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IMPLICATIONS OF STEADY-STATE OPERATION ON DIVERTOR DESIGN

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ABSTRACT

As fusion experiments progress towards long pulse or steady state operation, plasma facing components are undergoing a significant change in their design. This change represents the transition from inertially cooled pulsed systems to steady state designs of significant power handling capacity. A limited number of Plasma Facing Component (PFC) systems are in operation or planning to address this steady state challenge at low heat flux. However in most divertor designs components are required to operate at heat fluxes of 5 MW/m^2 or above. The need for data in this area has resulted in a significant amount of thermal/hydraulic and thermal fatigue testing being done on prototypical elements. Short pulse design solutions are not adequate for longer pulse experiments and the areas of thermal design, structural design, material selection, maintainability, and lifetime prediction are undergoing significant changes. A prudent engineering approach will guide us through the transitional phase of divertor design to steady-state power plant components.

This paper reviews the design implications in this transition to steady state machines and the status of the community efforts to meet evolving design requirements.

I. INTRODUCTION

Fusion experiments such as Tore Supra are already exploring the engineering challenges associated with design of PFCs for steady state operation. As the rest of the fusion community moves from the short- and medium-pulsed experiments towards steady state, PFCs are transitioning from inertially cooled pulsed systems to steady state designs. As shown in Table 1 this is true not only for ITER, but also was true of TPX and will be true of the proposed Wendelstein 7-X, SST-1 in India, STAR-X in Korea, and the long-pulse low aspect ratio machines currently being studied.

Table 1
Some Operating Or Planned Magnetic Fusion Experiments With PFCs Operating In A Steady State Mode

Existing Machines	Long-Pulse Transitional Machines	Steady-State Power Plant
Tore Supra	ITER, TPX, SST-1, W7-X, LHD, STAR-X	ARIES Starlite

Most of the current design difficulties are related to the transitional long-pulse machines. These technologies can be assumed to be well-developed before power plants become a reality. However, power plants will have their own challenges for the divertor designer, such as the need to maintain high exit temperatures for power conversion.

Lower heat fluxes allow the PFC designer to use a mechanically attached tile with an interface layer for heat transfer improvement such as in LHD and SST-1 (1,2). However in most machine designs, divertor components are required to operate at heat fluxes greater than 5 MW/m^2. This heat flux requires an excellent heat transfer across the interface and an intimate bond such as a braze or diffusion bonding between the plasma facing material and the heat sink. This bond often also serves as the mechanical restraint of the plasma facing material. Some of the long pulse experiments being planned will be operated with D-D or D-T plasmas. It is this type of high power density, neutron activated, steady state divertor component that is addressed in this paper. All PFC designs must deal with the issues of large eddy current and halo current induced forces. These are not issues specific to steady state designs. However, the cooling channel circuitry and high conductivity heat sinks required by steady state are often contradictory to designs for minimization of electromagnetic loads.

Steady state D-D or D-T design requires re-evaluation of past approaches in the areas of :

- Thermal Design
- Structural implications of the thermal design
- Component lifetime including operational reliability and erosion life
- Installation and Maintenance

II. THERMAL DESIGN

The divertor designs of TPX and ITER have received major community attention and are representative of current concepts of steady state systems. The steady state heat loads in these divertors are in the range of 5–7.5 MW/m^2 for ITER and TPX, respectively with requirements of short pulses up to 20 MW/m^2 for ITER.

To the rocket engine designer, this power density requirement seems meager, as thrust chambers have been designed up to 147 MW/m^2 (3). Yet fusion's other requirements of high erosion capability, long life, fusion compatible coolants, and reasonable pumping powers, result in systems that require heat transfer enhancement above the 5 MW/m^2 level and have difficulty meeting the 20 MW/m^2 level. The divertor has to be designed not only for the nominal power but also for disruptions (approaching 10^6 MW/m^2 for 0.1 to 3 ms time frames), (4) sawteeth collapses, edge localized modes (ELMs) and power excursions. The significant quantity of predicted erosion in the divertor plasma facing material requires a relatively thick sacrificial plasma facing surface. The optimal material for this plasma facing surface is presently elusive, with carbon based materials, beryllium, and tungsten being the front runners. However each of these materials has their separate difficulty. Carbon presents problems with radiation damage and tritium retention, beryllium with low melting point and little vapor shielding, and tungsten with possible contamination of the plasma.

