickness of the special regions, e.g., the limiter and inboard regions, is increased to allow for enhanced erosion caused by the preferential heat or particle fluxes. The 20% cold-worked Type 316 stainless steel, which is selected as the structural material, provides adequate radiation damage resistance for

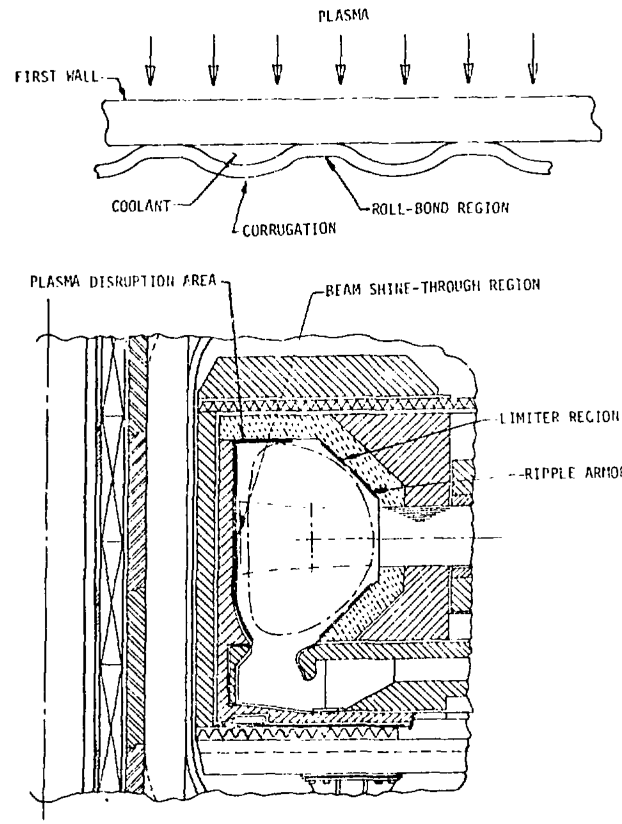

Fig. 1. First wall cross section (top) and first wall configuration (bottom).

full life and an allowable design stress intensity sufficient to meet code specifications for the reference conditions. The thin corrugated coolant channels in the panel-type construction selected for the first wall tend to minimize bending stresses and provide longer lifetime than tubes. The outboard wall is integral with the blanket and serves as the containment for the neutron multiplier.

The erosion rates and thickness requirements for the various regions of the first wall panel have been

Table 4. INTOR First Wall Operating Parameters

First wall	
Total plasma chamber area, m^2	380
Average neutron wall loading	1.3
Radiative power to first wall, MW	40
Charge-exchange power, MW	4
Charge-exchange current, s^{-1} (47% D, 47% T, 5% He, 0.5% C, 0.5% O)	1.3×10^{23}
Charge-exchange flux, $m^{-2}s^{-1}$	3.3×10^{20}
Charge-exchange energy, eV	200
Cycle time (Stage I/Stages II & III), s	145/245
Burn time (Stage I/Stages II & III), s	100/200
Total average neutron fluence, n/m^2	6.8×10^{26}
Total 14 MeV neutron fluence, MW-yr/m^2	6.5
Total number shots	7.1×10^5
Total disruption energy, MJ	220
Disruption time, ms	20
Total number disruptions	1080
Outboard region	
Area, m^2	266
Surface heat flux from plasma, W/cm^2	11.6
Surface heat flux from divertor, W/cm^2	3.4
Total surface heat flux, W/cm^2	15
Limiter region (outboard wall at R = 6 upper and lower)	
Width, m	1
Area (each), m^2	38
Total ion flux, s^{-1}	3
Total heat flux, MW	10
Total ion heat flux, MW	5
Heat flux density, MW/m^2	0.3
Peaking factor	1.5
Typical particle energy, eV	100
Duration, s	4
Period, s	t = 0-4
Ripple region (outboard wall at R = 6 upper and lower)	
Area, m^2	26
Heat flux (ripple = ±0.5%), MW/m^2	0.4
Peaking factor	2
Particle energy (D), keV	120
Duration, s	2
Period, s	t = 8-10
Inboard region	
Area, m^2	114
Surface heat flux, W/cm^2	11.6
Peak disruption energy density, J/cm^2	289
Beam shine-through region (inboard wall)	
Total power (15% of injected), MW	11
Particle energy, keV	175
Duration, s	2
Period, s	t = 4-6
Area, m^2	11
Heat flux, MW/m^2	1

