

Plasma Facing Material by Self-interstitial Solid Solution Strengthening –Problem, Proposition, and Solutions

Muhammad Musaddique Ali Rafique^{1,2,3}

¹Department of Fusion Engineering, Massachusetts Institute of Technology, Cambridge, MA 02139

²National Aeronautics and Space Administration, 39B, Florida

³Eastern Engineering Solutions Inc., Cambridge, MA 02139

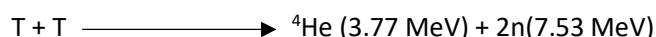
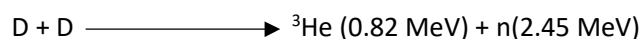
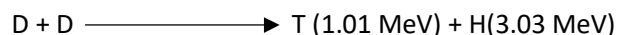
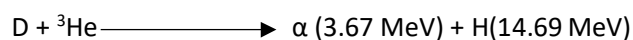
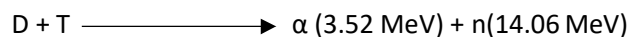
Abstract

Bubble (point defect) – a precursor of fuzz or under dense nanostructure formation is crystal lattice defect. Suitable selection of crystal lattice which inhibit Frenkel pair generation and intrinsically promotes self-interstitial solid solution strengthening contributes effectively towards making plasma facing material. For this, interstitial sites, their size, amount / fraction, positions, tendency of occupation and diffusion parameters (e.g. activation energies (Q), activation volumes) are determined. Fcc iron carbon alloys (austenitic stainless steels AISI / SAE 321, fcc structure, Pearson code $cF4$, space group $Fm\bar{3}n$) are proposed as suitable candidates. Along with their room temperature fcc structure having 12 interstitial positions (4 octahedral, 6 coordination sites and 8 tetrahedral, 4 coordination sites / unit cell) to allow insertion of self (iron) atoms, they have excellent corrosion resistance, thermal conductivity, and non-magnetic properties. After their melting, casting, and machining to required dimensions and geometry, stabilizing heat treatment is applied to precipitate all carbon as TiC and prevent formation of $Cr_{23}C_6$ (sensitization). This resist heat and surface degradation and yield excellent architecture which not only inhibit Frankel pair generation but will also allow bulk assimilation or surface annihilation (loop punching) of this lattice point defect. A superior thermal, fluid, and structural design augment above. A second choice is presented as Co base superalloys owing to same fcc crystal structure and excellent properties (such as strength, dimensional stability, oxidation resistance) at high temperature.

Keywords: point defect; crystal lattice; interstitial; austenitic; stabilization; superalloys

1. Introduction and background

Harnessing terra watts of energy available during nuclear fusion is an avenue of great interest and potential benefit to overcome tomorrow's energy requirements of society. In general, following reactions occur during fusion [1]



Of these, reactions 1 is most feasible. Its cross section (σ) is around 70 KeV but high D – T reactivity is reached already at 20 KeV. Various methods based on design [2-39], manufacturing and control have been proposed to harness this energy. However, mostly they are limited by availability of suitable material directly facing and interacting plasma. Ideal material is required to possess superior physical (high melting point, high temperature stability, high thermal conductivity [40], high corrosion resistance (both at room and elevated temperature)), design [41, 42], mechanical (high strength, high hardness, high thermal shock resistance, high creep resistance, high thermo-cyclic creep, high cyclic plasticity [43], high thermal fatigue resistance, fracture resistance [44]), structural (excellent shear stress, shear strain, bending moment, torsion, dimensional stability at elevated temperatures [45]), manufacturing (easy formability (e.g. rolling, bending, cutting, cladding, weldability [46])) and nuclear (low sputtering yield, low tritium retention) properties.

To this end, various materials (such as W [47-50], addition of Rh in W [51], use of bcc Fe [52, 53], Ta [54], W-Ta [55], Ta/Fe [56], Pd [57], nanocrystalline Cu [58], SiOC/Crystalline Fe nanocomposite [59], W-K [60], reduced activation steel [61], ferritic [62], ferritic/martensitic steels [63], Be pebbles [64-67], Be and beryllides [68], graphite, carbon fiber composite [69]) and high Z atoms (Zr, Nb, Mo, Hf, Ta) [70] have been tested but none proved satisfactory [71-74]. All show rapid surface degradation exhibiting surface blisters [75-78] and formation of fuzz [51, 79-82] or under dense nanostructure [40] after bubble. These rapidly degrade their physical (e.g. ability to contain plasma by inverse or re-radiation due to excellent surface finish), mechanical and aforementioned properties. It is hypothesized that production of surface nanostructure (nanochannels) [83-86] will help escape of He [60, 87-91] inhibiting formation (nucleation and growth (crystallography) [90] or precipitation and coalescence (physical chemistry)) of bubble and restricts surface degradation. However, this is poorly understood and wrongfully approached. It has been approached by physical chemistry, surface engineering, chemical engineering, process engineering, nuclear engineering, and nuclear physics point of view but none made use of fundamental physical metallurgy principles.

