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TO: M. ZARNSTORFF **DATE:** 14-SEP-98

SUBJECT: NCSX
GRAPHITE & BAKE-OUT
OPTIONS FOR
POWER HANDLING
& IMPURITY CONTROL

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Abstract

This report discusses graphite and the associated bake-out requirements for NCSX design and impurity control options. PBX-M and the predecessor tokamak embodiments in the candidate NCSX vessel used similar amounts of unbakeable graphite for power handling. In addition, PBX-M design used a liner-like close-fitting, stainless-steel faced stabilizer shell, and considerable internal hardware. These experiments provide useful bench-mark data for NCSX design considerations. The threshold amount of graphite, below which bake-out is not required, and above which 350°C bake-out becomes desirable for this facility is analyzed making use of the PBX-M results. Then the consequences and options for NCSX are explored proceeding from minimal graphite coverage for power handling to total surface coverage with graphite for versatile power handling and a low-Z first wall.

Contents

1. Introduction

Graphite is used extensively in high power fusion experiments for power handling and as a low-Z cladding for vessel structures. A useful review by R. A. Langley on "Graphite as a Plasma-Facing material in Fusion Experiments", in the Proceedings of the 13th Symposium on Fusion Engineering [1] notes that graphite as a low-Z material is particularly useful in fusion experiments because "it becomes stronger as its temperature increases; its strength is unexcelled at high temperatures; it has thermal and electrical characteristics typical of metals; it has excellent thermal shock resistance for a brittle material; and it is highly inert to chemical attack at low to moderate temperatures".

The main difficulties with its use in fusion applications arises from the relatively high porosity of manufactured graphite (~20). This results in very large connected internal surface areas in the range from 0.1 to 1 m^2/g. It is this effective large internal surface area that allows the copious adsorption and desorption of gases which results in the large investment in wall conditioning time required during typical daily operation of high power fusion devices.

Current practice resulting from the extensive use of graphite in fusion devices for over 20 years, its significant effect on daily operating efficiency, and the ongoing attempts to compensate for its limitations have been reviewed and summarized by D. Post in "Considerations for Bakeout and Conditioning Specifications for In-vessel Components in ITER"[2]. This report (given in Appendix-I) resulted form a world-wide electronic dialogue between experts from the major tokamaks who reviewed and commented on intermediate drafts and the final report. It includes in Table 1 a summary of the bake-out and operational experience of 9 present tokamaks. It is the prevailing consensus that a small amount of graphite in a fusion device can be conditioned without high temperature bake-out, but that the use of graphite beyond some threshold amount makes typical conditioning methods impractical and that bake-out at 350 °C is required to achieve acceptable plasma operating conditions within in a reasonable time.

This report discusses graphite and the associated bake-out requirements for NCSX design and impurity control options. This threshold amount of graphite, below which bake-out is not required, and above which 350°C bake-out becomes

desirable is discussed using the following methodology. First, the baseline vacuum characteristics of the candidate PBX-M vessel with its Liner-like Passive Plate system are summarized. Then the consequences and options for NCSX are explored proceeding from minimal graphite coverage for power handling to total surface coverage with graphite for versatile power handling and a low-Z first wall.

2. Graphite Usage In PDX, PBX, PBX-M

PBX-M was the third embodiment of a tokamak in the present 38 m^3 vessel. The initial configuration was the Poloidal Divertor Experiment (PDX) which started operations in 1979. The PDX design employed about 250 lbs of graphite for armor and limiters with a total plasma-facing area of about 5.2 m^2. PDX had four sets of armor tiles on the midplane opposite each neutral beam [3]. This armor, covering about 70% of the inner wall circumference, consisted of 1.27 cm thick, 10 cm wide, 12 cm long POCO graphite tiles coated with a CVD layer of titanium carbide. It had three functions; neutral beam armor for the vessel inner wall, a neutral beam calorimeter, and as a toroidal belt limiter test for TFTR. Small ATJ graphite poloidal limiters were used in the four divertors.

PBX and PBX-M used four sets of thick semi-cylindrical ATJ graphite mid-plane armor to protect the mid-plane pusher coil from neutral beam power and in-board plasma scrape-off. Power incident on the divertors and passive plates was limited by a poloidal array every 30°. These limiters, mounted on the Passive Plate system, were ATJ graphite, 6 cm wide by 2 cm high. The Passive Plates were 2.54 cm thick aluminum clad with an explosively bonded layer of 0.110" thick 304-SS. Hence, the plasma saw ATJ limiters and a stainless steel passive plate surface (first wall) 2 cm outboard of the limiters. PBX and PBX-M employed 460 lbs of ATJ graphite for armor and limiters with a total plasma-facing area of about 6.1 m^2.

3. PBX-M Liner-Like Passive Plates And Internal Hardware

In order to access the effects of graphite out-gassing in PBX-M during plasma operations, and its implications for NCSX, it is important to understand that PBX-M baseline vacuum characteristics were also determined by out-gassing from a large quantity of internal hardware. In addition, since the NCSX concept involves a large quantity of in-vessel hardware, it is useful to note for NCSX vacuum and

impurity control design that this quantity of PBX-M hardware may have equaled or exceeded the quantity of in-vessel hardware expected to be used in NCSX.

The installation of the electrically isolated PBX-M Passive Plate system required the installation of over 90 sheets, of 24"x30" x 0.030" mica. The surface of this installed mica exceeded 42 m^2, which yields over 84 m^2 of mica surface, if both sides are counted. In addition, the mica installation geometry may have created many regions of trapped surfaces and volumes. This greatly increased the potential for water and impurity retention of PBX-M during open periods.

The PBX-M vacuum vessel has 11 electrical feed-throughs, each containing 36 pins. Five of the 11 electrical feed-throughs provided for the flux loop system and have an average of 40 ft of wire attached to each pin. The subtotal, in-vessel, wire length for this flux loop system (5x32x40ft) is 6400 ft. The other six electrical feed-throughs have wires attached to each pin, averaging in length between 10 to 20 ft for thermocouples and other instrumentation. The subtotal, in-vessel, wire length for these instrumentation wires (6x32x15 ft) is about 2880 ft. Hence, the total, in-vessel instrumentation wire is about 9280 ft or about 1.8 miles.

These wires are not bare. Each wire has some degree of electrical insulation. Many wires have complex insulation and armor to provide electrical insulation from the vacuum walls, thermal insulation from plasma and/or high temperature components, and electromagnetic insulation to reduce EMI. The net effect on the vacuum of this instrumentation wire and associated material is to increase the impurity out-gassing surfaces and the amount of impurity trapping volumes.

The PBX-M vessel has 1 mid-plane coil, 8 upper coils, and 8 lower coils making a total of 17 interior magnetic field coils (IF, DF- 3,-5, -6,-8, and T -1, -2, -3, -4). Each coil is housed in a water cooled metal can. The associated coil insulation and support structures add additional mechanical complexity. This interior magnetic coil hardware increased the inaccessibility of vessel regions to manual cleaning, increased the net impurity out-gassing surfaces, and increased the amount of impurity trapping volumes.

The Passive Plate system consisted of five upper and five lower aluminum (304-SS clad) plates. These plates were electrically isolated from the vessel and each other, and in addition, each plate had an electrically insulated toroidal gap. These electrical insulation requirements were achieved using over 400 ceramic insulators and associated support structures. These many components contributed to cleaning inaccessibility of vessel regions, the net out-gassing surface and the net impurity trapped volume in PBX-M.

4. PBX-M Vacuum Preparations Before Pump-down

The opening of the PBX-M vessel for extended maintenance periods exposed the interior surfaces and trapped volumes to atmospheric gases. Although, the in-vessel work was performed by personnel following standard cleanroom and clean vacuum system procedures, these activities introduced significant amounts of water vapor and hydrocarbons. Prior to closing the vessel, standard vacuum system cleanup procedures were followed. The final scrub of the vessel prior to closing included a through vacuum cleaning and wiping with alcohol. However, due to the extensive internal hardware described above, many highly dense, confined regions of the interior are not accessible to these manual cleaning procedures.

5. PBX-M Initial Pump-down After A Vent

Using the 2600 l/s turbomolecular pump system, the initial PBX-M pump down from atmosphere, following an extended opening, proceeded promptly to a plateau in the low to mid 10^{-6} Torr range that was dominated by water vapor and hydrocarbon out-gassing from the large net internal surface area and trapped volume space. In an attempt to accelerate this process, and achieve baseline vacuum conditions in the ~$2x10^{-8}$ Torr range, hot water at temperatures up to 120 °F was circulated through water coils in the outer lower dome region. All inner-wall water coils that existed during PDX were removed in order to make room for the inner wall structures of PBX-M. This heating procedure was followed from several days to a week or more depending on conditions and the experimental schedule. At the end of this heating period, although the vessel was so hot that it could not be safely touched by human hands, many internal structures were still relatively cold. This occurred, in part, due to the poor thermal conductivity of the stainless steel, the large internal mass structures (beams, bars, plates), and the effective thermal

insulation produced by the extensive passive plate electrical isolation, and the cooling water in the coils which maintained the coil insulating epoxy below its sublimation temperature. Hence, during this heating process, via hot circulating water, only minimal heating of many internal structures occurred via limited conduction and radiation.

Following, or towards the end of the above mild bake-out, a Glow Discharge Cleaning (GDC) procedure was followed. An argon glow discharge was initiated between a probe electrode and the Passive Plate system, and adjusted to stay within the Passive Plate cage. The GDC procedure was effective in removing large quantities of water vapor and hydrocarbons from regions accessible to the glow. This procedure was continued until its effectiveness decreased, usually after about 1 to 2 days. At this point base pressures were in the mid 10^{-7} Torr range. After GDC, the $5m^2$ Titanium Sublimation Getter system was activated and usually was able to decrease the base pressure by a factor of 5 to 10.

After GDC, and the start of Titanium Sublimation Gettering, a Pulse Discharge Cleaning (PDC) procedure was applied for several days to further condition the near- and edge- plasma surfaces. This was continued until, after repeated attempts, it was possible to produce reliable 150 KA circular discharges. These circular discharges were current limited to about 150 KA which limited local surface temperatures to low values. In addition, these circular discharges were not able to reach into the divertor region.

In order to increase the temperature of the near- and edge- plasma surfaces and to reach deeper into the divertor region, it was necessary to increase the plasma current and contour the plasma into a bean shape. However, as the plasma current increased, previously conditioned surfaces were heated to higher temperatures and yielded additional out-gassing, and new, unconditioned surfaces were reached and started contributing to the overall impurity influx to the plasma. Hence, the conditioning process proceeded in slow, iterative, steps, so as to avoid large disruptions that subsequently required a new round of PDC to recover desirable plasma conditions. Once a parameter regime has been achieved, for example, a high beta regime, or a closed divertor regime, any attempt to push to more optimum values yielded large impurity producing disruptions and required a return to this conditioning process.

6. PBX-M Wall-Conditioning After Starting Plasma Operations

After Plasma Operations were initiated, above 4-6 weeks of Ohmically-heated plasmas were used to further condition the walls in order to prepare a suitable target plasma for initial Neutral Beam Injection. Once Neutral Beam Injection was initiated, the higher power levels provided additional wall-conditioning which proceeded for about another 1-2 months in conjunction with L-mode experiments until the first H-mode was achieved. At this point, typical morning base pressures about an hour after the start of the Titanium Getters were about 2 to 5 $x10^{-8}$ Torr.

7. Discussion Of PDX, PBX, PBX-M Experience With Unbaked Graphite

The PDX, PBX, and PBX-M physics results were paced by the above conditioning procedures. In particular, in the case of PBX-M, the high-β toroidal and poloidal experiments seemed to be paced by hydrocarbon and other low-Z impurity influxes from divertor-region as the plasma shape (indentation, elongation, X-points, and strike-points) was changed to access new regimes. Metal impurities from the close-fitting Passive Plate structure, although a concern were perceived as a secondary issue at that time. However, once the IBW induced Core-Confinement (CH) Mode was achieved, it became apparent that metallic impurities were also being confined very well and that metallic influxes needed to be controlled for the next push forward. It was not clear, however, if this impurity influx was due primarily to edge plasma interaction with the entire stainless steel surface of the Passive Plates, or from regions within 1 m of the IBW antennas. Cladding the Passive Plates with graphite and additional boronization were being considered as a tool to resolve this issue.