evaluated. The physical sputtering erosion rates are based on effective sputtering yields of 0.020 atoms per particle at 200 eV and 0.0072 atoms per particle at 100 eV for the particle composition given in Table 4. The calculated vaporization erosion caused by a plasma disruption is 8×10^{-4} mm per disruption for a 289 J/cm^2 energy density deposited in 20 ms. An uncertainty factor of two is used to obtain the design erosion allowance. It is assumed that the melt layer formed during a disruption does not erode.

Table 5 is a summary of the lifetime analysis of the first wall system. For the wall thickness requirements necessary to allow for this predicted erosion rates, all regions meet the design temperature, stress and fatigue criteria for full life operation under the reference conditions. The major uncertainty in this design concept relates to the stability of the melt layer formed during a disruption. A grooved inboard wall concept would accommodate erosion of up to 10% of the melt layer (~0.14 mm/disruption). Further research and development are required to confirm the stability of the melt region during a disruption.

4. TRITIUM PRODUCING BLANKET

A partial tritium breeding blanket will be installed on INTOR to reduce the cost of externally supplied tritium. Liquid (lead-lithium-bismuth) and solid breeders were considered. The technology for tritium extraction from a liquid breeder is relatively well understood, but the blanket design is rather complicated, relative to a nonbreeding blanket. The blanket design for a solid breeder is much less complicated, but the uncertainty about radiation effects upon tritium release is a major concern. Research programs are currently underway which should resolve this uncertainty within the next year or so. The solid breeder was selected for INTOR based on the lower risk of the engineering design and the fact that the present uncertainty about tritium release will be resolved in the near future.

The breeding blanket in INTOR uses the top and outboard portion of the 12 removable blanket/shield sectors. To permit a blanket design which is easily adapted to the varying width of the top region and to the changes in neutron wall loading with distance from the midplane, two key features were adopted: (1) a modular approach, by which the blanket is divided poloidally into a number of discrete segments; (2) coolant flow through the module across the full sector width, in the toroidal direction.

An isometric breakout view of a typical breeding blanket module is shown in Fig. 2. Design and operating parameters for the blanket are given in Table 6.

The blanket design features a solid lithium compound breeder, Li_2SiO_3, with lithium enriched to 30% of ^6Li. The solid breeder is fabricated at 70% of theoretical density. Adequate tritium breeding is achieved by using lead as a neutron multiplier. Graphite neutron moderator is used in the breeding zone to minimize solid breeder inventory. Tritium is removed from the breeder by a gaseous helium purge stream. Breeder minimum and maximum temperatures during operation are 400°C and 600°C. This temperature range and the 70% density facilitate tritium diffusion from the breeder. The low lithium inventory helps reduce the part of tritium inventory related to solubility. Pressurized light water (H_2O) is used to cool all parts of the blanket module. Type 316 stainless steel is used for all structural and pressure-carrying components.

The module design integrates the structure and cooling system of the INTOR first wall, blanket neutron multiplier, and blanket breeding zone. Total thickness of the first wall/blanket module is 50 cm. Module length poloidally is nominally 1 m, but this dimension can easily be tailored to fit the modules into the blanket sector geometry.