2. Bubble

Bubble is described in many ways. Historically, it has been described in chemical engineering sense as most of activities of fusion research started on east coast in US in Princeton Plasma Physics Laboratory where physicists and chemical engineers were handling experiments. In **physical chemistry** sense, He atoms have strong repulsion to W atoms [80, 92]. This ultra-low solubility forces He atoms to self-

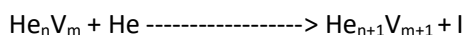
precipitate into small He bubbles [83] that become nucleation sites [90] for further void growth [93] under radiation induced vacancy supersaturations [94], resulting in material swelling [69, 86, 95] and high temperature He embrittlement [71, 96, 97], as well as surface blistering [75-78] under low energy and high flux He bombardment [54, 98] at elevated temperatures [99]. This may be mitigated by engineering structures in material which help in outgassing of He. These may be nanochannels, nanocavities and / or vasculated solids. These are purposefully built in material and their type, size, amount, density, configuration, geometry, orientation, and mode of placement heavily effect the degree and control they exercise on final properties (resistance to damage). Other structures such as Helium cavities are also observed which may be suppressed by defect sinks, nano clusters, or interfaces. In **physical metallurgy** terms, it is a point defect [100, 101] in a crystal lattice – a precursor of fuzz [56, 80, 102] or under dense nanostructure [40] formation. It is produced as a result of self-ejection of W from interstitial positions in its own lattice by He atom or cluster of atoms (usually 9 atoms at 700 K [40]) (Cluster: 7 – 8 atoms (bulk), < 7 – 8 atoms (surface) and its bonding with vacancy [88, 103-106] forming a Frenkel pair [107-111]. This is more prominently observed near the surface. In this layer, bubble density and diameter depend on surface temperature and helium flux / fluence [112].

$$\begin{aligned} \text{Number density of bubble } N_d &= f\left(t_s, \frac{df}{dx}\right) \\ \text{Bubble diameter } d &= f\left(t_s, \frac{df}{dx}\right) \end{aligned} \quad (1)$$

where t_s = surface temperature, $\frac{df}{dx}$ = Helium flux / fluence

Note: This is a plastic deformation in a sense that point defect is formed. However, it occurs athermally (at thermal equilibrium). It is diffusional process and compositional adjustment. This is physico-chemical adjustment. However, later part does not dominate.

This mechanism is tested and proven in conventional nuclear engineering (fission) applications where α and β particles, γ rays and neutron flux strike the material. Plasma does not constitute of aforementioned and thus it is not very applicable mechanism. Selection of suitable crystal lattice which inhibit Frenkel pair [111] generation and promotes host atom self-interstitial solid solution strengthening contributes effectively towards making plasma facing material – subject area of present study. More detailed mechanisms involves loop punching after spontaneous organization of W self-interstitials (SIA) into a prismatic <111> - dislocation loop, ligament thinning [80], crater and nano tendril formation and trap mutation reactions [80]



where V = W vacancy, I = self-interstitial atom. He_3V_1 is a stable cluster. The absence of large He_nV_m clusters suppresses fuzz formation [40]. 95.9% He is retained in He_3V_1 clusters. Maximum temperature observed so far [40] at which He retention = zero is 2500 K. In this case, unretained / emitted He cannot be trapped but move until they reach the surface and desorb or migrate to bulk.

Various morphologies are also observed such as nano tendrils [91, 113-115], nanochannels [83], large scale nanostructures [116], and structure in equiaxed nanocrystalline tungsten [117]. Role of impurities such as O is also observed and discussed [118]. Various production methods such as powder metallurgy [119] and other routes [120] also play an important part. Extensive simulation studies (such as Molecular dynamics [89], *ab-initio* [121] have also been carried out which helps. Fuzz itself is a surface phenomenon