The peak steady state heat removal capacity of the divertor is limited by: (1) critical heat flux (CHF) of the coolant and cooling scheme, (2) maximum permissible temperature of the coolant channel material, and (3) maximum temperature of the plasma facing material.

The best design would be such that ablation or melting temperature of the plasma facing material is reached before the coolant channel or CHF limit is reached. This type of design operates with greater safety as the coolant channel will not fail.

The are several coolants that may be used in the divertor of transitional machines. Each have their limitations, if a coolant like gaseous helium is used, there is no CHF. Considerable work has been done in last few years to study the feasibility of helium (5). These studies have shown that helium can be used to remove heat fluxes up to 15 MW/m^2 by using heat transfer enhancement techniques. The main concerns with a helium cooled system are the complexity of design, fabrication, size of manifolds and loss of neutron shielding (compared to water), and high pressure accidents.

With water cooling, the CHF is the main concern. The critical heat flux can be increased by lower inlet temperatures, larger velocities and enhancement techniques such as hypervapotron (6), swirl tape (7), and swirl rod insert (8). In most designs, the peak heat flux on the surface of the cooling channel (Wall Heat Flux: WHF) is larger than the incident heat flux (IHF) on the PFC surface by a factor of 1.3 to 1.5. Figure 1 shows that at a flow velocity of 10 m/s, with a **4:1** pitch swirl tape insert used for heat transfer enhancement and sub-cooling of 84°C, a WCHF of about 30 MW/m^2 can be obtained. These flow conditions can support an incident heat flux of 11.5 MW/m^2 (30/1.3/2.0) with a safety factor of 2.0. A safety factor is required due to uncertainties in flow velocity, coolant temperature, critical heat flux correlations, and fabrication tolerances. For the hypervapotron, the ratio of IHF to wall heat flux is about 1.0. Hence, although the WCHF for the hypervapotron is lower than that obtained with swirl tape geometry, the ICHF for both concepts is about equal. The thermal performance of the swirl rod with insert is expected to be similar to that of swirl tape.

The maximum permissible temperature for the coolant channel is controlled by degradation of its material properties and of course by melting temperatures. Higher critical heat fluxes for liquid coolants can be obtained by The maximum permissible temperature for the coolant

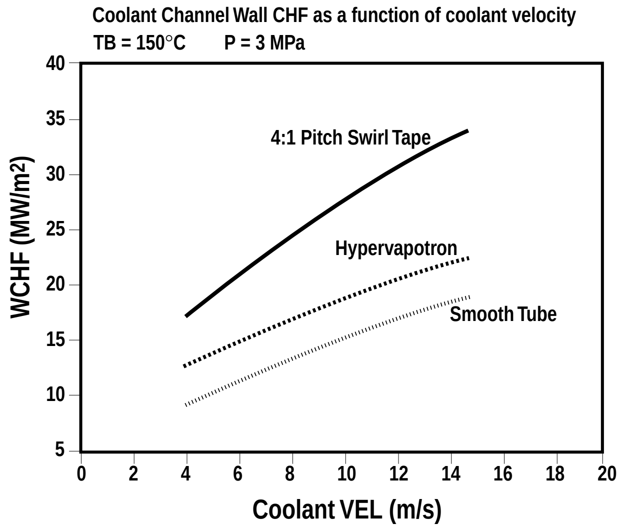

Fig. 1. Coolant channel wall CHF as a function of coolant velocity.

channel is controlled by degradation of its material properties and of course by melting temperatures. Higher critical heat fluxes for liquid coolants can be obtained by controlling the flow parameters. Thus if a scheme with large CHF is used, the peak temperature limit for coolant channel material will be reached before CHF. For Cu-Cr-Zr, the temperature limit for steady state condition is 450°C. Higher temperature limits of 500°C and 600°C maybe allowed during short term transients for Cu-Cr-Zr and Cu-Ni-Be, respectively. ITER transients of 20 MW/m^2 will require a capability of a WCHF of about 52 MW/m^2. However, these are 10 s transients and the PFC/heatsink interface remains below 750°C.