The first wall panel serves as the plasma-side containment for the neutron multiplier, and its cooling system removes part of the lead neutron multiplier's volumetric heat. The containment at the back face of the multiplier is also an actively cooled cor-

Table 5. *Summary of First Wall Lifetime Analysis*

Region	Total Thickness (mm)	Maximum Erosion (mm)	Maximum[a] Temp. (°C)	Maximum[b] Stress (MPa)	Fatigue Life, Cycles	
					No Erosion	With Erosion[c]
Outboard region	13.4	10.2[d]	310	460	4×10^5	$>10^7$
Ripple region	13.4	10.2[d]	340	460	4×10^5	$>10^7$
Limiter region	14.8	11.6[d]	330	540	2×10^5	$>10^7$
Inboard region	15.6	12.2[e]	280	420	8×10^5	$>10^7$
Beam shine-through region	15.6	12.2[e]	330	420	8×10^5	$>10^7$

[a]Maximum specified temperature = 360°C.

[b]Maximum allowable stress = 650 MPa plasma side, 765 MPa coolant side.

[c]Assumes erosion rate one-half of predicted rate, for conservative design.

[d]Physical scattering.

[e]Physical sputtering plus vaporization.

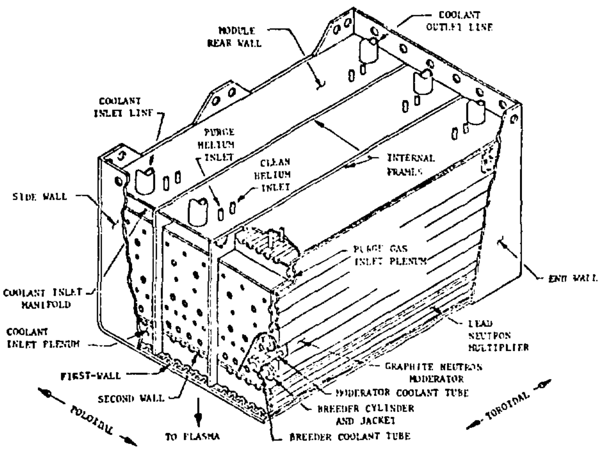

Fig. 2. Reference tritium producing blanket design.

rugated panel (second wall), which removes the remainder of the multiplier's volumetric heat. The first wall and second wall are joined by intercostals which extend through the 5 cm thick multiplier, thus combining the two panels structurally into a relatively deep two-cap beam. The panels, the module side walls, and end walls form a pressure boundary around the multiplier. Low pressure helium in this zone provides good thermal conductance at the multiplier/coolant panel interfaces. Maximum multiplier temperature is predicted to be 290°C, well below the 327°C melting point of lead.

The lithium silicate behind the neutron multiplier is formed in cylinders around single wall stainless steel coolant tubes. There are three separate rows, or banks, of breeder cylinder/coolant tube assemblies which are separated radially within the breeding zone. The graphite neutron moderator is located between banks and between the third bank and the back wall of the module. Separate, small-diameter coolant tubes cool the moderator.

A thin metal jacket surrounds each breeder cylinder. This jacket provides a pressure boundary between the helium purge gas which flows through the breeder cylinder, and the "clean" helium which fills the remainder of the breeding zone and the multiplier zone. The clean helium provides good thermal conductance between the moderator and the jackets. Jacket temperature during reactor operation is relatively low to maintain a low permeability barrier against tritium migration from the purge gas into the graphite.

Table 6. Summary of Reference Design Parameters for Tritium Producing Blanket

Neutron multiplier	
Material	Pb
Maximum temperature, °C	290
Melting point, °C	327
Thickness, m	0.05
Theoretical density, g/cm^3	11.34
Effective density, %	100
Second wall	
Form	Corrugated panel
Structural materials	316 SS
Maximum structural temp., °C	<150
Total structural thickness, mm	2.5
Coolant	H_2O
Coolant outlet temperature, °C	100
Coolant inlet temperature, °C	50
Coolant nominal pressure, MPa	0.7
Region thickness, mm	6.0
Breeding region	
Structural material	316 SS
Maximum structural temp., °C	<150
Breeder material	Li_2SiO_3
Theoretical density, g/cm^3	2.53
Effective density, %	70
Grain size, 10^{-6} m	<1
Breeder max./min. temperature, °C	600/400
Neutron moderator material	Graphite
Effective density, g/cm^3	1.9
Moderator zone gas	He (0.10 MPa)
Region thickness, m	0.43
Coolant	H_2O
Coolant outlet temperature, °C	100
Coolant inlet temperature, °C	50
Coolant nominal pressure, MPa	0.7
Tritium processing fluid	He (0.10 MPa)