and is only observed in plasmas containing Helium including mixed He-H plasmas but never in pure H or Ne plasmas. It is a strong function of temperature and incident ion energy. In W, it is observed only in temperature range of 900 – 2000 K and above threshold incident ion energy of approximately 20 eV and fuzz layer thickness depicts characteristic one – dimensional diffusional growth that scales with square root of fluence. At an ion energy of 10 keV and higher but below 20 KeV, a more complex 'coral-like' structure is formed which is not 'fuzz'. This also is undesirable as it deteriorates the properties at much lower incident ion energy than proposed as threshold value. At $T < 900$ K surfaces without any specific or pronounced damage are observed. They appear to be decorated with He bubbles but it is just depiction as no lattice point defect starts at this temperature. At $T > 2000$ K only micron-size holes on the surface are formed in the irradiated tungsten which is not fuzz but again is a damage and is undesirable. Materials specially to be used in first wall [12, 17, 28, 85, 119, 122-133], blanket [134-143] and diverter [10, 24, 28, 36, 125, 144-150] of ITER reactor [10, 16, 17, 24, 25, 85, 129, 136, 145, 147, 151-165] and experimental DEMO [3, 4, 9, 13, 19, 31, 32, 36, 122, 123, 133, 163, 166-173] fusion device are important.

3. Self-interstitial solid solution strengthening – crystallography of plasma facing materials

Refractory materials such as W have gained popularity to serve as potential plasma facing material [150]. It has high melting point, low activation, good thermo-mechanical properties, good weldability, good corrosion resistance, low sputter erosion/redeposition, low tritium retention/co-deposition and aforementioned nuclear properties. However, upon exposure to plasma, it shows rapid surface degradation exhibiting surface blisters [75-78] and formation of under dense nanostructure – fuzz as described above. It is supposed to happen as W has bcc structure with 12 small tetrahedral interstitial sites and 6 large octahedral interstitial sites [100, 174] with lower atomic packing factor and coordination number than fcc (bcc 8, fcc 12) [175], small cation size and a higher tendency for external interstitial atom to occupy 12 small tetrahedral sites [175, 176], poor tendency for host / self (W) atom to occupy 6 large octahedral sites, thus poor tendency for self-interstitial solid solution to form and strengthening to occur. Lower atomic packing factor (bcc metals) means atoms are less densely packed in a unit cell and it is easy to eject them from their own lattice e.g. W metal. In these types of metals (e.g. bcc α Fe, W), it is easy to insert external (impurity) atom (e.g. C, N, H) in lattice provided it is of right size, have tendency of preferentially occupy certain sites (12 small tetrahedral sites) and low enough activation energy. Comparative tendency of strengthening by self-interstitial solid solution strengthening is lower and tendency of strengthening by external atom interstitial solid solution strengthening is higher (provided atom is of right size, shape, and valency). (Note: In general, self-diffusion in bcc metals is higher than that in fcc metals at same homologous temperature [177]). Activation energy for self-interstitial formation is approximately 2 – 5 x larger than for vacancy. This energy can be reduced by so called dumbbell configuration [177-181] in which two atoms share one lattice point; the lattice point is usually at their common center of mass. However, this is independent of the tendency of self-atom (Fe) to occupy interstitial positions and form solid solution).

For interstitial

$$D^i = \frac{a^2 v_0 \rho}{6} \exp\left(-\frac{\Delta g_m^i}{k_B T}\right) = D^i \exp\left(-\frac{E_d^i}{k_B T}\right)$$

where ρ = number of neighboring interstices and $E_d^i = \Delta h_m^i$ (Gibbs free energy pf formation)

This may also be represented as [180],

$$C_l^{eq} = g_l \exp\left(-\frac{G_l^F}{k_B T}\right) = g_l \exp\left(-\frac{S_l^F}{k_B}\right) \exp\left(-\frac{H_l^F}{k_B T}\right)$$

G_l^F = Gibbs free energy of formation, S_l^F and H_l^F the corresponding formation entropy and enthalpy and g_l = geometric factor (fcc = 3)