The effect of adding more unbakeable graphite to PBX-M was estimated approximately by assuming a linear relationship between the total weight of graphite in the vessel and the range of typical wall-conditioning times needed from pump-down to reach the first H-mode.

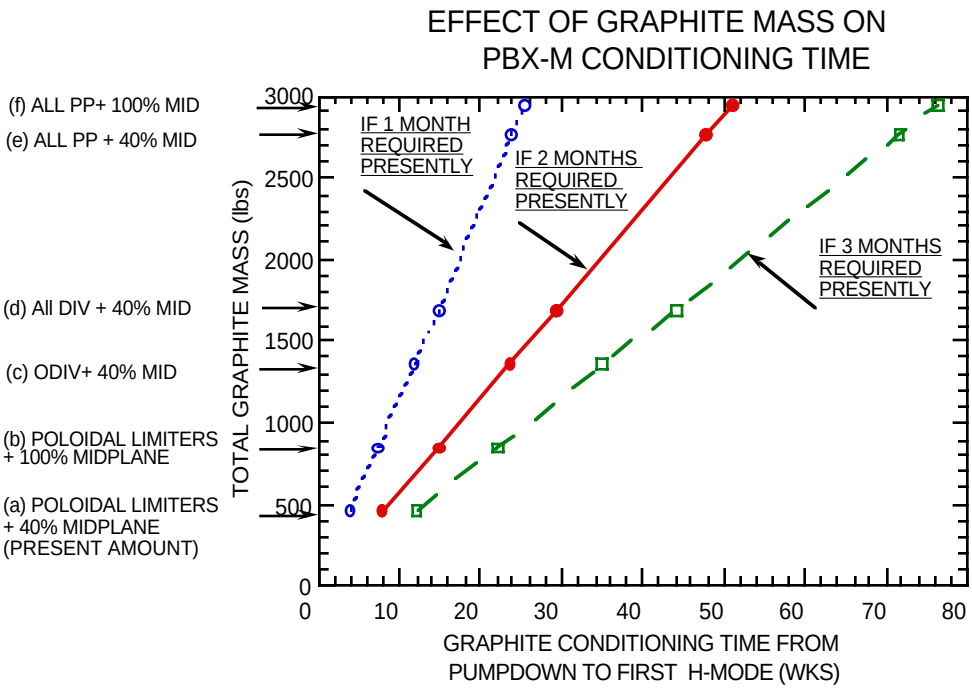

Fig. 1 shows the effect of graphite mass on PBX-M conditioning time assuming (a) the present graphite load, if the conditioning time to reach first H-mode was between 4 to 12 weeks. Shown also are (b) the effects of keeping the poloidal limiters unchanged but increasing coverage of the mid-plane pusher coil from 40% to 100%, (c) adding 100% coverage to the outer divertor while keeping everything also unchanged, (d) adding 100% coverage to the entire divertor, (e) cladding all of the Passive Plates, (f) complete coverage of all plasma facing surfaces. It is seen that in this linear model, it would take approximately 26 weeks of high power neutral beam operation to reach the first H-mode, and probably many more weeks to reach equilibrium conditions.

The conclusions from Fig. 1 are that with the available auxiliary power levels of the PBX-M system about 500 lbs of graphite could be conditioned sufficiently without baking to achieve the first H-mode in a reasonable number of weeks, but amounts beyond this required conditioning times that were a prohibitive fraction of the experimental year.

8. NCSX Power Handling With Half-Liner And Minimal Unbaked Graphite

Consider a simple NCSX proof of principle design with an in-board semi-poloidal stainless steel liner to protect in-vessel coils and inner wall from edge plasma heating. Let this stainless steel liner support a toroidal belt limiter, about 0.5 to 1 m wide, and clad with 1.27 cm thick graphite tiles. In addition, let there be a 6 cm wide graphite poloidal limiter enclosing the plasma every 30°. This would require about 400 to 600 lbs of graphite. If the neutral beam injection angles were such as to allow power to reach the inner limiter, then it would also act as neutral beam armor, similar to that used in PDX [3]. If the neutral beams were aimed tangential then a far wall armor would be needed for low density operation, (requiring about an additional 100 lbs of graphite) although this was never required for PBX-M tangential injection. Hence, a total unbaked graphite usage of 400 to 600 lbs, or possibly as high as 700 lbs would be of order that used in PBX-M (460 lbs) and, hence, using similar high power conditioning procedures, the expected wall-conditioning requirements would be similar to those indicated in Fig. 1. Under these conditions, the amount of daily operating time devoted to wall-conditioning could be reduced significantly if cooling lines on the rear-face of the liner were used to provide a mild Bakeout, e.g., 50-120 °C, and aggressive on-going wall-coating deposition were employed. The use of this half-liner would maximize pumping conductance to the 2600 l/s mechanical pumping system, the lower dome 5 m^2 Titanium Gettering system, the four Neutral Beamline 400 Kl/s cyropanel pumping systems, and boron deposition on the Liner and vessel outer wall. This design would limit plasma shape changes to those allowed by the graphite power handling coverage. Changes in plasma shape and position would heat unconditioned regions and would likely require a new round of conditioning. Hence, daily and weekly experimental planning would be steered by the existing conditioned geometry.

9. NCSX With Full-Liner And Minimal Unbaked Graphite

The use of the same amount of graphite with a poloidally complete liner might reduce impurity influx from the vessel outer wall, especially if the liner would be mildly bakeable while the outside wall was colder. Such a liner would have cutouts for the diagnostic ports, the neutral beam ducts, and RF antennas. If there was an open pathway between these cutouts and the lower dome, Titanium Sublimation

pumping system could be used. The use of staggered poloidal gaps Instead of a poloidally continuos limiter, would provide optically complete coverage while still allowing pumping conductance to the dome and edge region. This geometry would be similar to the poloidal gaps in the PBX-M Passive Plate system. While this geometry would reduce outer wall influxes, the net conditioning requirements would be about the same as those for the semi-poloidal liner, since the same amount of graphite would be involved

10. NCSX Bakeable Full-Liner With Full Graphite Coverage

A toroidally complete liner capable of supporting the full range a NCSX operating conditions could have a surface area of about 40 m^2. If this area was clad with 1.27 cm thick graphite tiles, it would require about 2000 lbs of graphite. Requiring thicker tiles at some locations could easily cause the total graphite mass to approach 3000 lbs. This amount of graphite could be of order the amount indicated in Fig. 1 for complete coverage in PBX-M. It is seen in Fig. 1 that this amount of unbaked graphite would require an impractical amount of high power conditioning, and is well within the regime of graphite mass requiring bakeout at the consensus bakeout temperature of 350 °C [Appendix I]. The liner supporting the graphite tiles need to have cutouts for the diagnostic ports, the neutral beam ducts, and RF antennas. If there is an open pathway between these cutouts and the lower dome, then the Titanium Sublimation pumping system could be used effectively. The use of staggered poloidal gaps Instead of a poloidally continuous limiter, would provide optically complete coverage while still allowing pumping conductance to the dome and edge region. Simple shutters at each of the mid-plane cutouts would reduce the cooling requirements on the vessel wall during bakeout while allowing the vessel mechanical pumping system to exhaust out-gassing during the bakeout.

It should be noted that it was difficult to increase the PBX-M bulk vessel temperature during the mild bakeouts due to the many large beams, plates, and other internal components. Some of these components only received radiant heating from more easily heated internal components. Considering the reverse situation, given the difficulty of removing heat from the large vessel structures, it might be more convenient for NCSX to have simple radiation shields between thermally shielded back-side of the first wall and the vessel wall for heat removal, thereby allowing the vessel to remain below 50 °C during 350 °C bakeout of the liner graphite.

11. NCSX Impurity Control Considerations

In addition to sufficient power handling to allow the desired range of experimental configurations, NCSX should have a comprehensive impurity control plan. This should include suitable pumping systems with sufficiently conductive pumping geometry, sufficient bakeout capability to remove graphite out-gassing as an issue, Glow Discharge Cleaning (GDC) during maintenance periods, automated HeGDC between discharges, and suitable wall coatings such as appropriate boronization and lithiumization techniques.

12. Summary and Conclusions

The PBX-M experience with unbaked graphite indicates that with the power levels of the PBX-M system about 500 lbs of graphite could be conditioned sufficiently without baking in a reasonable number of weeks to achieve the first H-mode, but amounts beyond this required conditioning times that were a prohibitive fraction of the experimental year. The feasibility of a steady state hot first wall should be explored. If a design is capable of bake-outs for 1-2 weeks, perhaps it could also allow operating with continuously hot walls. NCSX liner and graphite designs should include sufficient conductance for adequate pumping, and should be part of an overall impurity control plan.

References

1. R.A. Langley, “Graphite as a Plasma-Facing Material in Fusion Experiments”, Proc. IEEE 13th Symposium on Fusion Engineering, V 1, 52, October 2-6,1989, Knoxville, Tennessee.

2. D. Post, “Considerations for Bakeout and Conditioning Specifications for In-vessel Components in ITER”, ITER JCT, Jan. 20, 1995 Revised May 2, 1995.

3. H. W. Kugel and M. Ulrickson, “The Design of the Poloidal Divertor Experiment Tokamak Wall Armor and Inner Limiter System”, Nucl Techn./Fus., V2, 712 (1982).

Appendix I

The physics basis for baking graphite at 350 ° is given below in the ITER Report prepared from the draft titled "Considerations for Bakeout and Conditioning Specifications for In-vessel Components in ITER" prepared by D. Post, ITER JCT, Jan. 20, 1995 Revised May 2, 1995. Table 2.5.1 lists the conditioning experience of major Tokamaks. The final report was reviewed by the contributors from the major Tokamaks. Their comments/suggestions were summarized in Table 2.5.1. Their interesting verbatium comments/suggestions are given in Appendix II.

2.5. Conditioning, Bakeout and Vacuum

2..5.1. Introduction and Overview of ITER Conditioning Issues:

Conditioning Issues for ITER:

Like all tokamaks, the plasma facing surfaces in ITER will need to be conditioned before operation, after openings, vents, and major leaks, and continually during operation. Conditioning before and during operation and cleaning during manufacture, assembly, operation and maintenance are necessary in ITER: 1) to obtain a low impurity plasma and a low base pressure, 2) to achieve stable and reproducible start-up conditions with adequately low impurity levels and low out-gassing rates during start-up, 3) to control the density and avoid uncontrollable out-gassing, 4) to reduce the hydrogen levels in the plasma during DT operation, 5) to reduce the dust production and inventory and 6) to reduce the tritium inventory in the surface layers of the plasma facing components. The conditioning and bakeout requirements are determined by the materials used and the conditions under which they are used.

Present tokamaks use a variety of techniques to remove loosely bound low Z impurities from the plasma facing components or passivate them, include baking, glow discharge cleaning, pulsed discharge cleaning, RF discharge cleaning, deposition of chemical getters (e.g. deposition of lithium, beryllium, boron, titanium, etc.), deposition of wall coatings such as silicon and tokamak operation. The conditioning and bakeout procedures employed by the present generation of tokamaks is summarized in Table 2.5.1. Proper conditioning plays a key role in optimizing plasma performance[1-4].

The use of these techniques must be adapted to the specific design features and design parameters of ITER (i.e.: single null divertor, B_T = 5.7 T (superconducting coils), I_p = 21 MA, R = 8.1 m, a = 2.8 m, $\kappa \sim 1.6$, $\tau_{pulse} \sim 1000$ s, $P_{fusion} \sim 1.5$ GW and $P_{aux} \sim 100$ MW). ITER differs substantially in these parameters from present tokamaks. It is much larger in volume, surface area,

heating power and pulse length. In addition, the toroidal magnetic field will be on from weeks to months at a time due to the use of superconducting toroidal field coils.