Table 7. Divertor Operation Conditions

Design concept	Single null poloidal divertor
Total energy to divertor, MW	80
Ion energy to divertor plates, MW	35
Electron energy to divertor plates, MW	35
Charge-exchange energy to throat and walls, MW	5
Radiation energy to throat and walls, MW	5
Energy to channels, MW	
Outboard	40
Inboard	40
Peak energy flux to channels at null (normal to separatrix), MW/m^2	
Outboard	8
Inboard	4
Total ion flux to divertor, s	5.5×10^{23}
Average energy of ions, eV	400
Peak ion flux to channels at null (normal to separatrix), m^2/s	
Outboard	6×10^{22}
Inboard	3×10^{22}
Tot.. neutral flux to divertor throat and walls, s	1.6×10^{23}
Average energy of charge-exchange neutrals, eV	200
Uniform neutral particle flux, m^2/s	7×10^{20}
Peaking factor of deposition load	2

Table 8. Divertor Collector Plate Design Parameters

Design concept	Tungsten tiles mechanically attached to a water-cooled stainless steel heat sink.
Angular position of plates with respect to separatrix, deg	
Outboard	20
Inboard	45
Peak energy flux to tiles, MW/m^2	3
Peak ion flux to tiles, m^2/s	2.2×10^{22}
Tile:	
Material	Fine grained recrystallized tungsten
Dimensions, cm	$10 \times 10 \times 2.5$
Effective sputtering coefficient	2.2×10^{-10}
Tungsten loss rate (peak ion flux), m/s	7.7×10^{-10}
Lifetime (peak ion flux - 50% duty factor), yr	1.5
Maximum temperature (top surface - end of burn), °C	2030
Minimum temperature (back surface - end of dwell), °C	1150
Heat Sink:	
Material	316 SS
Coolant temperature (in/out), °C	50/100
Dimensions, mm	
Top plate thickness	1.0
Back plate thickness	14.0
Conductance between tile and sink, W/m^2-K	568
Peak heat flux to sink (end of burn), MW/m^2	1.75
Lifetime, cycles	$>10^6$

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Coolant inlet and outlet temperatures are 50°C and 100°C. The coolant is pressurized to 0.7 MPa (100 psi). The moderator coolant tubes and breeder coolant tubes all connect to coolant inlet and outlet plenums located at the module sides (in the poloidal plane). These plenums also connect to the first wall and second wall. At the rear of the blanket the plenums are widened, to serve as a manifold region. These manifolds are connected to large-diameter coolant lines which extend through the bulk shield behind the module.

Helium purge gas inlet and outlet plenums are located between the coolant plenums and the breeding zone. The purge gas flows inside the jackets through several narrow gaps which extend radially through the breeder cylinder. A low partial pressure of oxygen in the 1 atm pressure purge helium reacts with the free tritium at the surfaces of the breeder particles to form T_2O and HTO, which enter the purge gas stream. The purge gas plenums are connected to small-diameter lines which pass through the bulk shield.