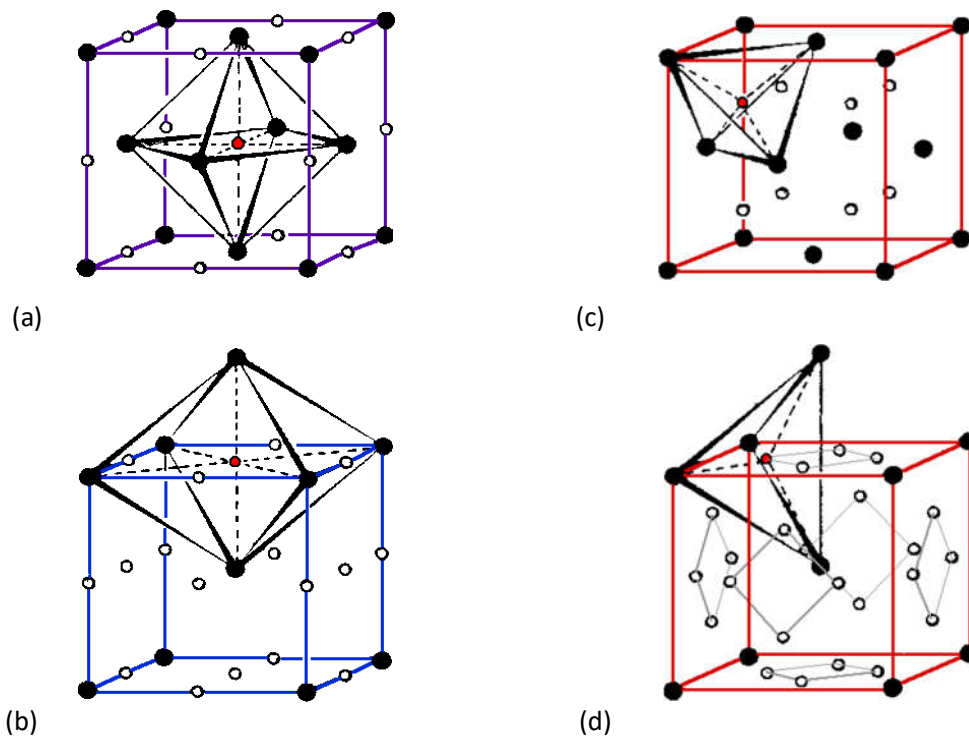


Figure – 1: (a) Octahedral sites in an fcc lattice. $12 / 4 + 1 = 4$ positions per unit cell. (b) Octahedral sites in bcc lattice. $12/4 + 6/2 = 3 + 3 = 6$ positions per unit cell. (c) Tetrahedral positions in fcc lattice. 8 positions. (d) Tetrahedral configuration in bcc lattice. $(6.4) / 2 = 12$ positions per unit cell [182, 183].

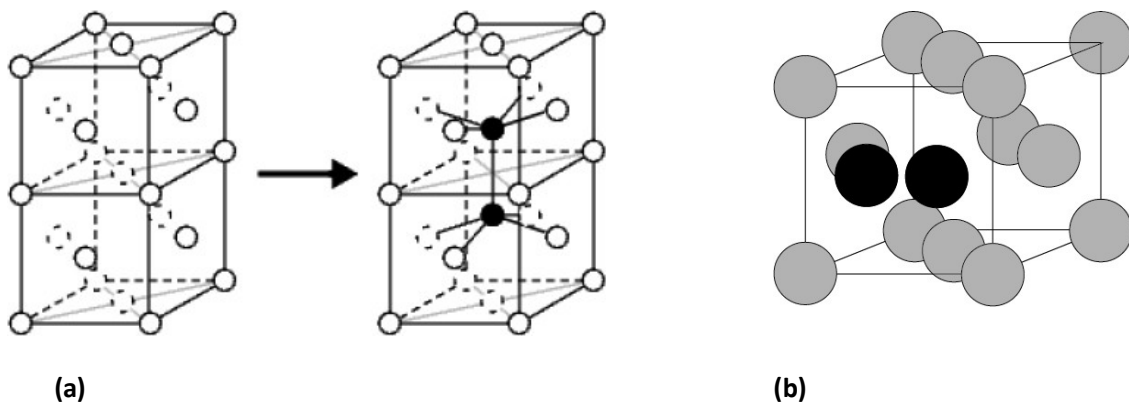


Figure – 2: (a) FCC 100 dumbbell, (b) Dumbbell configuration of self-interstitial in an fcc lattice [180].

Further, W has high radius (139 pm) than iron (126 pm) which also restricts formation of interstitial solid solution. It also has very high stacking fault energy at room temperature which requires a lot of force before considerable plastic deformation. Plastic deformation happens along certain planes of highest possible planar density [184]. Various complimentary surface engineering techniques such as physical vapor deposition is applied to counter this. These form surface architecture (nanochannels) and vasculated solids which help escape of He hence hinder degradation. However, this is additional expensive step without understanding underlying phenomena. In essence, bubble is not a physical feature and any generation of nanochannels will not help their escape. Fcc iron carbon alloys (austenitic stainless steels AISI / SAE 321) are proposed as suitable candidate. Along with their fcc structure, they have excellent corrosion resistance, high thermal conductivity, and non-magnetic properties. Careful control of their bulk and surface microstructure by alloying during melting and casting and followed by heat treatment is proposed to yield excellent architecture which not only inhibit Frankel pair generation but will also provide paths (suitable sites) for movement of any of this point defect in lattice, its bulk assimilation and surface annihilation. Historically, it is also a material of choice [185, 186].