Be, C and W are all active candidates for the plasma facing components. The ITER reference design has Carbon Composite Fiber divertor plates (~100 m^2), tungsten liners in the divertor and baffle (~ 400 m^2), and a Be clad first wall (~1200 m^2) and divertor dome (Fig. 2.5.1). The vacuum vessel and internal structural components are SS316L (~ 7,200 m^2). All of the plasma facing components are mounted on water cooled Cu alloy heat sinks. The pumping system consists of 16 cryopumps located in the divertor ports with an effective pumping speed through the divertor of 200 m^3/s for DT. Some additional pumping will be available through the neutral beam ports. The targeted base pressure for room temperature conditions is 10^{-5} Pa for hydrogen isotopes and 10^{-7} Pa for impurity gases. The outgassing rate is required to be less than $4.2\ 10^{-7}$ Pa $m^3\ s^{-1}\ m^{-2}$ for ~ 7,200 m^2 of surface exposed to vacuum. On JT-60, JET, TFTR and DIII-D, the total outgassing rate is in a range between 4.3×10^{-9} to 3×10^{-7} Pa $m^3\ s^{-1}\ m^{-2}$, indicating that the requirement for ITER is consistent with present tokamak experience. During operation, the neutral pressure in the divertor is expected to be between 0.1 Pa and 10 Pa. The fuelling system is being designed to provide 200 Pa m^3/s of gas injection and 100 Pa m^3/s of pellet injection. The 100 MW of auxiliary heating will likely consist of two 50 MW systems to be selected from three candidate systems: 1 MeV neutral beams, ICRF with a frequency to be picked between 20—90 MHz and ECRH with a frequency between 90 and 170 MHz. In addition, there will be two 3 MW ECRH start up systems. Each system is step tunable between 90 and 140 GHz within a day. The toroidal field magnets can be ramped from 0 to 5.7 T (on axis) in 2 to 3 hours. Rampdown takes the same length of time. The invessel components can be baked at temperatures up to 240°C.

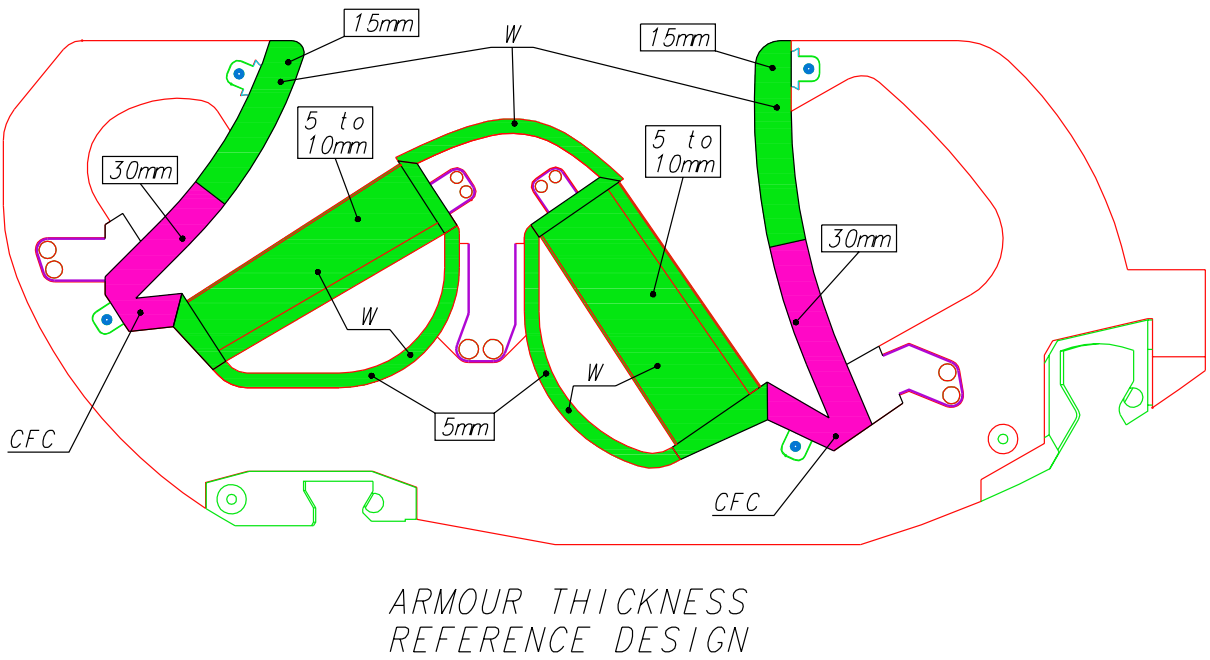

FIGURE 2.5.1. Divertor plasma facing materials: W is tungsten, CFC is carbon fiber composites.

The use of graphite makes conditioning especially critical because graphite is potentially a large source of oxygen impurities. Significant influxes of oxygen to the plasma can result in large radiation losses at the plasma edge leading to shrinkage of the current channel followed by plasma disruptions. The primary candidate sources of oxygen include outgassing of water from in-vessel components, the desorption of CO, H_2O, and molecular ions from in-vessel components due to particle and photon fluxes on those components and water leaks from the water cooling system. The outgassing rate is a strong function of temperature[5]. One of the major sources of water is due to graphite absorption of water from air during an opening and subsequent release into the plasma when the plasma facing components are heated or are struck by ions or electrons. The oxygen level in present experiments is controlled by removing the water from the in-vessel components by baking (both before and during operation), by gettering and by high speed pumping during the discharge.

Graphite components can absorb up to 0.2 Pa m^3/g of water from exposure to water vapor from air, water leaks and outgassing[5,6]. Removal of the water from graphite can be accomplished by a 300 °C to 350 °C bakeout with simultaneous pumping to remove the water vapor. The required bakeout period depends on the level of absorbed water. Bakeout for several days may be necessary following an opening to air, but shorter periods may be adequate to remove water due to outgassing of other components following a disruption. The absorbed water leads to problems with both start-up and

operation and with recovery from disruptions[7,8]. For TFTR, it is necessary before continuing operation after a major disruption to use a series of disruptions ("disruptive discharge cleaning") to heat the limiter surfaces to remove the water that migrated to the limiters during and following a major disruption because it is impossible to bake the TFTR in-vessel components at temperatures above 150°C.

Water can be removed from dense, close packed Be and other metals such as SS and W with bakeout at temperatures of ~ 150-200 °C[9]. In addition the oxygen gettering action of Be mitigates the potential problem that water absorption on Be might produce.

Chemical gettering of oxygen with Li, Be, B, Si, Ti and Cr has also been a successful technique used on many tokamaks for minimizing oxygen contamination[10-13]. Gettering has been generally been used in conjunction with other techniques such as baking and discharge cleaning. Coatings are likely to be incompatible with long pulse ITER operation and will need careful analysis if they are to be relied upon for ITER. An exception to this would be the Be cladding for the first wall, provided the surface of the Be does not become saturated or can be regenerated. Active chemical purging with Diborane has also proved successful on TEXTOR for removing oxygen bound to the surface[2]. B appears to be a very benign wall coating and there is extensive experience with it on many tokamaks[2].

Several experiments with graphite plasma facing materials, including ASDEX/U, JET (with the Mark II divertor) and TORE-Supra, have successfully avoided oxygen contamination problems without high bakeout temperatures by a combination of 150—200°C bakeout, chemical gettering by Boronization and Beryllium and high speed pumping during and between operation. This suggests the possibility that ITER may be able to use a lower bakeout temperature if gettering due to Be and other conditioning materials together with the installed pumping system provide a sufficiently low oxygen contamination level.

Alcator C-Mod has only Molybdenum plasma facing components (no graphite). It has no oxygen problme with a bakeout at 130°C and ECDC (even without boronization). The use of boronization reduces the oxygen level to negliable levels. Boronization also reduces the Mo source rates by an order of magnitude.

Additional vacuum issues

The problems associated with dust and dust clean-up introduce new clean-up and conditioning issues. Due partially to activation problems, tritium

retention, and explosion hazards, the production of dust should be minimized during all phases of construction and operation. Methods for removing dust need to be developed as well.

Control of the absorption and desorption of hydrogen isotopes is necessary for density control and useful for minimization of the tritium inventory[14] and conditioning will play a role in achieving the level of control needed. Contamination of the plasma with hydrogen during DT operation must also be minimized, so that methods for removing hydrogen from the plasma facing components are needed.

Graphite absorbs Hydrogen on the surface and through "co-deposition", a process involving the formation of graphite layers by the deposition of hydrogenated carbon ions and atoms on surfaces during and following plasma discharges[15]. Graphite plasma facing components which have been "loaded" with recycling hydrogen can have significant outgassing rates. Recent experiments on DIII-D and TORE-Supra and other tokamaks demonstrate that particle control can be achieved with sufficiently high levels of active pumping in the divertor [16,17]. Bakeout temperatures of 800 to 1000 ^{o}C are necessary for significant outgassing of hydrogen to occur so that bakeout for hydrogen removal is impractical for ITER[5]. Glow discharge cleaning with He and other gases can remove much or most of the co-deposited hydrogen if the co-deposited layer is sufficiently thin that the ions from the glow discharge can penetrate the layer (several 10's of Angstroms)[14]. However, the deposited layers will probably be much thicker in ITER than present experiments due to the long pulse (1000 s compared to ~ 1-10 s in present experiments) so that discharge cleaning may not be an effective technique in ITER for reducing the absorbed and implanted hydrogen in the graphite. This is because the range of the discharge cleaning ions will be shorter than the thickness of the co-deposited layer. The effects of hydrogen isotopes outgassing from Be can likely be controlled by active pumping during the shot. The outgassing hydrogen isotopes will be ionized in the scrape-off and convected to the divertor where they will be removed. In addition, hydrogen isotopes absorbed in beryllium can be removed by bakeout at temperatures between 200^{o}C and 300^{o}C[9]. Another technique for removing tritium from the wall involves introducing air with some moisture content[18]. The hydrogen in the water vapor appears to undergo isotopic exchange with the bound tritium. Then the water can be removed with the usual techniques.

These tritium retention issues have assess in the divertor DDD (WBS 1.7) and in Federici, et al [19]. Their assessment indicates that the tritium retention in co-deposited graphite could be as large 10 g / pulse. Even though further

analysis is expected to reduce this projection substantially, clean-up of tritium will be a very important fuction of the conditioning systems. As outlined above, baking in vacuum and conventional He glow discharge cleaning is not expected to be very successful. Method presently being pursued include baking in wet air or oxygen. The wet air bake appears to involve primarily isotopic exchange between the hydrogen in the water and the co-deposited hydrogen. Oxygen baking appears to work by oxidizing the co-deposited hydrogen, however, it also appears to require temperatures of ~ 400°C, somewhat above the allowed baking temperature in ITER. Oxygen discharge cleaning, either with glow discharges or with RF formed plasmas is also promising[20] and is being investigated[19].

H is not very soluble in SS and highly insoluble in W, so that hydrogen absorption by these metals will not likely be a problem. However, hydrogen is usually introduced in SS during fabrication (possibly from hydrocarbons), and, if not removed after fabrication, can outgas and provide residual level of hydrogen fuelling that cannot be eliminated[21]. The requirement for a high temperature bakeout of the steel components before assembly needs to be assessed.

The ability to identify and locate leaks is a major design issue and techniques are being developed to find and repair leaks[22].

2.5.2. Conditioning Techniques

The features of ITER place constraints on the type of conditioning that can be used. The use of superconducting magnets means that changing the toroidal field is a significant operation. It takes several hours to ramp the field fully up or down, and only about 1000 TF cycles are planned over the life of the machine. Thus the use of most kinds of glow discharge cleaning will be limited by coil discharging and recharging considerations. RF discharge cleaning may require ramping the toroidal field slowly over a limited range which can be done in less than the full rampdown time. Eddy current losses and heating in the superconductor stabilizers and coil cases will limit the use of time varying poloidal fields needed for pulsed discharge cleaning. The use of water cooling with a maximum allowed pressure of 4 MPa in the pipes, combined with concerns about stress corrosion in the pipes due to salt deposits that might be formed if the water is dried from the pipes, limits the bakeout temperature to ~ 240°C. The large surface area of Be (~1200 m^2) together with surface temperatures of up to 600°C implies that the Be first wall will likely be an active getter for oxygen and other impurities, and may function much like the boron and lithium getters now used in TEXTOR and other machines. Concerns remain that any of the proposed gettering techniques will be of limited value for long pulses, since they may saturate before the end of the pulse. Continual replenishment of the getter or coating

material may be required. For Beryllium, fresh layers of beryllium may form continuously on the portions of the first wall due to migration of Be through the surface layer of BeO, BeC and contaminants on the Be tiles, but the extent of the migration is difficult to estimate. Continual injection of Li pellets or other getter materials may also be necessary. The effects of the buildup of these materials, including their potential for tritium retention, is an issue which will need assessment.