Figure 1 represents constant coolant conditions. In practice, as the coolant velocity is increased, the exit pressure and hence the sub cooling decreases. Therefore the critical heat flux can actually decrease due to increase in velocity beyond a certain magnitude.

The maximum permissible PFC temperature is determined by erosion rates and melting temperatures. The acceptable temperature limits for materials under consideration for steady state operation as PFC's are (9):

Be: 800°C CFC: 1500°C W: 1500°C

Figure 2 shows a plot of PFC temperatures for these materials as a function of incident heat flux. Material properties are used as a function of temperature. Aerolor A05 CFC is assumed. Based on finite element analysis, the peak heat flux in the coolant channel wall (WHF) was assumed to be 1.3 times the incident heat flux for this plot.

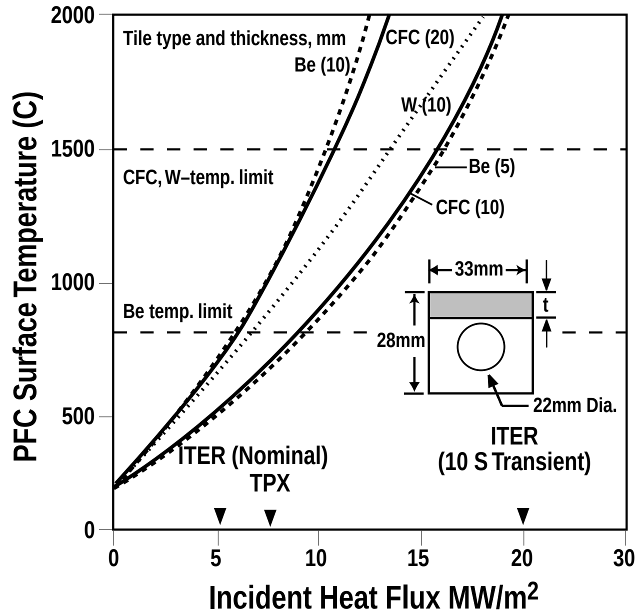

Fig. 2. Limits on steady state divertors.

The flow conditions were: water with a flow velocity of 10 m/s, temperature of 150°C and pressure of 3 MPa. These conditions are consistent with present ITER divertor cooling parameters.

For these coolant conditions and tile geometries, except for the 10 mm beryllium tile, the peak surface temperatures of all materials would be acceptable for both ITER and TPX designs. It can be seen that the tile surface temperature limit can not be met for the present ITER requirement for 20 MW/m^2 for 10 s and all materials will exceed these temperature limits for short durations.

III. STRUCTURAL DESIGN

The structural design of a plasma facing component operating in steady state conditions is largely dominated by the heat transfer solution. The largest stresses are often the thermal stresses caused by differential thermal expansion between the plasma facing surface and heat sink. In an elevated temperature bonding method (such as brazing) maximum stresses are the residual thermal stresses that occur at cooldown from the bonding process. It is desirable to analyze the design of the interface between the plasma facing material and heat sink with the finite element method. However a stress singularity occurs at the free edge of the tile at the intersection of the thin bond between dissimilar materials. (This singularity can be greatly reduced through use of a monoblock or macroblock tile types which limits the singularity to the end of the tile). It is generally felt that the finite element method gives accurate results everywhere except near the free edge (10,11). Yet, at or near this free edge is generally the site of crack initiation. The designer's reaction to this problem has varied from developing stress singularity and stress concentration factors (12) to estimating the stresses by averaging stresses determined by the finite element method over the volume of the material unit cell size (13). Methods of tailoring local interface geometries or using interfaces with low yield stress materials have been used to reduce the effects of the singularity and stress concentration at the free edge (14).