4. Size of interstitial sites in a lattice - Largest interstitial void radius in fcc lattice

Interstitial solid solutions are those in which the solute atoms occupy the interstitial positions (holes between the atoms) in the crystal lattice of the solute. Interstitial solid solutions always have limited solubility of the solute. One of the requirements of measurable solubility is that solute atoms must be small enough to fit into the interstitial positions of solvent. Electronegativity differences are also important. For example, carbon shows measurable interstitial solubility in Iron while O_2 and F_2 do not. Even though atoms are smaller than C. At above lower critical line in Fe-C diagram, Fe can adopt a face centered cubic (fcc) structure. Radius of the largest interstitial void in this lattice at $(\frac{1}{2}, 0, 0)$, $(0, \frac{1}{2}, 0)$, $(0, 0, \frac{1}{2})$ type positions may be calculated as. (The atomic radius of Fe = 0.129 nm).

Let the radius of an Fe atom be R and that of the interstitial atom at the position $(0, \frac{1}{2}, 0)$ be r . Then from the figure 3 we see,

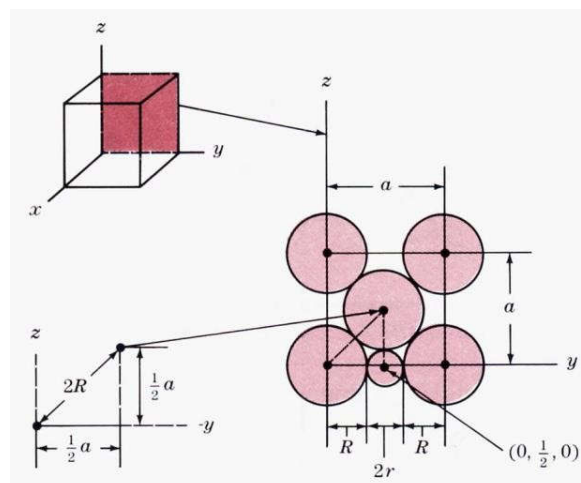


Figure – 3: Schematic of interstitial void in fcc lattice

$$2R + 2r = a$$

In addition, from geometry we see that

$$\begin{aligned}(2R)^2 &= \left(\frac{1}{2}a\right)^2 + \\ \left(\frac{1}{2}a\right)^2 & 4R^2 = \frac{1}{4}a^2 \\ & + \frac{1}{4}a^2 \\ (2R)^2 &= \frac{1}{2}a^2\end{aligned}$$

Solving for a give

$$\begin{aligned}(2R) &= \left(\frac{1}{\sqrt{2}}\right) a \text{ or} \\ a &= 2\sqrt{2}R\end{aligned}$$

Hence,

$$2R + 2r = 2\sqrt{2}R$$

or

$$r = (\sqrt{2} - 1)R = 0.414 R$$

$$r = 0.414 (0.129 \text{ nm}) = 0.053 \text{ nm}$$

By comparison, the radius of a carbon atom is 0.075 nm. Hence, interstitial C atom in an fcc Fe crystal will cause considerable local distortion but still the criteria of $r/R < 0.59$ is satisfied [187, 188]. Radius of Fe atom 0.129 nm. This atom will cause a lot of local distortion in fcc Fe crystal. Thus, self-interstitial solid solution strengthening is possible with a lot of local distortion.

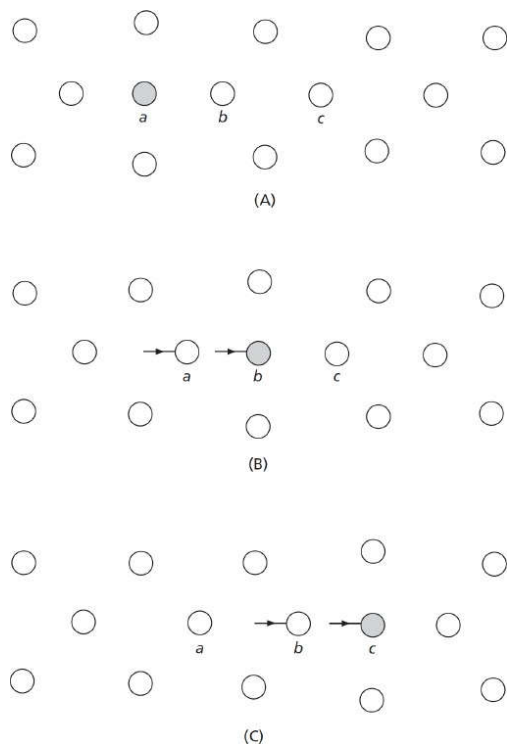


Figure – 4: (a) 100 FCC plane in fcc containing self-interstitial (shaded), (b) Interstitial motion – neighboring atom is pushed into another interstitial position, (c) each atom moves only short distance – low activation energy for interstitial motion (~0.1 eV) – diffusion of self-interstitial is faster than diffusion of vacancies [178].