For these reasons, the reference plan is to use a combination of high temperature bakeout at 240°C, glow discharge cleaning when B_t=0, ECRF DC discharges (steady-state ECRH sustained plasmas) or possibly ICRF, reactive gas purging and other other chemical gettering techniques and long pulse ohmic and auxiliary heated discharges to clean the machine. In addition, consideration will be given to thin film coatings of B, Li, etc., and ICR discharge cleaning. Also, cleanliness will be emphasized during manufacturing and component assembly.

Primary Techniques

1) Baking: The baking temperature in ITER is limited to ~ 240°C by water pressure considerations(Fig. 2.5.1). As noted earlier, graphite components can absorb as much as 0.2 Pa m^3 of water per gram of graphite during exposure to air at room temperature[5]. Unfortunately, rapid removal of the water requires bakeout temperatures of ~300°C or higher. Bakeout of graphite components at ~ 240°C may take a long time, perhaps as long as 1 to 2 months (Fig. 2.5.2). The temperature of all of the surfaces will be maintained at 150°C or higher to minimize impurity re-absorption and H trapping.

2) Glow Discharge Cleaning
GDC can only be used when the toroidal field is zero. Therefore, GDC will be primarily useful during the initial commissioning of the tokamak and following major openings and vents. The GDC system consists of eight movable probes with a total current of ~ 100 A[23]. The system is designed for H_2,D_2, He and reactive gases. A typical GDC probe will have a discharge current of ~ 10—20 A for ~ 100 hours and be actively cooled by gas or water with thermocouples to measure the operating temperature. To avoid arcs, the DC power supply will be designed for constant current. GDC will begin gently with low currents, keeping the impurity partial pressures $\leq 10^{-3}$ Pa, then increase the surface flux to ~10^{20} $m^{-2}s^{-1}$ (~ 100 amps) for rapid conditioning.

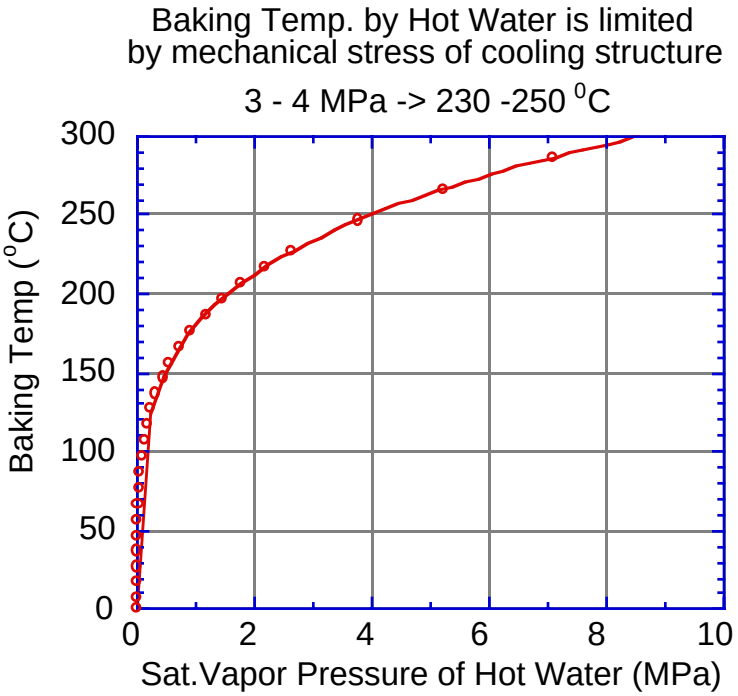

FIGURE 2.5.1 Baking Temperature as a function of the water pressure.

3) RF conditioning:

Two techniques are being considered: ECRF and ICRF. ECRF conditioning has been used on JFT-2M[24] and Alcator C-Mod. On ITER, the two 3 MW 2-second systems, one at 100 GHz and the other at 125 GHz will allow the discharge to reach all of the invessel plasma facing components in the main chamber provided the field can be ramped down as low as 3 T (Figure 2.5.3). If the 170 MHz system becomes available, then all of the chamber can be reached with ~ 3.9 T. A slow sweep of B (~ hours) will be used to clean all in-vessel interior surfaces. Based on exptrapolations from JFT02M and TEXTOR, at least 1 M W of power is required[25]. It might be possible to use this technique to locally raise in-vessel surface temperatures above coolant (H_20) temperature limit. The effectiveness of this technique for conditioning the divertor components needs further assessment. ICRF is being studied on TEXTOR[26] and TORE-Supra[27]. It would use the planned ICRF system if ICRF is chosen for one of the heating options for ITER.

The most effective strategy for using ECRH depends on the mechanism responsible for conditioning the wall. The first proposed mechanism is for the ECRH discharge to create hot electrons which then dissociate H_2 into Franck Condon neutral atoms which then strike the wall and either dislodge bound impurities through kinetic impact like the energetic ions in GDC or react with the surface O and other contaminates, producing H_2O and other

volatile gases which can then be removed by the pumping system. If this mechanism is effective, then it may not be necessary to provide extensive sweeping of the frequency and toroidal field. A few locations of the resonance layer will probably be adequate to provide an energetic neutral flux onto all of the exposed surfaces. The gas pressure should be sufficiently low that the neutral atoms can reach the surfaces before thermalizing or recombining. Pulsed discharges may also be useful to allow the volatile contaminants to be pumped out rather than dissociated (similar to Taylor discharge cleaning[28]). The effectiveness of this technique for removing water trapped in porous graphite below the surface needs to be demonstrated. Atomic hydrogen incident on the surface of a graphite component may not react with Oxygen in the pores of the graphite. This mechanism has the potential for reasonably efficient conditioning of the limiter and divertor plates even if the ECRH discharge doesn't reach all portions of the divertor chamber.

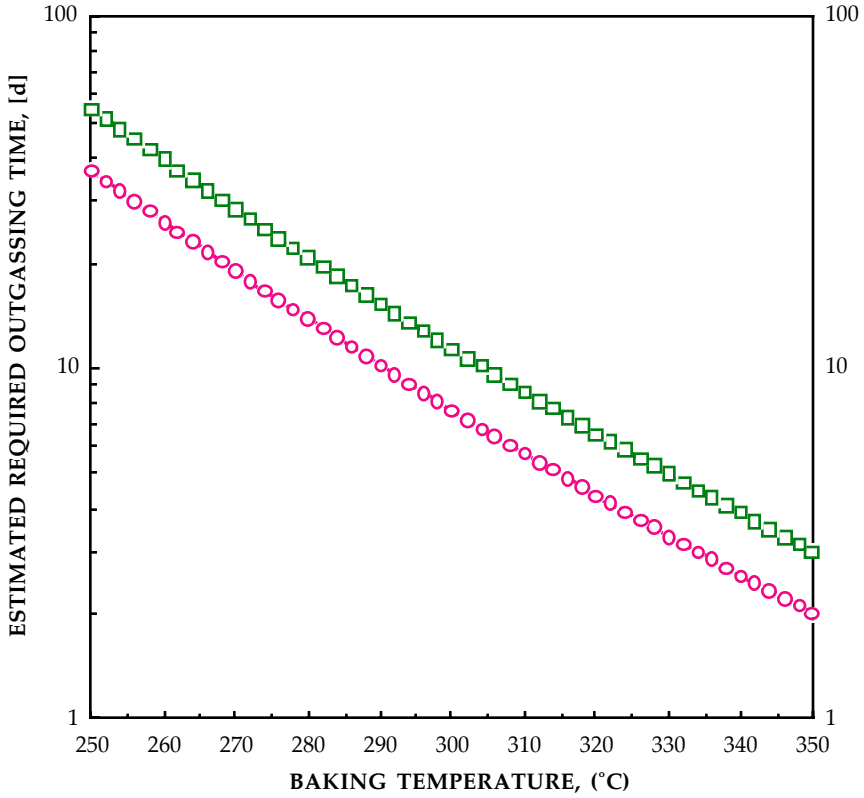

FIGURE 2.5.2 Design range for baking. The range between the two curves includes the uncertainties of the available experimental data and the variation of material properties among different types of carbon-based materials.

For the second mechanism, the ECRH discharge creates fast ions and electrons which strike the plasma facing components reacting with and/or dislodging the surface contaminants. The charged particles also penetrate into the surface layer, displacing and dislodging implanted hydrogen and other atoms. In

addition, the ion and electron flux can heat the surface locally and increase the surface desorption rate. The flux may also sputter clean the surface. If this mechanism is dominant, the width of the resonance layer could be important. If the width is much smaller than the minor radius as is likely to be the case, then the RF frequency and/or the toroidal field have to be swept to ensure that all surfaces are subjected to the ion and electron flux. The effectiveness of this "local" cleaning needs careful evaluation, since local cleaning may just lead to a migration of impurities from the area in contact with the RF generated plasma to areas not in contact with the plasma.

A technique has also been proposed to use ECRF to produce a hot plasma and then direct the heat flux onto specific components using an externally supplied poloidal field from the PF system[29].

Ion cyclotron resonance (ICR) conditioning is also being investigated. It's effectiveness is based on the desorption characteristics of energetic neutral atom impact which perform a function similar to the energetic ions in glow discharge cleaning. It is possible that the random impact angle of a neutral would provide somewhat faster conditioning, be less prone to surface texturing, and would penetrate shadowed regions better than GDC. Calculations indicate that the large charge-exchange cross-section of He at a few hundred eV can result in substantial fluxes of energetic He neutrals from low density, weakly-ionized ICRF plasmas. By directly energizing the He^+ ion, ICR can produce energetic neutrals for desorbing impurities while keeping electron energies and gas pressures low to reduce impurity reionization and provide rapid evacuation from the torus. Since the ions are distributed around the torus at nearly thermal velocities following excitation at RF antennas, long ion mean-free paths or low pressures are required. Model calculations show that pulsed-rf or frequency sweeping (22.6-30.7 MHz) can be used to enhance the He^+ plasma uniformity for the toroidal geometry. Under appropriate conditions, the cumulative effect of cross-field ion drifts for a charge exchange ion cascade will result in local ion flux enhancements of less than 20%. The model predicts energetic neutral fluxes of the same order as currently-used GDC fluxes can be produced using modest RF power (20-30 kW) at accessible RF frequencies.

The ICR approach also appears compatible with the use of He-O_2 gas mixtures for removing C-T codeposits. Experiments have shown this removal to be a two-step process involving oxidation of surface carbon atoms, followed by impact-induced desorption of CO. An ICR plasma will produce oxidation by thermal O-atoms and CO desorption by energetic neutral He. Experiments have shown that minimizing the oxygen concentration in such a mixture results in little molecular oxygen to penetrate material pores, which causes the buildup of residual oxygen contamination. ICR conditioning for the removal of surface species and the erosion of C-T codeposits both need

further assessment. Since this involves the use the planned RF launchers, no new systems need to introduced into ITER. One drawback of this technique is that He-O2 pressure is low so that the ion flux to the surface may be to low to lead to sufficiently rapid clean-up. For this reason, ICRF and ECRF produced oxygen plasmas are being investigated for use in "burning" the graphite and co-deposited hydrogen. This approach has been successful in cleaning up plasma processing reactors[20]. Once oxygen has been introduced, it would be necessary to clean it up prior to plasma operation. However, conventional hydrogen glow discharge cleaning could potentially be adequate. Since no water (from vented air, etc.) would be introduced, the water content of the major graphite components would be small.

4) Chemical gettering and coatings: Introducing chemical getters and coatings into ITER that will difficult to remove will need to be carefully considered, and will probably not be done if the other techniques are effective. The contribution of these materials to the T inventory issue, to dust formation, etc. will need careful assessment. The large surface area of Be in ITER (~1200 m^2) could potentially provide a great deal of gettering for oxygen and other impurities, particularly since it will operate at elevated temperature during ignited operation. However, the gettering effect will depend on the formation of new layers of atomic Be on the surface. BeO is very stable, and the Be surface will likely be saturated fairly quickly. It is planned to introduce diborane into the device with the pumps off to provide active purging of the oxygen on the surface since baking at 240°C may not prove to be completely effective for removing water from the graphite tiles. Lithium injection is also being considered.