Because of the need to experimentally demonstrate the capabilities of high heat flux systems and partially because of the questionable results of stress analysis of bonded PFC tiles, there has been a large amount of high heat flux testing of prototype high heat PFC components completed world wide. Figure 3 is a summary of the range of thermal cycles covered by present day thermal fatigue testing. Most of this testing is done with power deposition by e-beam or neutral beam. A majority of the components were fabricated with a graphite or CFC surface, although there has been some fatigue testing with beryllium and tungsten plasma facing surfaces. Work is continuing on joining techniques for all PFC surfaces including alternate beryllium tile bonding methods (15).

The peak requirements of the TPX and ITER machines have been included in Fig. 3 for reference. Data has been included for both minimum life required by TPX nd ITER General Requirements Documents (29,36) (assuming three divertors are required during TPX lifetime) and also the maximum operational cycles expected for the machine [10^4 cycles for TPX and the ITER Basic Performance Phase (BPP)]. ITER data has been included for both normal operation and 200 slow transients of 20 MW/m^2 per Ref. (30). In reality both nominal and transient power depositions contribute to a cumulative damage of the component but this effect was not included in this illustration. Disruptive events have also not been included. As shown in Fig. 3 the minimum cyclic lifetime requirements of TPX and ITER are well within the fatigue capabilities shown by small sample testing. Furthermore a limited number of tests have been carried out near the maximum number of fatigue cycles during the operating phase (BPP for ITER).

Because of the significant time that would be required to test actual operational cyclic life, thermal fatigue tests are commonly conducted with a rapid cycle time determined by the time it takes the components to reach thermal equilibrium. Although from Fig. 4 it can be seen that most of these fatigue tests fall short of the actual expected tokamak operating time by large amounts, it is usually argued that failure is dominated by dynamic fatigue and not dependent on operational time.

There may be some need to revisit the decision that short cycle fatigue tests completely describe what is happening in long pulse experiments. For example, initial analyses completed for the TPX divertor indicated that, for a monoblock type of tube-in-tile design, fatigue of the CFC/Oxygen Free High Conductivity Copper (OFHC) interface during plasma operation was not the most likely type of failure (37). In fact the most damaging condition occurred during the 350°C bake out where the copper tube was maintained at higher temperatures than during operation. Elastic-plastic-creep-fatigue analysis (without consideration of more troublesome stress singularities) showed that there was little creep damage during operation and the operational fatigue damage was high but in the

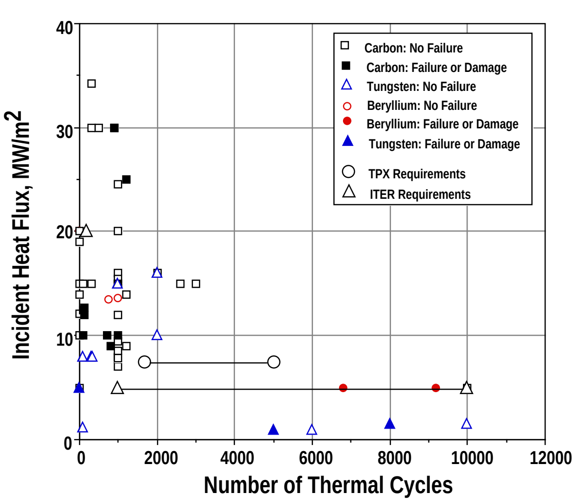

Fig. 3. Thermal fatigue behavior of divertor prototypes (16 through 36).