5. Self-diffusion coefficient

Self-diffusion is commonly observed in pure metals and is mainly governed by vacancy mechanisms. It is seldom observed in alloys (such as steels) hence corresponding self-diffusion coefficient is seldom measured. Various models such as Zener model [189], cBQ [190-192] are employed to measure this.

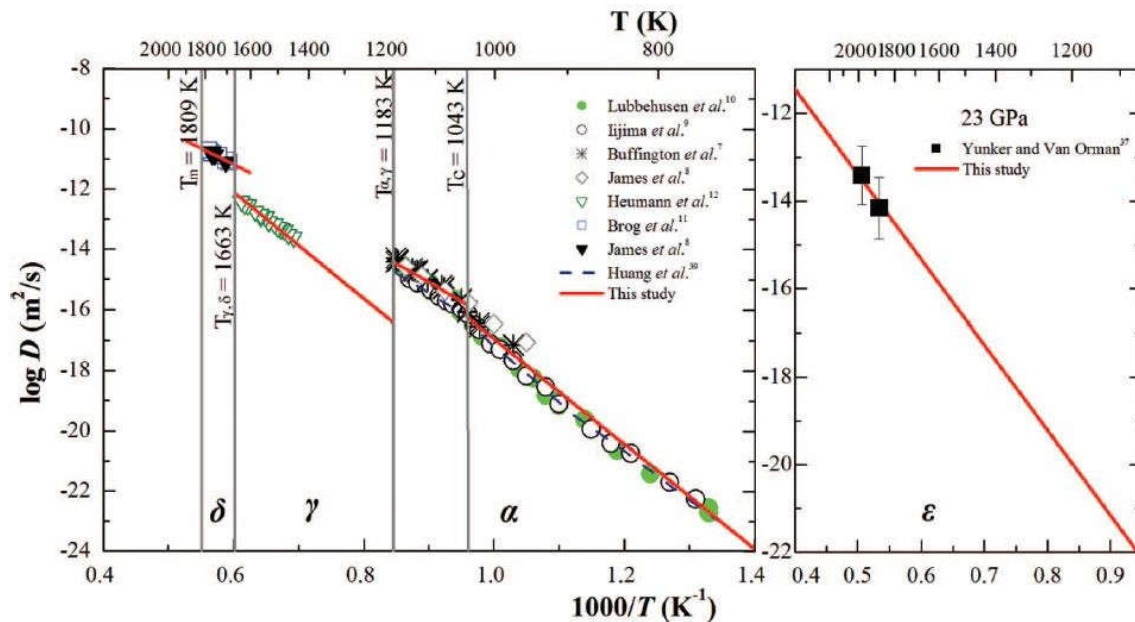


Figure – 5: Predicted temperature dependence of calculated self-diffusion coefficients derived from cBQ model (red lines) Vs published experimental data in α , δ , γ and ϵ – Fe [190-192].

Clearly, Self-diffusion coefficient of Fe in bcc (α) iron is higher than in fcc (γ) Fe. This is because of reasons mentioned earlier (Atomic packing factor of bcc is lower than fcc. Thus, it is easy to insert host / self (Fe) atom in its lattice irrespective of position).

6. Other plasma facing materials and conditions

Besides W, Be and carbon (in particular, fiber reinforced graphite) is the most frequently used PFM in today's magnetic confinement experiment. Depending on the selected fiber type and architecture, carbon fiber reinforced graphite can be manufactured with thermal conductivities equal to or higher than that of copper (up to about 400 W/m/K)

$$K_{cfr} \geq k_c$$

However, such an excellent thermal conductivity will degrade rapidly under the influence of energetic neutrons. In D – T burning fusion reactors with carbon walls tritium containing hydrocarbon deposits are formed on all in vessel components. This will finally result in an unacceptable tritium inventory in the fusion reactor under current licensing laws and limits or

- 1) Tritium ($^2\text{H}_1$) is retained in crystal lattice of Carbon hexagonal close packed structure.
- 2) This result in $^2\text{H}_1$ rich inventory.

For these two reasons, carbon has been discarded as a PFM for ITER. A study details thermal conditions experienced by W plasma facing material during operation [193]. These are, W armor plate (Type: Castellated, $W = 4\text{ mm}$, $H = 5, 8, 10\text{ mm}$, $T_h = 10\text{ mm}$, gap width = 0.5 mm (Figure 6)), base surface-temperature $400\text{--}780^\circ\text{C}$, steady state load with power density $5\text{--}10\text{ MW/m}^2$, applied transient heat load 0.2 GW/m^2 , time 0.5 ms in order to avoid plastic deformation of W. However, actual conditions are harsh and plastic deformation happens as discussed above.