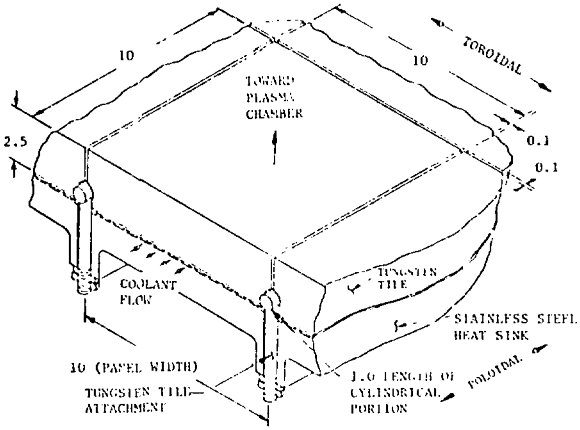
The tritium inventory in the blanket should be about 1 kg, based upon present knowledge of tritium release data. Radiation effects upon tritium release could possibly result in a significantly larger inventory, but these effects are uncertain at present.

5. DIVERTOR COLLECTOR PLATE

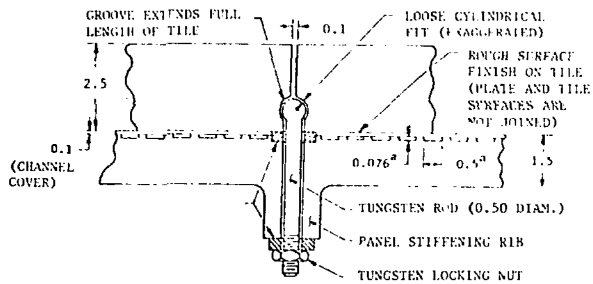
The divertor collector plate design is illustrated in Fig. 3, and the design operating parameters are listed in Tables 7 and 8. The basic plate assembly consists of tungsten tiles in the shape of rectangular plates that are mechanically attached to an actively cooled stainless steel heat sink. Water is used as the coolant in the heat sink. The collector plates are designed with a poor thermal conductance between the tile and the heat sink, which allows the tungsten tiles to increase in temperature to ∿2000°C. At this high temperature, 40-50% of the incident heat is radiated back to the divertor and plasma chambers, thereby reducing the thermal gradient in the tile and the heat flux incident upon the heat sink. The amount of heat that is radiated back into the plasma chamber deposits an additional ∿3.4 W/cm^2 upon the outboard first wall. This additional heat load is predicted to not adversely impact the lifetime of the first wall.

The mechanical attachments allow the tile to freely expand and rotate as the temperature changes during the burn cycle. This design significantly reduces the thermal stresses in the tile and allows the tile thickness to be increased in order to increase the sputtering lifetime. A two-dimensional thermal-hydraulic and stress analysis has been performed on the tungsten tile, and the results indicate the temperatures and stresses in a 2.5 cm thick tile remain in an acceptable range during the burn cycle. The maximum stresses occur during the ramps up or down in power. Unfortunately, the absence of high temperature fatigue data prevent the tile fatigue lifetimes to be accurately estimated.

The principal concern for the low conductance design (and possibly for any design employing refractory metals) is the potentially significant chemical sputtering by oxygen. A simple model of tungsten oxidation predicts an oxidation loss rate approximately three-fourths of the physical sputtering loss rate. The complex nature of the environment in front of the collector plate introduces a high uncertainty on this prediction, however. Additional theoretical and experimental work on chemical sputtering are required to resolve this problem.



(a) Isometric cutaway through typical plate assembly.



[a]Channel dimensions shown are for peak heat flux region only.

[b]All dimensions in cm.

[c]All dimensions are typical.

(b) Cross section through typical plate assembly (full scale; looking in poloidal direction).

Fig. 3. Reference divertor collector plate design.

The heat sink is constructed out of Type 316 austenitic stainless steel. Compared with copper, stainless steel has the advantages of being a standard structural material and of having a known resistance to radiation damage It also has poor thermophysical properties that lead to large thermal stresses. Thermal stress calculations based upon the divertor operating conditions indicate that it is possible to design an austenitic stainless steel heat sink that meets the ASME guidelines for stress and fatigue lifetimes. Based upon available radiation effects data, a heat sink of Type 316 stainless steel is predicted to last the reactor lifetime. In addition, a stainless steel heat sink will experience a much lower magnetically induced torque during a disruption, because of its relatively higher electrical resistivity.

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