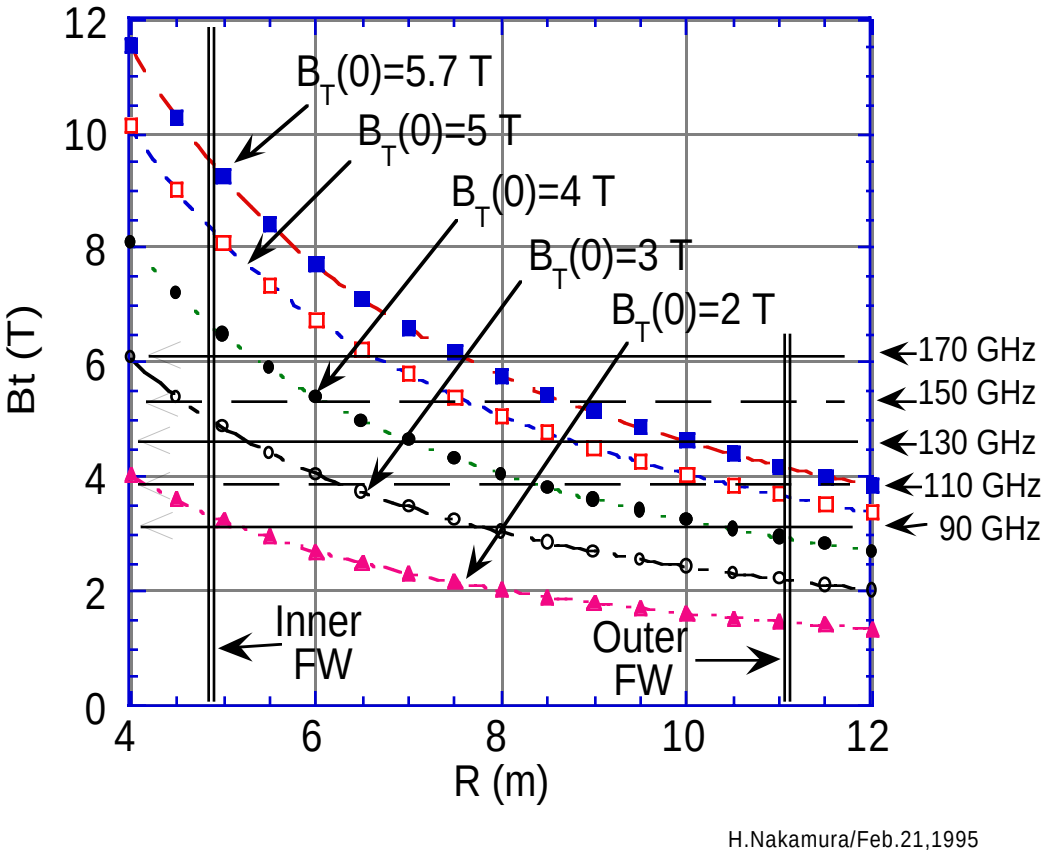

FIGURE 2.3.3. Location of the ECRH resonant layer for ITER for 5 different frequencies and toroidal fields. Sweeping between 5.7 T and 4 T would move the 170 GHz resonant layer from R = 8 to R = 5.5, and move the 110 GHz layer from R = 12 to R = 7 m, thus covering all of the first wall[30].

Secondary Techniques:

All of the techniques described above are being designed into the machine or can easily be implemented at a later date (e.g. Li injection), should they prove necessary. Additional techniques primarily involving pulsed discharge cleaning are also being investigated.

5) Pulsed AC discharge cleaning: This may require RF breakdown assist to facilitate breakdown at low B. Assessment of this technique for ITER would require and analysis of the null magnetics and avalanche with and without rf and analysis of the AC heating of conducting in-vessel structures and of PF and TF coils. A likely scenario involves an 'on' period of pulsing (~ 100 s) followed by a ~ 100-s 'off' period to allow neutral gas pumpout (10 s time constant). A key question is ability to reconnect the PF supply voltage supplies to allow a loop voltage ≥ 20 V at less than the maximum current.

Rudimentary plasma equilibrium control (vertical field and/or elongation) will also likely be needed. The object is to create a cold plasma (10 eV) to facilitate neutral gas exhaust to pump out volatile impurities.

6) Cleaning with tokamak discharges: High frequency (> ~ 1 Hz) pulsed discharge cleaning will be difficult in ITER. The blanket/shield has a loop resistance of ~ 5 micro-ohms, so that penetration of oscillating fields to the plasma will be slower than in present experiments. In addition, eddy current heating of the coil cases and thermal stabilizers in the magnet conductor cables will limit the frequency of the pulsed discharges. These and other issues are being assessed by the ITER design team, but the preliminary result is that rapidly repeating pulses will not be feasible. Cleaning with low current longer pulse discharges (≥ 1 s) will probably be feasible. Provided that ohmic and auxiliary discharges are not quenched by impurity influxes, a long, high power plasma discharge should provide some conditioning for the limiter and divertor plates. However, the discharge cleaning scenarios for ITER need to be further assessed.

Additional issues:

7) Hydrogen contamination: During DT operation, H will be an impurity. At fixed beta, even a 5% contamination level can reduce the fusion power by almost 10%. The potential sources of the hydrogen include: 1) implantation in the plasma facing components during H operation, followed by desorption during DT operation, 2) break-up of water and other hydrogen molecules that are desorbed from the first wall during operation, and 3) outgassing of steel during operation. The development of techniques to remove the hydrogen from the plasma facing components would be very useful. Efficient and rapid isotope change-over between H and DD and between H/DD and DT operation will be very important. The feasibility of baking steel components before assembly to remove the hydrogen trapped in the surface layers during manufacture needs to be assessed.

8) Hydrogen isotopes trapped in the plasma facing components: Conditioning techniques to remove hydrogen isotopes trapped in the plasma facing components will be important. The surface temperatures of the plasma facing components may transiently be as high as 600°C for Be and 1500 °C for graphite. At these temperatures, H will be desorbed very rapidly. If the components heat up too rapidly, the outgassing of the plasma facing components may be sufficiently rapid to over-fuel the discharge and terminate the plasma in a disruption. Even if the outgassing is not strong enough to produce a disruption, it may be strong enough to make density control difficult. Control of the density is essential for controlling the fusion power, accessing the H-mode and maintaining low recycling and low neutral pressure conditions in the main chamber to maximize the confinement time.

In addition, the tritium trapped in the plasma facing components, especially in graphite/Be codeposited layers, can be a very large part of the tritium inventory in ITER. Estimates range as high 10 g of T per pulse for co-deposition rate. Even with lower estimates which are more probably correct, T cleanup will be be required [19]. Almost all estimates lead to the deposition of several kilograms of tritium in the graphite necessitating the development of a clean-up procedure. There has been some success in using He glow discharge cleaning to accelerate the desorption of hydrogen isotopes from graphite in tokamaks, but more efficient methods for removing the tritium need to be developed. In particular, it is important to determine if co-deposited H be removed by He glow discharging and what is the depth of the co-deposited layer and the depth of H implantation after a 1000 s pulse.

This issue is being addressed by the design team and the ITER expert groups. A code has been written which models the tritium uptake in the plasma facing components[31]. The code is being validated using data from various tokamak experiments, including TORE-Supra, JET and TFTR. The preliminary results indicate that the tritium uptake is very sensitive to the surface recombination coefficient for hydrogen[19]. The uncertainties in the calculations should be reduced by extensive code validation and sensitivity studies.

The possibility of using partial venting with moist air or other gases to induce isotopic exchange is being actively considered[18]. As discussed earlier, He-O2 glow discharge cleaning has been used successfully on present tokamaks, but the co-deposited layers on ITER may be too thick for rapid oxidation of the co-deposited layer. Baking in wet air or oxygen also works in present experiments, but again the thickness of the codeposited layers in ITER may be too large for these techniques. In addition, the baking temperatures in ITER (240^{o}C) may be too low for oxygen baking to be effective ($\geq$ 400^{o}C). The high temperatures are required to initiate oxidation of the hydrocarbon layers by O_2 producing H_2O and CO_2 which can pumped out. RF produced oxygen discharges can produce oxidation of hydrocarbon layers by the production of atomic oxygen (O), ozone (O_3) and oxygen ions (O^+), all of which can oxidize hydrocarbons even at room temperature. This technique has been used successfully to clean plasma processing vacuum chambers [20]. These techniques are being explored, and if successful, will be incorporated into the ITER conditioning strategy.

ITER will have an active pumping system with ~ 200 m^3/s of pumping for DT through the divertor. On present experiments, active pumping greatly facilitates control of the density by pumping the outgassing hydrogen before it can fuel the main plasma[32]. On many of these experiments, density control

was very difficult before the installation of the active pumps due to the uncontrolled outgassing of the plasma facing components.

9) Dust removal and clean-up: The potential dispersal of dust activated by the neutron flux in ITER is a safety hazard. The dust can escape during accidental vents and other accidents. Cycling the machine up to air and removing internal components can also allow the dust to escape into the environment surrounding the machine. Tritium implanted in the dust both contributes to the tritium inventory and to the radiological hazard of the dust. Depending on the size of the Be dust particles, the exothermic Be steam reaction could represent an explosion hazard in the event of a water leak. In addition, a large enough quantity of sufficiently small particles of graphite could also pose an explosion hazard during a vent to air. An active program led jointly by the ITER safety and divertor groups is actively assessing the impact of dust production and methods for detecting and removing the dust. The RF produced oxygen plasma cleaning technique is being investigated for "burning" the graphite and hydrocarbon dust and for passivating the Be dust.

2.5.3. Wall Conditioning Strategies and Procedures

Four general type of cleaning tasks are envisaged using the methods described in the previous sections.
1) Preparation (commissioning) of the tokamak for initial operation. The invessel components will be cleaned before and after assembly with water and possibly other solvents. The plasma facing components will be baked at 350°C or higher before installation. After assembly and prior to first plasma all of the plasma facing components will be baked at 240°C followed by glow discharge cleaning (B_T=0) and ECR cleaning (B_T>0).
2) Following major openings, vents and significant leaks. Conditioning will consist of baking at 240°C, followed by glow discharge cleaning (B_T=0) and ECR cleaning (B_T>0).
3) Daily or weekly conditioning during operation. ECR DC conditioning will be the main technique and other techniques will be used as appropriate.
Conditioning will provide an initial set of clean surfaces that will allow the initiation of long pulse, high power discharges. Cleanliness will be maintained during these discharges with divertor pumping which will provide continuous exhaust of the impurities from the walls that are ionized in the scrape-off layer and swept into the divertor.
4) During and between shot conditioning.

2.5.3.1 Pre-Plasma Conditioning and Bakeout Procedures

The options for pre-plasma wall conditioning are listed below in the likely order of application prior to the first tokamak plasma.

Prior to "first plasma":

1) Prior to closure of the vacuum vessel and initial pumpdown: All of the invessel components should be cleaned by water and other solvents to remove oil and other contaminants prior to and after installation. Manufacturing and assembly techniques which minimize contamination are to be used to the greatest extent practical. Prior to installation, a high temperature bakeout of the plasma facing components (especially graphite) should be performed. Dust production during assembly and installation should be minimized, and an effort should be made to remove as much dust as possible following each assembly step. Initial leak checking should be done before putting water into the cooling tubes. After water has been put into the pipes, leak checking will require the lengthy process of draining the water from the cooling pipes and drying the pipes with hot gas before He can be introduced into the pipes to begin leak checking.

2) Cleaning following initial pumpdown: In-situ baking of VV and in-vessel components and ports at temperatures of up to 240°C for a period of weeks to 2 months as required is the first step of the pre-operation vacuum conditioning procedure. A high temperature bakeout (≥350°C) for any graphite based material is extremely desirable, but is not possible without some technique for heating the surface of the plasma facing components. The temperature of the cooling water is limited to 240°C by the maximum pressure allowable in the water pipes. RGA monitoring of exhaust gas composition will be needed to monitor the cleanup process.