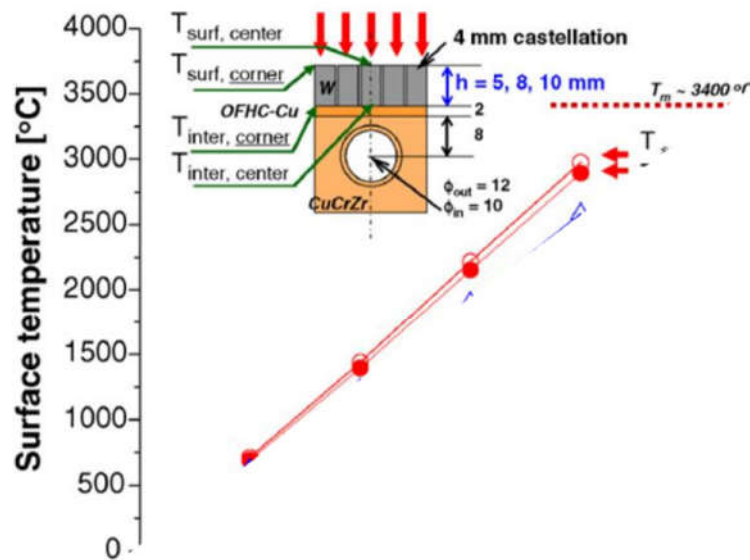


Figure – 6: Surface temperature Vs Power density for steady state heat transfer analysis of divertor castellated W plate using ANSYS [193].

Clearly, it can be seen, surface temperature reaches around 2500°C at power density of 20 MW/m^2 and interface temperature exceeds 650°C (exceeding operating temperature of Cu). Thus, is not desirable. Other designs include, 5.5 cm thick divertor consists of $\sim 40\%$ W-1.1TiC alloy, $\sim 12\%$ ODS ferritic steel, and 18% He coolant, by volume [194]. The 14 MeV source neutrons will activate the armor and generate radioactive materials at the end of the divertor service lifetime [97, 195].

7. Manufacturing, processing, and characterization

7.1 Phase diagram

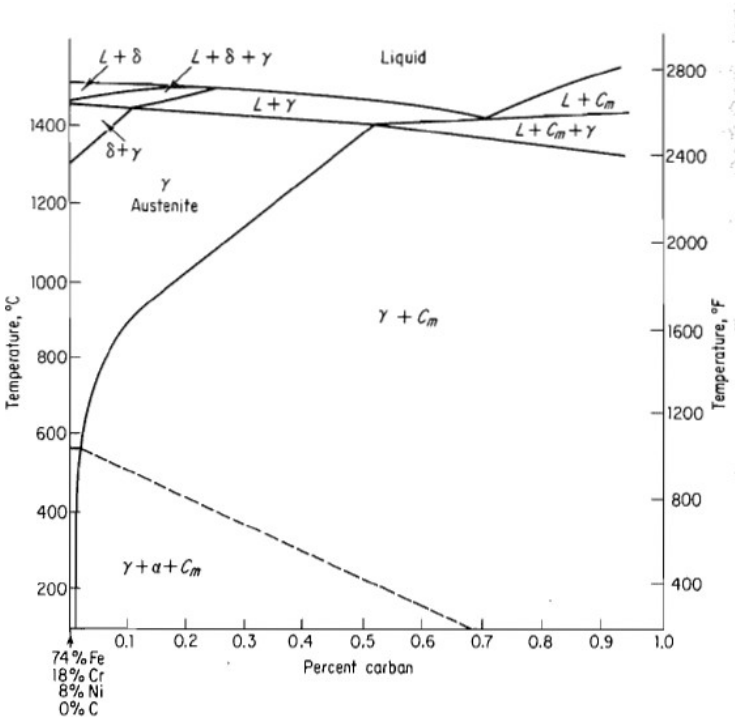


Figure – 7: Tentative cross section diagram showing trend of reactions in steels alloyed with 18% Cr, 8% Ni ([100, 196]

Above modified phase diagram of stainless steel is used to identify chemical composition, temperatures, and phases to be formed in alloy. Typical composition of 321 S.S is as follows

Table – 1: Composition of 321 S.S

Alloy	Designation	C	N	Cr	Ni	Mo	Mn	Si	Other	Other	Other
321	S32100	0.05	0.01	17.7	9.1	0.03	1.0	0.45	0.001S	0.03P	0.4 Ti

Typical production cycle consists of melting and permanent mold casting of plates using air induction of vacuum induction melting and casting furnace. A vacuum induction furnace exercises better control on chemical composition and resulting properties. This may be substituted with a better melting, control and use of fluxes during air induction furnace but it may result in poor quality both during melting and casting (such as air entrapment, misruns, inclusions, pin holes, gas holes, shrinkage, hot tears and surface degradation along with improper and poor control on chemical composition).