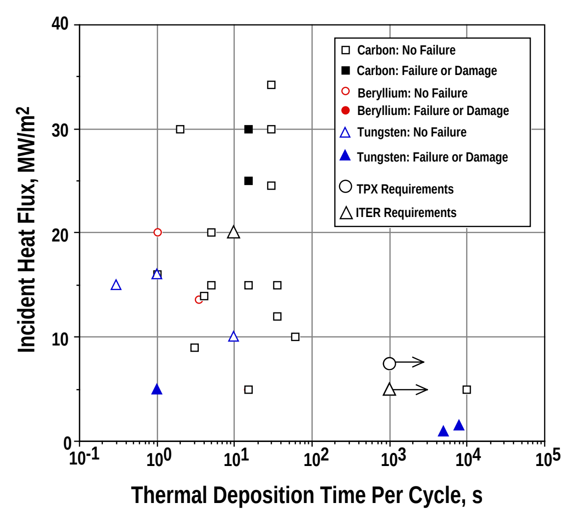

Fig. 4. Incident heat flux deposition time of prototype tests.

acceptable range. However during bake out both creep and for TPX was use of higher strength copper alloy. Thus fatigue damage exceeded allowables. A proposed solution there can be cases where neither the stress level nor fatigue damage occurring during plasma shots are necessarily what limits the design. For designs incorporating material such as OFHC as coolant channels or compliant layers operating at high temperatures with significant cycles or time at high stress levels, creep fatigue may need to be considered. Operational failure of OFHC tubing surmised to be thermal fatigue has occurred in small sample testing (21). In contrast recent data also shows some OFHC heat sinks have been operated for both 10^4 cycles of a 5 s thermal loading and for a continuous 10^4 s without failure (34). It can been seen that all failure mechanisms need to be evaluated for each specific geometry and loading cycle.

Figure 5 illustrates the summation of total power deposition time from the cyclic tests compared to the operational requirements of TPX and ITER. The most test data is one to two orders of magnitude less than what would be seen in TPX or ITER. This prototype data is shown to be even further from the planned operation of these machines when it is recognized that for most of these tests the PFC/heat sink interface was at operating temperature for only a fraction of the test period. In addition most testing is completed with near room temperature water—not the 140°C inlet water required by ITER. The tests' lower temperature coolant allows the interface to operate at a lower temperature.

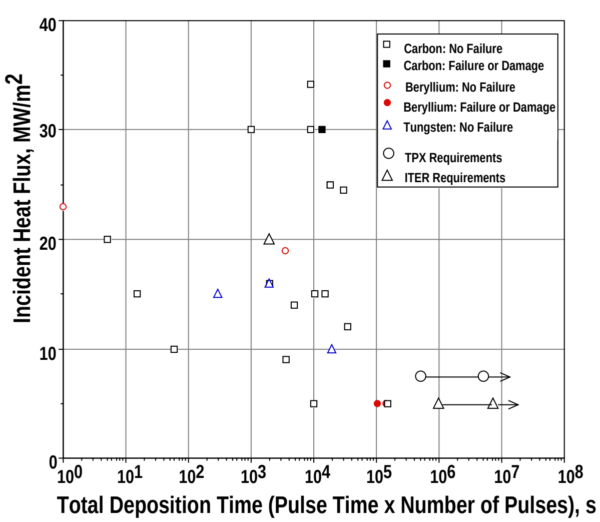

Fig. 5. Summation of total deposition time versus surface heat flux.

Fatigue testing has not yet included the deleterious effects of neutron induced material damage or differential strains. Limited tensile testing was done for tungsten samples irradiated at CM-2 and BOR-60 reactors. Neutron irradiation led to a ductile to brittle transition temperature shift with W embrittlement after low neutron dose (38). Efforts are ongoing at the Judith high heat flux testing facility to investigate the effects of irradiation on carbon and beryllium high heat flux test samples. The irradiation is being done at uniform temperatures of 350°C and 700°C the High Flux Reactor (HFR) at Petten (0.5 dpa) but will not include the irradiation induced differential strains that will occur in a tile operating with a thermal gradient (39).