3) DC glow-discharge cleaning requires B = 0 and has the following characteristics: internal glow electrode(s), I = 10-100 A at ~ a kilovolt, gas species of He, H , Ar, possibly O_2,....) and a period of 10-1000 hours. It should be done with 'normal' elevated wall temperatures, with RGA monitoring and with active pumping. It may be interleaved with bakeout periods. One could use special electrodes in divertor cassettes and port stubs to get the glow discharge into as many corners and sheltered surfaces as possible. Measurements of the plasma density and temperature (1-10 eV) with Langmuir probes or other techniques and spectroscopic diagnostics during the glow and discharge cleaning phases are needed to monitor and understand the progress in cleaning up the tokamak in-vessel components.

Following the "first plasma":

After the "first plasma", begin reference tokamak operation (breakdown with full toroidal field), and use 4), 5) and 6) below (all variants of 'tokamak' operation) for further plasma facing component surface cleanup or if the gas/impurity influx stifles the plasma current startup.

4) RF-assisted cleaning: Use the available EC (or IC) power (1 to 50 MW), CW or pulsed, and ramping the B field to optimize breakdown and deposition as a function of R. The cleaning period would be day(s) to week(s).

5) Pulsed AC discharge (Taylor). This can be accomplished by the application of pulsed 'square-wave' or 'sine-wave' ac voltage to PF coils (~ 20 V/turn) at frequencies on the order of 0.1 Hz to obtain bi-directional plasma currents of approximately 0.5-1 MA @ ~ 10 V/turn. A toroidal field of ~0.5 T would produce a $q \geq 3$ plasma. The cleaning period would be day(s) to weeks(s). The total pulses (1/2 ac cycle) would be ~100/hr for a total of 2500 — 10^5.

6) Pulsed mini-tokamak and full tokamak discharges. Mini-tokamak discharges involve repeated mini-pulses, with properties similar to the "Taylor" discharge cleaning pulses (≤ 2 MA, B = 0.5-1.0 T), with a rising plasma current and slightly higher field to produce disruptive termination. The period is day(s) to weeks(s) with a pulse frequency of ~100/hr for a total number of pulses of 2500 -10^5. Plasma equilibrium control issues, AC losses in the vessel, and other structures and the increased PF supply voltage requirements and coil fatigue limits need analysis.

7) Chemical getters and coatings: If necessary, utilize coatings such as Li, Be, B, Si, etc. to getter O and other impurities and to coat the plasma facing components. The ~1200 m^2 surface area of Be may be an effective getter.

2.5.3.2 Discharge cleaning and bakeout procedures following a major opening

A combination of the techniques proposed for plasma commissioning will be used to bring the tokamak back into operation following a major opening. This should take less time than original commissioning, but may be lengthy nonetheless. Where possible, exposure to air should be avoided as much as possible. If vacuum is broken, then backfilling with dry nitrogen or other inert gases during maintenance is recommended. Exposure of the graphite components to water should be minimized. Replacement components should be cleaned and baked at high temperatures before installation.

1) Invessel cleanup: A strong effort should be made to clean up contaminants introduced into the vessel during and before maintenance, especially following a water leak or other contamination of the invessel components. Dust produced during the previous run period and during maintenance should also be removed.

2) Baking: Exposure to air will introduce water and other contaminants. A bakeout period of weeks to months will be needed for graphite, depending on the amount of water that needs to be removed from the graphite

components. Bakeout of other components at temperatures of ~ 200^{o}C is also required.

3) Conditioning: Conditioning will be done after bakeout beginning with conditioning with B=0 (which may be interleaved with bakeout) followed by RF and plasma discharge cleaning and chemical getters if necessary. Isotope control (H removal) will be an important goal of conditioning. Conditioning to remove Tritium from the PFC's before the opening will be used if possible.

4) Chemical gettering and coating: If needed, chemical getters and coatings such as Li, Be or B can be used.

2.5.3.3 Daily/weekly conditioning procedures

Consistent with the operating experience on present tokamaks, conditioning between shots and during operation may prove to the useful.

1) Baking

Given the time required for set-up for baking, baking will probably not be feasible for short times (on the order of a day). However, it may be useful if the tokamak is under vacuum, but not operating for several days.

2) Conditioning

RF conditioning, discharge cleaning and chemical gettering should all be feasible during operation, and possibly between shots. The Be first wall should provide some gettering, and the 1000 s tokamak pulse will provide some conditioning.

3) Chemical gettering and coating: If needed, chemical getters and coatings such as Li, Be or B can be used.

2.5.3.4 During and between discharges conditioning procedures

The major techniques used during and between discharges will include the benefit of the 1000 s discharge, the active pumping during the discharge and the Be gettering and pumping during the discharge.

1. Long pulse, high current, high power discharge cleaning.

The ITER discharge will have a long pulse, as long as 1000 to 10000 s. The heating power will be high (up to 400 MW). The walls and divertor plates will have large heat loads, and the particle loads in the divertor will be large. These will help clean contaminants off the wall and divertor plates during the discharge.

2. Active pumping during the discharge

Sixteen active cryo-pumps will provide 200 m3/s of pumping to remove contaminants from the recyling neutrals. Contaminants from the wall will be ionized in the scrape-off layer and removed to the divertor where they can be pumped.

3. Be wall gettering

The ~1000 m^2 of Be wall area has the potential to provide some gettering, although the effectiveness of the Be wall remains to be determined. If the wall is fully saturated with O, the effect may be small. On the other hand, sputtered Be would be a very effective getter.

4. Active Li injection

The potential for injecting Li pellets during the discharge is also being assessed. This would provide additional conditioning during the discharge.

5. RF conditioning

RF helium conditioning can be used between discharges to remove low Z volatile impurities.

2.5.4. Commissioning and Operating Scenarios

A set of candidate plasma commissioning scenarios is outlined in Table 2.5.2.

2.5.5 Vacuum Requirements

A low base pressure between pulses has been a standard requirement for tokamaks to ensure that the leak rate and outgassing rates are acceptably low. Consistent with the experience in present tokamaks, the minimum base vacuum pressure should be no greater than 10^{-5} Pa for hydrogen and 10^{-7} Pa for impurity gases. The maximum leak rate for all of the invessel components should be less than 10^{-7} Pa m^3/s.

2.5.6 Diagnostics Requirements

Plasma and component diagnostics systems are needed to measure the plasma impurity level, the constituents of the pumped gas and the temperature of the plasma facing components. Candidate diagnostics to accomplish these measurements include: residual gas analysis, surface temperature IRTV systems, thermocouples in the PFC's, H_α and UV and visible spectroscopy to measure the impurity levels. Among the most important diagnostics needed is the ability to measure the leak rate and to identify the sources of leaks.

Table 2.5.1 Bake-out and operational experience in present tokamaks:

(Temperatures in ^{o}C)

Alcator C-Mod (B. Lipschultz, J. Irby)
Materials: Mo, SS
No conditioning between shots. ECDC (Electron Cyclotron Discharge Cleaning), 1.5 hours every day before run. B conditioning every several weeks done with ECDC. Li pellet conditionig with Li pellets was tried but not found to be helpful.

bakeout:	130^{o}C for 3-5 days	running:	vessel at room temperature tiles up to 300—500^{o}C	between shots:	room temperature

ASDEX/U (J. Neuhauser)
Materials: C, SS
B conditioning overnight every few weeks to months, He glow discharge cleaning between discharges, 150^{o}C baking (only after vessel opening)

bakeout:	150^{o}C	running:	vessel < 60^{o}C tiles<1000^{o}C tile edges< 3000^{o}C	between shots:	<60^{o}C

DIII-D (G. Jackson, D. Hill)
Materials: C, Inconel
He glow discharge conditioning between shots, active pumping system now used. When pumping is used, density control is possible without He glow discharge conditioning. Boronization is done every 1 to 2 months. A lithium pellet injector has recently been insatlled.

bakeout:	350^{o}C	running:	30—700^{o}C (on tiles)	between shots:	≤50^{o}C

JET (G. Janeschitz, G. Vlases, Gabrielle Saibene and Philip Andrew)
Materials: Be, C, Inconel VV
Be evaporation occasionally, presently use in-vessel cryo-pump in which case GDC and other conditioning techniques between shots are not employed

bakeout	initial bakeout at 200°C for 1.5 days, followed by 40°C cooled tile and 320°C bakeout of the rest of the tokamak. With Mark II divertor, baking of divertor is limited to lower temperatures	running	250—320°C for the vessel walls, but tiles may reach 1000°C	between shots:	≥250°C (divertor tiles cooled down to 40°C between shots)

JT-60/U (M. Shimada/M. Nagami/M. Azumi)
Materials: C, SS
Baking at 300°C is done for 3 days after vacuum vent, normally once or twice a year, high running temperatures help operation, with 300°C, 3 high T_i shots/hour. With 150°C, the number of good high T_i shots drops from 3/hour to 1/hour. Boronization is done once or twice per year. Normally there is 1 hour of He glow discharge cleaning per 6 hours of operation, with 3 to 7 hours additional per night. There are 5 to 10 minutes of helium TDC after disruptions. Lower chemical erosion occurs with lower wall temperatures (less C impurities)

bakeout:	~300°C	running:	100—300°C	between shots:	~200-300°C (40°C for water cooled divertor tiles between shots)

T-10 (K. Razumova, G. Notkin)
Materials: Inconel VV, Graphite limiters
8 hours of baking every night, TDC as well, glow discharge sometimes used before boronization

bakeout:	250-300°C	running:	limiter reaches 250°C	between shots:	20°C

T-15 (K. Razumova, G. Notkin)
Materials: SS VV, Graphite limiters
not much operating experience yet, baking planned up to 150, glow discharge beginning to operate

bakeout:	—	running:	limiter up to 40°C	between shots:	wall at 0°C

TEXTOR (J. Winter)
Materials: carbon limiters, Inconel liner, stainless vacuum vessel
density control with pumped limiter, active pumping keeps the wall inventory low, conditioning with boron and silicon coatings

bakeout:	350°C	running:	150-175°C	between shots:	150°C

TORE-Supra (C. Grisolia)

Materials: SS, Boronized graphite inner walls (~100m^2), actively cooled graphite (~15m^2)
B conditioning, active pumping help control the wall inventory

bakeout:	210°C for a few days, D_2 GD, then He GD	running:	150°C	between shots:	≥150°C

TFTR (M. Ulrickson)
Materials: C, Inconel
B and/or Li conditioning before operation, disruptive discharge cleaning with glow discharge cleaning

bakeout:	150°C	running:	40°C	between shots:	≥40°C

Table 2.5.2. PLASMA COMMISSIONING SCENARIOS

	Plasma parameters	Goals	Systems to be ready
1a. DC glow cleaning Days/weeks of application	gas pressure of 5 10^{-1} to 5 10^{-2} Pa B = 0	Surface cleanup/conditioning	Insertable electrodes, dc supplies (100-300 A total @ ~ 1 kV), gas system (H,He, Ar,....), pumping system, RGA, plasma density diagnostic; impurity spectroscopy
1b. RF discharge cleaning (ECRF or ICRF) Days/weeks of application	1-10 eV/10^{15}/10^{16} m^{-3} ; B = 1-5.7 T (depends on rf frequency)	Localized surface heating, cleanup and conditioning. Pulsed or CW rf options. Possible slow sweep (~hours) of TF to move resonance over in-vessel major radius	rf system and in-vessel antenna(e) or launcher(s); rf coupling monitors; TF (slow sweep); pumping, gas system, RGA, plasma density and spectroscopy monitoring
1c. Discharge cleaning Number of pulses: 10^4-10^5 (low field/current)	B = 0.5-1.0 T; 0.5-2 MA. ac mode @ ~0.1 Hz; pulsed mini-tokamak mode at ~ 0.1 Hz	Surface cleanup/conditioning	TF and PF supplies (can we reconnect PF supplies at higher voltage?), rudimentary plasma equilibrium control, gas, pumping and RGA systems, plasma current and density diagnostics, impurity spectroscopy
2. Limiter configuration Number of pulses: 500	Limiter configuration; circular to moderate elongation Ohmic discharges. Hydrogen plasma I_p = 5 - 10 MA B_T = 4 - 5.7 T n_e = 0.2 - 0.5 10^{20} m^{-3} T = 1 - 2 keV Flat top = 50 - 100 s Operation at low premagnetization. Limited NB power for plasma rotation	- Plasma breakdown and current initiation - Commissioning of plasma control system - Commissioning of plasma diagnostics - Commissioning of plasma fueling and pumping - Limiter commissioning	- PF system, - Fueling and pumping, - Plasma core diagnostic from category 1 - ECH break down system - Error field compensation system