7.2 Heat treatment

Stabilizing heat treatment is performed to precipitate all carbon as TiC in 321 S.S (Ti variant) and prevent formation of Cr₂₃C₆ (sensitization). For titanium carbide, solubility may be determined from.

$$\log[Ti][C] = 2.97 - \frac{6780}{T}$$

Typical heat treatment cycle is

Heat = 1625 F, Hold = 4 hrs and cool

A detailed cycle may be obtained from below chart (Figure 8) based on carbon content of material.

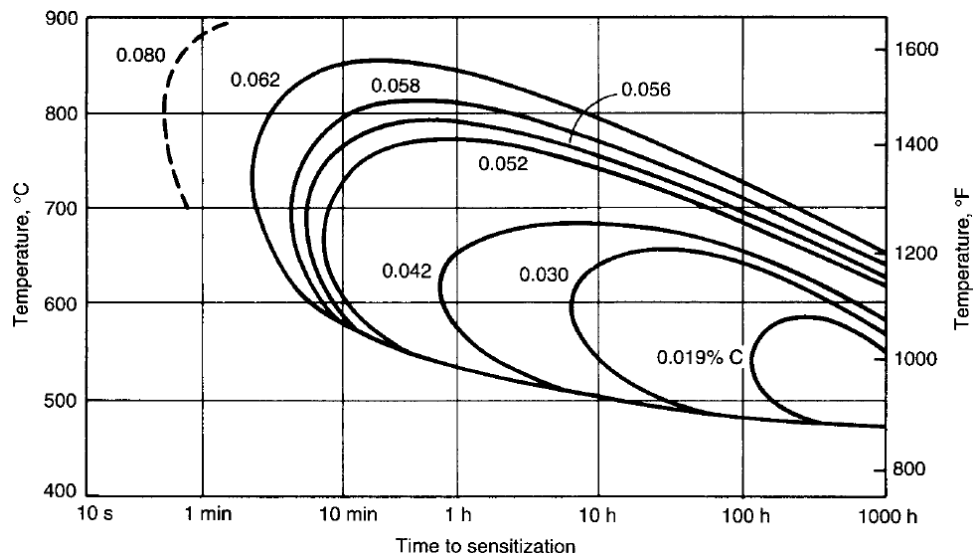


Figure – 8: The precipitation rates of Cr_{23}C_6 as a function of carbon content.

Surface heat treatment cycle (flame hardening) depends on part geometry and part thickness. The design calls for 10 mm thick plate directly acting as plasma facing material. In diverter region, a 0.5 cm thick sacrificial armor layer is used to protect the diverter during operation. Flame is operated at above lower critical line (A_3) for subject material. For a typical composition and above section thickness below cycle may be followed,

Heat = 700°C, Hold (60 – 120 seconds intermittently). A proper heat transfer analysis may provide actual time and temperature [197, 198].

7.3 Characterization

Bulk and surface characterizations are performed to assess and ascertain material properties prior and post testing. This includes detailed physical, chemical, mechanical, and nuclear property determination. Some of these are listed as under. **Physical:** SEM (SE and BSE), EDX (Chemical composition), WDX (phase), Thickness measurement (Ultrasonic probe during Non-destructive testing (NDT)), **Mechanical:** Yield strength, ultimate tensile strength, parentage elongation, hardness, and creep testing. **Nuclear:** Amount of eroded material, Optical emission spectroscopy (OES), Target mass loss technique, *In-situ* Quartz microbalance (QMB), *Deuterium* retention, Thermal desorption spectrometry and Nuclear reaction analysis.

8. Conclusion

1. Bubble is identified as crystal lattice (point) defect – not a physical / geometric feature.
2. A crystal lattice having the tendency to form self-interstitial solid solution has been identified.
3. FCC lattice γ – iron is identified as suitable crystal lattice.
4. Insertion of Fe (0.129 nm) in interstitial site (0.053 nm) will form interstitial solid solution with large lattice strain.
5. 321 Stainless steel is proposed as suitable material.
6. Co base superalloy is presented as second choice.

Self-interstitial solid solution strengthening is possible with a lot of local distortion which may be removed by surface heat treatment after stabilizing heat treatment.

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