IV. RELIABILITY

The divertor must provide impurity control, helium and hydrogen pumping, power exhaust and high reliability over its lifetime. Reliability of these complex divertor systems is fundamental to overall machine operation. Not only are these expensive components (the cost of CFC material for the TPX divertor was between \$2M and \$20M depending on the type of CFC selected), the consequences of remote repair and replacement on machine availability is large. Yet at this time development of attachment techniques for some of the ITER divertor high heat flux surfaces is just in the prototype stage. Steady state operational experience is limited primarily to Tore Supra limiters. No large modularized divertor component has been fabricated or used.

Individual tile failures in high heat flux areas will cause removal of complex components. TPX and ITER are designed with approximately 64,000 and over 200,000 (2.5×2.5 cm) tiles on their target plates, respectively. High reliability of these overall systems imply very high reliability of individual tiles. ITER desires to operate with a divertor in a detached regime with power transfer in the radiative mode. But during initial operating scenarios, transients, and semi-attached plasmas, there will be some direct contact of separatrix lines with the tiles. If there is local individual tile failure the low angle of incidence can make edge loading of remaining tiles a severe problem

Rocket engines must have a high probability of success. But even these experience some routine local failures in use. Reusable engines can operate with local failures in the cooling tubing that forms the chamber and require repair after flight to continue use. Tokamaks can not operate with even small leaks in PFCs. Remote repairability in tokamaks is being investigated for some metal systems by the plasma spray method. But real time demonstration of this repair system is distant and must address coating adhesion and overspray clean up problems and methods to spray around diagnostics and other edges. Initial efforts using this technique for beryllium have been conducted at Los Alamos (40).

There is presently limited data on the reliability of large area PFCs that are commercially manufactured. Tore Supra found a initial failure rate of about 2% on their initial inboard toroidal limiter which contained approximately 8,600 brazed graphite tiles that nominally operate up to 1 MW/m^2. This failure rate has increased to 7% through continuing operation (41). They have since replaced a sector of this limiter with a re-engineered system that is anticipated to have much higher reliability. The Outboard pumped limiter fabricated by Sandia National Laboratories under a CEA-US DOE agreement was operated at a 3 MW/m^2 average loading with up to 17 MW/m^2 on the leading edge (42). This device had some residual braze flaws, and experienced tube failure from a Loss of Cooling Accident (LOCA) due to control failure and electron runaways.

In addition to tile reliability, the cooling system reliability has a high impact on machine availability and safety. Loss of local cooling by exceeding the CHF can result in burnout. Thus it is necessary to provide sufficient margin in the as-manufactured component to accommodate a reasonable variability in the manufactured product and to resist designing for operation at the edge of the operational envelope (Fig. 2). Analysis done for ITER showed that coolant loss from other faults resulting in a LOCA or Loss of Flow Accident (LOFA) can result in melting of the heat sink in less than 4 s at 15 MW/m^2. (43). Tore Supra has had the majority of fusion experience in actively cooled PFC and since 1988, has experienced 13 incidents of significant machine leaks caused by fabrication, electromagnetic loads, and excessive local heat flux or electron runaways (41).

The ITER BPP plan assumes that prior to installation on the machine, each sub-system will be fully tested as an independent system (44). It may be difficult to test the divertor components under an accurate simulation of operating conditions.

V. EROSION

In the steady state divertor the limiting factor in component lifetime will probably be the erosion of the target plate tiles. The divertor has to withstand erosion from ion and charge exchange particles and from evaporation and melting. In TPX divertor life was determined by conservatively assuming non-radiative conditions in normal operation. ITER assumes operation of the dynamic gas target radiative divertor with low erosion magnitude in normal operation but with significant erosion in slow transients and disruptive events. With limited data on candidate materials in tokamaks operating under conditions predicted for future machines, determination of the erosion magnitude is largely the result of a series of computer modeling calculations for plasma conditions, with estimates of sputtering, evaporation, vapor shielding, and melt stability.