3. Ohmic divertor configuration Number of pulses: 500	Divertor configuration Ohmic discharges Hydrogen plasma I_p = 10 - 15 MA B_T = 4 - 5.7 T $n_e = 0.2 - 0.5\ 10^{20}\ m^{-3}$ T = 1 - 3 keV Flat top = 50 - 100 s Limited NB power for plasma rotation	- Commissioning of plasma control system, - Divertor commissioning - Commissioning of disruption mitigation system - Fueling and pumping control	- Plasma core diagnostic of category 1 and 2 - Divertor/SOL diagnostics - Pellet injector(s) for plasma fueling - "Killer pellet " system
4. Plasma heating commissioning Number of pulses: 1000	Divertor configuration Hydrogen plasma I_p = 10 - 15 MA B_T = 4 - 5.7 T $n_e = 0.2 - 0.5\ 10^{20}\ m^{-3}$ T = 1 - 20 keV $P_{aux} \leq 100$ MW (or as much as available) Flat top = 50 - 200 s	- Commissioning of major plasma heating - Divertor commissioning - Plasma scenario commissioning - Physics basis confirmation studies: - H- mode transition and sustainment - Plasma operational limits (beta limit, density limit) - Heating and current drive efficiency - Divertor power and particle exhaust characteristics and control - Disruption and VDE frequency and characteristics	- Heating systems - Diagnostics
5. Reference regime Number of pulses: 700	Divertor configuration Hydrogen plasma I_p = 15 - 21 MA B_T = 5.7 T $n_e = 0.5 - 1.2\ 10^{20}\ m^{-3}$ T = 10 - 20 keV $P_{aux} \leq 100$ MW Flat top = 100 - 1000 s	- Full development of reference scenario - Completion of physics confirmation studies (with H) - Preparation for DD experiments -Pre-D/DT tests of magnetic and kinetic plasma control; disruption and VDE mitigation	- All systems ready

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Appendix II: Comments/Suggestions From The Major Tokamaks

The physics basis for baking graphite at 350 ° is givenin Appendix I in the ITER Report prepared from the draft titled “Considerations for Bakeout and Conditioning Specifications for In-vessel Components in ITER” prepared by D. Post, ITER JCT, Jan. 20, 1995 Revised May 2, 1995. The final report was reviewed by the contributors from the major Tokamaks. Their comments/suggestions were summarized in Table 2.5.1[Appendix I]. Their complete interesting verbatium comments/suggestions are given below.

Appendix: Comments/suggestions from the major tokamaks:

This section is a verbatium compendium of the comments and suggestions of the scientists we asked to review the draft requirements. They represent almost all of the major tokamaks and a number of surface physics groups. Their recommendations form much of the basis of the draft requirements for bakeout and conditioning. They were each given an opportunity to review and modify their comments before I attached them to this document.

J. Winter, TEXTOR

I would like to make a few comments to your draft set of baking temperature: I agree with the draft set of baking temperatures, which are 350ºC for all major wall components. An added argument for a high baking and operation temperature is that carbon does not retain He beyond about 300ºC which might have been implanted into the components during the discharge. Furthermore I think that it will be indispensable for ITER to have the flexibility of a wall which can be baked hot. This is important for most of the advanced conditioning techniques and for Be. The diffusion of Be through BeO yielding fresh surface layers for gettering is known to occur at elevated temperatures. The effect of medium high temperatures (350ºC) plus particle impact still has to be measured by somebody, but a high temperature certainly helps! Disruptive discharge cleaning as in TFTR which can only go up to 150ºC is something we don't want in ITER.

TEXTOR bakes the wall to 350ºC before operation, the standard operation temperature is 150ºC between shots, limiter tiles go up to 750ºC (average). Density control and recycling control is no problem because of the pumped toroidal limiter ALT II. ALT II also controls

the wall inventory! This is a big effect. The same experience exists at Tore Supra. Thus discharge inherent scenarios (D/He changeover) may do the job of recycling control and help the T scenario in ITER. Also here hot walls help. As far as the between shot and "running" temperatures are concerned, the temperatures are rather determined by the cooling and materials properties.

J. Neuhauser, ASDEX/U

ASDEX-Upgrade data and the ASDEX-U experience:

- bake-out: 150^{o}C (only once after machine opening)
- boronization every few weeks to months
- running: SS vessel near ambient temperature; target tile temperature during discharge up to 1000^{o}C, tile edges up to 3000^{o}C (dependent on divertor power load); between discharges target plates and vessel cooled towards ambient temperature
- He glow discharge conditioning between shots

After vessel opening bake-out at 150^{o}C - supported by boronization - is obviously sufficient to get excellent plasmas immediately (as was true at old ASDEX). Boronization alone after vessel venting does not allow reasonable operation, i.e. baking (at 150 degree) is essential. Baking without boronization gives a lower density limit, i. e. dirty plasma.

Gary Jackson, DIII-D

The specification of a bakeout temperature of 350ºC for ITER is reasonable. A high temperature bake has certainly been one of the conditioning techniques which has helped DIII-D immensely. Listed below are some general comments, and some corrections on DIII-D wall temperatures and baking procedures.

1. It is true that graphite absorbs large quantities of water (15-T-l/g as you quoted). What is not so clear is how long it takes to remove this amount of water. In DIII-D, at 350ºC, the decay constant for water measured by the RGA is very long, probably days; we don't bake that long so we can't accurately measure it. The reason we can successfully operate is that a short (typically 8-16 hr. bake) removes water from the surface and near surface layers. Once we cool to our operating temperature of < 50º C, diffusion from the bulk is small.

Although I don't know all the details, ITER certainly has a very different operating scenario from DIII-D. The advantage is that you can bake for a much longer time. However, the operating temperatures will be significantly higher. I would be a little concerned that bulk diffusion of water might play a role in your long pulse shots. Perhaps we need more lab data on diffusion of water out of bulk graphite at 350C? Ideally, you should bake the graphite to a higher temperature than you expect to operate at, but I don't think this is an option on ITER.

One thing that we have had success with on DIII-D was a careful pre-conditioning of the graphite tiles before installation. We baked them to 800-1000ºC, bagged them in an argon atmosphere, then installed them with an air exposure of 2-6 weeks (this has been documented in the literature if you need references). This seemed to significantly reduce outgassing but on a large tokamak such as DIII-D it is difficult to do a controlled experiment, especially during machine pump down and commissioning. You might consider a similar procedure for ITER, although the air exposure times of graphite before pumpdown would obviously be considerably longer.

2. Although not specifically called out in your preliminary draft of temperature specifications, I think that in addition to baking some sort of conditioning such as low power pulse discharge cleaning (TDC) will be needed to remove surface oxides. You don't want to remove just water and have oxygen still on plasma facing surfaces.

3. You noted that all surfaces should be maintained at 350ºC, not just graphite, Be, etc. I think this point should be emphasized. There

will be pressure, I'm sure, to relax this specification in port areas, etc. If these areas aren't baked, this will be a nice source for water influx during ITER operation. [note: In further discussions, G. Jackson agreed that simultaneous bakeout with graphite at 350°C and steel and metals at 200°C would likely be adequate].

4. Some corrections on the DIII-D experience in your draft proposal:

a. Our typical bakeout temperature is 350°C average (inner wall about 400°C, outer wall about 300C). We have gone as high as 400°C average when the machine required it. I'm not sure what number you wish to report, typical bake, or highest temperature ever used.

b. For tokamak operations ("running" in your table), surface temperatures during operation have gone as high as 700°C. Disruptions can be somewhat higher, which is perhaps where you got the 1000°C number.

c. We DO NOT bake each night. Baking is only done when the machine is notably 'dirty' such as after a vent or large air leak. We also bake the machine in conjunction with boronization. On rare occasions we will bake to reduce recycling the next day.

d. Between shots, the bulk temperature is typically less than 50°C. (The Inconel backing plates are water cooled).

Notes from M. Shimada and N. Nagami summarizing the experience on JT-60/U (through discussions with Y. Shimomura):

Baking conditions	300° C	3 days after vacuum vent
		normally 1~2 times/year
Running normal 300° C		(vacuum vessel)
Experimental range 100 – 300° C		
	150° C	only one shot/1 hour for good high T_i mode
		(normally 3 shots at 300° C) (lower recycling and lower impurity)
Between shots		
normal	300° C	
normal	200° C	for gas cooled divertor tiles
(only after a few shots	40° C	for water cooled divertor tiles)
Experimental range	100° C – 300° C	

Others

B coating >1000 A° once/year(1994) 2 times/year(1993)

Ports and other extensions: 100 – 150° C

1 hour helium glow/ 6 hour operation

2 hour helium glow during the night

5 – 10 min. helium TDC after disruption

τ_p*=1 sec after change limiter to divertor

After a few seconds, the divertor plate is saturated.

K. Wilson, Sandia

I strongly support the bakeout requirements that are set forth in the "Preliminary Draft of Temperature Specifications for In-vessel Components in ITER for the Purpose of Initiating Discussion"[*which called for 350°C bakeout for all components, D. Post*]. A bakeout capability of 350 °C is required if graphite is planned to be used in any significant application in the machine. This temperature is also required if porous beryllium (85% dense), which might be typical for in situ plasma spray repair of the FW coating, is used in the machine. Your table of bakeout and running experience in operating tokamaks says it all. If you bake at 350°C, then you have rapid recovery in carbon based machines from disruptions, air exposures, or even water leaks. If you bake at 150°C (like TFTR or ASDEX-U), then you are forced to use chemical means such as boronization to remove the oxygen-bearing species. It is not apparent that such techniques will work in ITER with its 1000s pulses. It should be noted that beryllium can serve the same purpose as boronization for an oxygen getter to clean up graphite outgassing. Hence you might luck out by having both carbon and beryllium in the machine. More investigation should be made of the JET experience during their mixed beryllium-graphite operation phase to see if beryllium can lower the bakeout requirements of a machine with graphite plasma interactive components.

Regarding co-deposition of tritium-carbon films, I agree that the thicker films will be far harder to remove between shots with present techniques, which just "scratch the surface". Our measurements in TFTR, DIII-D etc. support an erosion rate of on the order of 30 Angstrom per second in high flux regions, and the formation of codeposited films at about the same rate in low flux areas. So the numbers you quote of 100's to 1000's of Angstroms thick are per shot, since codeposition is continuous. We found films of 50-100 microns thickness in TFTR after typical run campaigns (which were the equivalent of only a few ITER discharge).

Regarding bakeout to remove implanted tritium I agree that you can't bake the tritium out of bulk graphite at any reasonable temperature. For metals, you have to understand the mechanisms for tritium retention in a metal like beryllium or tungsten. Tritium can be trapped in the saturated layer (to many percent over the range of the incident ions and neutrals), trapped in the bulk neutron damage,

or migrating freely in the bulk metal lattice (looking for a free surface to get out or a neutron damage trap to bind with). Baking to 350°C will decompose the saturated layer in beryllium and probably tungsten, and bakeout with a deuterium glow discharge will help to deplete the mobile tritium. However, the trapped tritium in the radiation damage is strongly bound up in beryllium and tungsten, so baking even at 350°C probably won't help deplete this inventory significantly. This is a topic of active research and modelling by the JCT/Home Teams. Federici (Garching JCT) has been working closely with Rion Causey (SNL) and others to calculate the techniques for tritium inventory control for various materials selection for ITER.

You can keep the temperature between shots low if you do other conditioning techniques such as GDC. However, between shot conditioning like GDC is a problem due to the magnetic field. Cowgill at SNL has been working on an ICR technique to get around the magnetic field problem.

J. Wesley, Early DIII and DIII-D operating experience:

My direct experience was with DIII and early DIII-D operation:
1) Initial operation with metal FW (Inconel 625) and small quantities of graphite with ~250°C bakeout and TDC (Taylor discharge cleaning) gave OK plasma results (including H-mode, but VH modes, etc. weren't known yet)
2) Switching to larger amounts of graphite required escalation of baking temperature to 350-450°C (inner wall of vessel is hotter than outer wall) to get acceptable results. Glow cleaning began to replace TDC and when optimized, was more effective.