In TPX a 10 year lifetime dictated by the General Requirements Document (GRD) was incompatible with the erosion that would occur with the CFC 1400°C maximum target temperature and 7.5 MW/m^2 target handling capability. Without consideration of the benefits from radiative divertor operation, calculations by J. Brooks, M. Ulrickson, D. Ruzic (45,46) showing erosion rates of 1.1×10^{-6} cm/s resulted in an ~2.25 year life time of the 1 cm CFC thick tile with the plasma operational parameters required by the GRD (2×10^5 s of D-D operation per year) (29). This lifetime assumed that 0.5 cm of tile surface remained at the end of life.

ITER assumes operation of the dynamic gas target with subsequent reduction in peak heat and particle loads. The ITER divertor must be designed to withstand the thermal loads and erosion from ion and charge exchange bombardment during operation in the BBP without the necessity of complete or partial repair for at least 10^3 pulses at nominal parameters including 200 full power disruptions and 200 slow transients (30,36). Sustained 1000 s burn for 1000 pulses begins in the 6th year of BBP operation, while in the later years this can represent as little as one half a year of operation even without consideration of steady state (36).

Determination of expected erosion and recommendations of optimum materials is an ongoing problem with changing direction. In 1994 it appeared that beryllium was the best candidate for the first phase of ITER BPP operation due to ease of commissioning, plasma compatibility and capability of moderating density limit disruptions and reducing runaway electron production. A 1 cm thick beryllium surface was predicted to withstand greater than 10^3 cycles of operation on the divertor side walls. Vapor shielding and a stable melt layer produced an acceptable disruptive life time (47). However results of the recent JET melting experiment (48) have indicated little vapor shielding with significant melt layer loss. Based on this data new predictions of beryllium lifetime predict 120–230 transient shots and results in the conclusion that beryllium cannot be used for high heat flux divertor components near the strike point. Analysis of W3Re and CFC show lifetimes of 2400–7700 shots and 5800–8200 shots respectively and result in the conclusion that they may be used in the high heat flux area (9).

VI. REMOTE HANDLING

Most present day tokamaks require hands on maintenance for PFC installation or replacement. This is true both of the inertially cooled tiles that are taken off individually in tokamaks such as JET, JT-6OU, and DIII–D, and the modularized PFC/heatsink systems in a steady state hydrogen machine such as Tore Supra. Remote handling is a requirement in either D-T or long pulse D-D experiments due to neutron activation. Analysis of TPX after 2 years of limited operation (3.2×10^{20} D-D neutrons) and one month cool down, revealed a contact dose rate of 25 mrem/h for 304L stainless steel and 100 rem/hr for the Cusil ABA braze that was a candidate for CFC bonding (49). Even though TPX was intended to operate before ITER, it emphasized low activation materials. The TPX design philosophy was to use low activation materials to reduce the need for remote handling and meet the requirement that hands on maintenance be possible a minimum of two years after startup. ITER has also recognized the need to check out their one of a kind systems before operation in a neutron environment. ITER will begin operation in a hydrogen mode.

A fully remote design of the divertor and other internal components is something that the fusion community has not yet demonstrated. The JET design team has led the way in this area. During the installation of the JET Mark I divertor the remote handling boom was used to strip out the major components at the start of the shutdown (50). Following the JET tritium experiment (DTE1, Deuterium-Tritium experiments $1–2\times10^{20}$ neutrons in <4 months) in the fall of 1996, the JET divertor will be reconfigured from a moderately closed slot divertor into the Mark IIB-GB geometry to provide a test of the gas bag concept. This reconfiguration will be done by remote handling since it would take 18 months of cooldown to allow manned work in the vessel (51). However JET has not achieved the fully remote handling capability with rapid cycle time that will be required for future experiments. It is this remote handling capability that needs to be demonstrated.

The increased need for remote handling has been addressed in varying ways that is consistent with the evolution of tokamak experiments towards long pulse machines. After the JET DTE1 phase, individual tiles and tile carriers will be removed remotely. Achievement of a reasonable maintainability through increased divertor modularization has been the goal of both the TPX, ITER, and ARIES designs. As shown in Table 2, increasingly complex designs are attempting to reduce down time and increase tokamak availability by increasing the size of internal modules. However this direction increases the magnitude of the remote handling task, one in which the fusion community has little experience.

The TPX remote handling design was a midway approach between the JET concept and ITER and attempted to shorten machine downtime by modularizing the components. Because of the small vessel and closely spaced toroidal field coils, TPX components had to be sized for removal through large horizontal ports. The divertor was divided into 16 toroidal segments and these segments further divided into two modules each. Removal through the horizontal port required diagnostic and water line hookup to be done inside the primary containment. The major removal time for TPX was moving the handling machine in place—not the actual handling effort. Once the handling machine was in place, operating time of the TPX maintenance system was determined by the requirements imposed by the small working space within the vacuum vessel (52).

Because of its large size and available space between coils, ITER is able to remove divertor components between the coils with out raising them to the mid-plane. This ability does however require toroidal division of the divertor into 60 segments and for 56 of the cassettes to be slid toroidally to a vessel port for removal.

Table 2
Remote Handling Parameters of TPX, ITER, and ARIES

Tokamak	Maximum Module Weight	Replacement Time† 1 Module	Replacement Time† Total Divertor	Nominal Operational Life Before Removal
TPX (52)	360 kg, 1/16 sector	6 days	2 mos	2.25 years
ITER (36,53)	15 tonnes, 1/16 sector	Requirement ≤2 mos goal: 1.4-1.7 mos depending on location	Requirement ≤6 mos goal:1.7 mos	1 to <1 year* equivalent to10^3 pulses (minimum scheduled replacement)
ARIES (54)	Blanket+divertor 150 tonnes, 1/16 sector	2 weeks‡	2 mos	2.5 years based on dpa

†TPX removal time does not include nuclear cooldown (~4 weeks). Double shift labor is assumed for both TPX and ITER. ITER assumes handling machines operating simultaneously in 4 ports (53).
‡Current Estimate, multiple handling machines
*Approximately 3 complete removals during the BPP (36).

In the ARIES concept the modularization of PFC design has continued by combination of the blanket and the divertor into a toroidal segment that is handled as one piece. This modularization allows prediction of continued reduction in replacement time. The trend of reduction in handling time by modularization into larger components is limited and needs to be substantiated.

VII. SUMMARY

Through the design efforts of the transitional machines TPX, ITER, and other long pulse devices, divertor design is making significant headway towards achieving the steady state, remotely handled design concept. Prototype testing of non-irradiated samples have been completed in the range of ITER's initial BPP requirements. However, based on present results, we will be stretching to meet ITER's requirements even in the BPP stage without consideration of the enhanced performance phase (EPP) and the deleterious effects of significant neutron exposure. We have a long way to go in developing a candidate design that exhibits the required reliability. Consideration of irradiation induced effects will only make the concept more challenging. Unfortunately we will never be able to test significantly sized systems under actual conditions except in the tokamak. We should be prepared for learning experiences. Development of alternate low erosion materials should continue. The fusion community should avoid becoming fixated on the current designs of tiles bonded to heat sinks that are the only maturing technology for present designs and continue to develop techniques such as free liquid metal, self-healing, divertor surfaces.

A free liquid metal divertor may not be the future solution but is certainly moving in the correct direction. Although plasma contamination, stability in the magnetic fields, and safety are serious concerns, this type of system has less of the fabrication, reliability, heat transfer, and material concerns of present day systems. Liquid metal limiters have met with some success in the form of both films and droplets in the Russian tokamak T-3M and can theoretically remove heat fluxes up to 100 MW/m^2 (4).

Alternately, further reduction in operating and transient heat fluxes would result in more reliable systems.

ACKNOWLEDGMENT

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