After I left, my understanding is that He glow cleaning between pulses, boronization AND nightly bakeout (? with TDC) gives good plasmas, good disruption recovery and better particle control for H- and VH-mode.

My anecdotal information is that tokamaks with graphite without high-temp bakeout (e.g., TFTR) can get the same 'good' performance, albeit with heroic conditioning efforts and not as good disruption recovery. Clearly they would (bake to 350+) if they could.....

In all present cases, $\tau_{cond} >> \tau_{pulse}$(per shot or per shift). ITER will have $\tau_{pulse} >> \tau_{cond}$, especially when we can't turn off TF field between shots. Also present He conditioning may saturate in < 1 ITER pulse, so the effects of anything other than 'deep conditioning' may be irrelevant. I don't claim to be an expert, but < 200°C bakeout plus significant quantities of graphite + no-between-pulse conditioning seem to me to be a invitation for a major failure vis-a-vis getting sustained burn.

Experts know much more of the history and details. But in receiving their input, we need to keep ITER's ~50% duty cycle, 1000+ s pulse and continuous TF-on operations requirement always in mind. Also, it sounds like getting set up to bake will take a major operations action.

K.A.Razumova and Gena Notkin: T-10

The T-10 chamber is baked up to 250-300ºC for 7- 8 hours very night together with Taylor-type discharges in turn during experimental run periods . Glow discharge cleaning is only used before boronization, or in other special cases. The T10 first wall material is Inconel and graphite limiters are used. The total amount of graphite amount is not large. The T-10 wall temperature is about 20 ºC; The T-10 limiter temperature changes from 80 ºC (in case of high density shots) to 250 ºC (after several low density shots). The influence of disruptions on the next shot parameters is negligible if gas puffing feed back system is adjusted properly. We have no experience with ECR discharge cleaning. It's too expensive from one hand and gyrotron pulse (<500 ms) is too short from the other hand.

Victor Vdovin had ideas of RF waves using for T-15 cleaning, but we have no experience as of now.

Kingston Owens: TFTR

Comments on the ITER bake-out-specifications.

1) Retention and recycling from the co-deposited material is an issue. In principal, once H/D/T is co-deposited, it is no longer available for plasma fueling. However, ITER survives because of redeposition (aka co-deposition). How the redeposited layer behaves when codeposited with H/D/T is at best poorly understood. Active pumping for density control is probably the only reliable technique for long pulse operation. In other words, bake-out has little effect on the co/re-deposited layer. On the other hand, the JET model for tritium retention which matches their measurements relies on diffusion in the bulk and surface recombination, both of which are very temperature dependent. One should check P. Andrews Nuclear Fusion article (1993, p. 1396)on the JET experience of removing tritium following their PTE experiment. Hence the operating temperature may have a strong influence on inventory.
2) In several places you mention 'glow discharge cleaning with H'. Do you really mean with He or 50:50 D/T? You really want to remove H to avoid fuel dilution. [*Yes, D. Post*]
3) The use of a getter in the vessel will reduce clean-up time by increasing the effective pumping speed for impurities. Clean-up, as I see it, is removing dirt from the place where you don't want it to someplace where you don't care. If pure pumping is used, it is slow because water and oxygen tend to stick to the vessel walls rather than move down the pump duct, ie, you are pumping speed limited. Putting a getter in the vessel removes this problem. The junk is still in the vessel though so you better be sure it is tightly bound in a 'safe' place.
4) Finally, I view these cleaning techniques as a way of getting reasonable plasmas. One of the missions of ITER is learning how to make quasi-steady-state discharges and how to handle particle and impurity control under reactor-like conditions. It is doubtful if gettering, GDC, or bake-out will help much in this regard.

Dennis Mueller: TFTR

I have three concerns that I am sure others have raised, but in the event they have not:
1) Oxygen outgassing can indeed be controlled (within some limits) by operational machinations (discharge cleaning of various

forms and gettering), but this comes, at least on TFTR, at a huge cost in terms of time and effort. I am particularly concerned that gettering will only temporarily hide O which will later become available to the plasma. For ITER and long pulses hidden O could be re-released during the long pulses. Basically removal is better than burial. So I would encourage in situ bakeout of the graphite components to remove water and oxygen. This could be done with some local heating of the graphite, but thermo-mechanical stresses must be investigated if a local heating method is chosen. Furthermore, for local heating, the possibility of not removing O, but merely chasing it from one component to another is a concern.

2) Tritium retention on ITER will be dominated by codeposition, especially if Carbon is in intimate contact with the plasma. Unless the regions of codeposition are frequently(constantly) depleted of Hydrogenic species, the tritium inventory in wall/diverter components could grow unacceptably large. Baking to 1000^{o}C is probably not possible, so some mechanism may be needed to drive H/D/T from graphite between shots. Use of Be for gettering of O will also provide H/D/T gettering and could also cause tritium inventory problems unless some release mechanism is provided.

3) Transient operational excursions, whether full blown disruptions, minor disruptions, or transient loss of confinement could cause surface temperature excursions up to many 100^{o}C. Such events would outgas near surface regions and make either O or excessive H/D/T available to the plasma. It is such excursions that make removal preferable to burial, since much operational time will be spent in recovery if such outgassing occurs.

For the above reasons, ITER should be bakeable to 300^{o}C or higher. Also, some consideration should be given to provide capability of heating at least the surface of graphite components to 1000^{o}C.

Don Cowgill: Sandia

In general the summary looks good.

One addition, however, concerns the bakeout requirements for a mixed Be/graphite system. If one could guarantee (e.g. by design) that all of the graphite gets heated to >350^{o}C during plasma shots, then elevated temperature bakeout for the graphite components may not be necessary. Water will be rapidly outgassed during the long pulse operation. [*This is probably true if the burn is not*

interrupted by a disruption due to the outgassing water/oxygen, D. Post]

Also, tritium retention in the graphite can be reduced significantly by designing the thermal link between these components and the actively-cooled plates so that the graphite is maintained at T>350ºC during shots. Since these components are intended to be high-flux components, such a design may not be too difficult.

To prevent C-T codeposition, it is also necessary to maintain adjacent (carbon deposition) areas at this temperature. One possibility is to cover exposed areas with a liner appropriately heat-linked to the plasma heated tiles.

Thus it appears to me that, with careful thermal design, graphite components could be included in ITER without increasing the requirements on bakeout and conditioning.

G. Vlases, Gabrielle Saibene and Philip Andrew: JET

The present bakeout procedure in JET is:

1. All water is removed from the machine cooling circuits (freon remains in the divertor coils), and the machine is baked to 200 ºC. This includes the divertor tiles, because their support structure is water cooled. The baking phase lasts about a day and a half.

2. The machine is cooled to 100 ºC and the water is restored to the cooling circuits.

3. The machine is baked at 320 ºC by circulating hot gas between the two walls of the vacuum vessel; the ports are baked by electrical heating tapes. During this time the divertor tile temperatures remain very close to the 40 ºC which is maintained in the water cooled rails upon which they are mounted. Conditioning by GDC and plasma pulses is carried out during this period.

4. The wall temperature is normally lowered to 250-275 ºC during plasma operation. During the course of a day of operation, the divertor tile temperature (bulk, measured by thermocouples) cycles between 40 ºC and a maximum of 350 ºC, which is the allowed maximum temperature, achieved only in high energy pulses. The cooling time constant for the tiles is approximately 36 min.

Measurements at JET indicate the presence of a "second reservoir" for water, CO, etc, which is released only at temperatures of about 320 ºC; i.e. prolonged baking at 200 ºC won't get rid of it. Gabrielle feels quite strongly that any graphite plasma facing surfaces in ITER must be bakeable to 350 ºC unless they are conditioned by direct plasma-wall interaction (divertor strike zones, limiters, etc.)

To summarize, we would bake the divertor tiles to 320 ºC if we could, but we can't. However, they get conditioned fairly rapidly by plasma discharges. On the other hand, the other carbon components in the vessel have to be baked to this temperature in order to remove the oxygen. Earlier in the campaign, when the vessel walls were only baked to 250 ºC, there was more residual water vapor and the restart after a vent was more difficult.

TORE-Supra: C. Grisolia (3/31/95)

The Tore Supra results are presented hereafter and complement the results obtained from other machines.

1) Materials: SS covered by actively cooled graphite tiles (more than 15 m^2). All the inner walls are coated by means of boron doped graphite (boronisation) (coated surface = 100m^2).

2) Bakeout:

- electrically up to 300°C. This way is used only when the water has been removed from the water cooling loops.
- by mean of a water loop. Up to 230°C at 40 bars. Used routinely to fix the first wall temperature of Tore Supra.(routine use at 150-170°C)

3**) Glow discharges (GD) conditioning:**

- 2 reciprocating electrodes installed with RF available for the breakdown. The electrodes are placed in the center of the torus for glow operation.
- operating pressure = 0,3 Pa, glow current = 5 A (up to 10 A recently installed), 1000V (D_2) to 1500V (He) at the breakdown, 300-400V during GD.

4) procedure:

4-a) After a shut-down:

- 4 days at 210°C needed before D_2 glow discharges (pumping capabilities installed in Tore Supra: 10 m^3/s in D_2 in the Torus). When H_2O pressure is less than 1% of GD working pressure, GD are turned on (T is kept constant). This is done to minimise oxygen redeposition.
- 3 to 4 days of D_2 GD needed to remove oxygen from the graphite vessel walls. Wall "clean" when CO production less than 3 10^{16} mol./s/A (200°C).
- 2 to 3 days of He GD to detrap hydrogenic species. Walls desaturated when deuterium production is less than 10^{17} mol/s/a (200°C).

-if this timing is followed, only 3 to 4 tries are necessary to get the first He pulse, 1/2 day for a good He plasma (1.5 MA, 8s current plateau, 1V loop voltage), 2 days for additional heating. The key for

success is : NO disruptions during this phase. So plasma current is increased slowly.

- after one to two weeks of operation, boronisation is done.

4-b) maintenance:

- almost every night: He GD (10 to 12 hours at 170°C).
- helps to restart the first next shot
- criterion of D_2 production recovered 2 to 3 hours if no disruption at the end of the experimental day (5 to 8 hours if disruption)
- problem : He walls trapping due to high electrode-vessel potential (He seen in the first D_2 shots)

4-c) Boronisation:

- working gas: $He+B_2D_6$ (15%) and T_{wall} = 150°C
- long life coating (2 months # 800 shots)
- results :
    * residual gas improvements.
    * plasma oxygen content improvements
    * better particles after shot recovery
    * easier restart after a disruption.

4-d) walls desaturation:

Tore Supra has a permanent toroïdal magnetic field and preliminary results show that GD do not work with permanent field. So, new technique has been developped at Tore Supra using the pumping capabilities of the outboard pumped limiter to control the wall particle inventory (see Loarer, Chatelier, Miodusewski et al, 20th EPS Lisbonne, 1993)

5) Wall temperature problem:

- in Tore Supra, during almost all the operation time, wall temperature is high : greater than 150°C. During the night, when HeGD are done, T_{wall}=170°C.
- After a water leak: 10 days needed to recover i) good vessel vacuum spectrum and ii) good level of oxygen in shot.
- Boron is used (on the oxygen point of view) as a getter of oxygen to avoid water recirculation due to temperature gradient in the vessel walls.
- in Tore Supra, even at high power density (0.5 MW/M^3), the oxygen level during shot stays at low level.

Conclusion : oxygen is not a problem during Tore Supra operation.

- 200°C is better than room temperature and than 100°C:
  - better particle recovery during shot
  - faster conditioning procedures

- higher temperatures are not neccessary except (maybe) at the beginning of a restart just after a shut down to decrease the time to recover a good vacuum. In this case it would be possible to use electrical bakeout before to put water in the pressurized water loop.
- key for a good conditioning procedure : avoid the temperature gradient in the vessel to suppress water recirculation.

“Dr. Grisolia also proposes: ‘I think that it would be very interesting to create an E-mail address where all the people working on conditioning could exchange information on procedures, techniques etc... A data base could be easily added. Do you know if this exists? If not I propose to create this E-mail where people can write and read all these informations.’

We are considering this as part of the ITER Expert Group Program. “ —D. Post: