Neutronics Scoping Studies for Experimental Fusion Devices

NEUTRONICS SCOPING STUDIES FOR EXPERIMENTAL FUSION DEVICES

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Abstract

A new Affordable Robust Compact (ARC) Fusion Reactor, which meets its goal in a cheaper, smaller but even more powerful, faster way to achieve Fusion Energy, with new high-field, high-temperature superconducting (HTS) magnets, has been designed in the US. SPARC will be the research tokamak aimed at the development of many ARC technologies. Ignitor is a proposed compact high-field tokamak that share with SPARC and ARC some design concepts, showing the convenience of this tokamak design development line. Neutronics and radiation damage scoping studies have been carried out for both designs. A general-purpose macroscopic model, set up by some of the authors in previous studies, has been used to estimate the radiation damage on selected machine components for the two cases. Solutions to solve the problem of radiation damage of the Toroidal and Poloidal Field Coils materials have been explored.

1. Introduction

Progress in technological fields such as High Temperature Superconductors, additive manufacturing and innovative materials has led to new scenarios and to a second generation of Fusion Reactors designs. A new Affordable Robust Compact (ARC) Fusion Reactor, which meets its goal in a cheaper, smaller but even more powerful, faster way to achieve Fusion Energy, with new high-field, high-temperature superconducting (HTS) magnets, has been designed by MIT (Massachusetts Institute of Technology) and PSFC (Plasma Science and Fusion Center).[1,2] ARC too might face radiation damage problems, besides neutron–induced activation, either for the Vacuum vessel, or for the Poloidal and Toroidal Field Coils (PFC and TFC). For ARC, the selected superconducting material for the magnets is Rare Earth Barium Copper Oxide (REBCO), a high temperature superconductor that can work at temperature up to 80 K, which is higher than the one for Nb3Sn used for ITER. ARC will breed its own tritium by means of a FLiBe self-cooled blanket.[1]

Ignitor is a proposed compact high-magnetic field tokamak[3] that includes in its design many features that served, together with the ALCATOR design and experimental results, as a basis for the development of the SPARC-ARC concepts. Ignitor is aimed at studying plasma burning conditions in Deuterium-Tritium plasmas: it has a major radius of 1.3 m, minor radii of 0.47 m and 0.87 m, a peak plasma temperature of 12 keV, a peak plasma density of 10^{21} ions/m^3, at

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[1] The MIT PSFC has also begun developing a conceptual design for SPARC, a compact, high-field, net fusion energy experiment. SPARC would be the size of existing mid-sized fusion devices, but with a much stronger magnetic field. Based on established physics, the device is predicted to produce 50-100 MW of fusion power, achieving fusion gain, Q, greater than 3. Such an experiment would be the first demonstration of net energy gain and would validate the promise of high-field devices built with new superconducting technology. Even if SPARC will have lower fusion power than ARC, it will have smaller dimensions, and thus activation and radiation damage aspects must be studied too.

a maximum fusion power of 90 MW. Pulses at different power levels are planned, with either DD or DT operation, distributed over a global operation time of 10 calendar years. The peak neutron wall loading at maximum fusion power is about 2 MW/m^2. The tokamak main components are: a graphite first wall, an INCONEL625 vacuum vessel, the Cu-based toroidal magnets, and the AISI316 machine structure (named "C-Clamp").

A comparison of the two designs is available in Table I. In this paper, some neutronics scoping studies will be carried out for both tokamaks (Ignitor and ARC). A general-purpose macroscopic model, set up by some of the authors in previous Ignitor studies, has been revised and used to estimate the radiation damage on selected machine components for the two cases. Solutions to solve the problem of radiation damage of the toroidal and poloidal field coils (TFC and PFC) materials have been explored.

Table I - Ignitor and ARC Tokamaks parameters

PARAMETER	Ignitor	ARC
Major radius	1.32 m	3.3 m
Toroidal Magnetic Field	13 T	9 T
Plasma Volume	10 m3	141 m3
Fusion Power (MW)	90	525
Electric Power (MWe)	--	283
Q	Infinite	3
Pulse flat top	10 s	infinite
Magnets Temperature	4 K	80 K

2. A model for neutronics and radiation damage scoping studies

Previous Ignitor studies[4] have been found relevant to set up a model for the ARC radiation damage case too.

In a simple *macroscopic* model the number of displaced atoms depends on the total available energy E_a and the energy required to displace an atom from its lattice position E_d

$$\mathrm{DPA} = \kappa \frac{E_a}{2E_d} \tag{1}$$

If the neutron flux and spectrum is known, then we have:

$$\mathrm{DPA} = \frac{\int dV \int dE \Phi(\vec{r}, E) \sum_{i=1}^{N} \rho_i \sigma_{R,DPA,i}(E)}{\sum_{i=1}^{N} \rho_i} \tag{2}$$

summed over all N isotopes in the material mix, where $\sigma_{R,DPA,i}$ is the DPA cross section for the isotope i, and ρ_I is the atomic density.

The most recent version of the inventory code FISPACT,[5] which has been extensively updated (now called FISPACT-II) is now able to compute radiation damage quantities, such as dpa (Displacement per Atom), Gas Production, KERMA (Kinetic Energy Release to Matter) factors. The code has been found a suitable tool for our neutronics scoping studies. FISPACT-II, in fact, can calculate dpa rates directly using the above formulae, with the latest cross-section nuclear data libraries, such as the TENDL-2017 library.

3. Neutronics and radiation damage scoping studies for Ignitor

Radiation damage calculations for Ignitor have been performed by means of the MCNP code.[6] In particular, the total dose on the TFC insulator DTFC) turned out to deserve some attention. In particular, results are as follows:

$$DTFC = 9.01\ 10^{-23}\ MGy/(neutron\ generated\ in\ the\ plasma) \qquad (3)$$

If we now evaluate the dose on the TFC insulator based on the total neutron production foreseen for Ignitor[7], namely 9.22 10^{22} neutrons in 7 years of DT operation, we have:

$$DTFC = (9.01\ 10^{-23}\ 9.22\ 10^{22})\ MGy = 8.3\ MGy \qquad (4)$$

The TFC insulator is made of G10CR, a bisphenol-A epoxy resin, chlorine-free, low-N, low O, with a boron-free high-strength E glass tissue. The radiation resistance of G10CR material has to be carefully considered, since it appears to be poor, and possible substitute materials must be looked for. Alternatively, some additional shielding must be foreseen.

The G10CR resin was found to have a very low radiation resistance, with a degradation of its mechanical properties as early as an irradiation of around 3 MGy was applied.[8] On the other hand, the similar material G11CR did not show any relevant degradation. Another early important work in the field of radiation damage of those materials Ref. 9) stated that the G-10CR matrix is a conventional, heat-activated, amine-catalyzed, bisphenol-A, solid type epoxy resin chosen for proven performance at cryogenic temperatures, while the G-11CR matrix is an aromatic-amine-hardened, bisphenol-A, liquid-type epoxy resin expected to provide improved resistance to radiation damage.

The radiation resistance limit for the epoxy used in ITER is 10 MGy (Ref. 10). That resistance does not seem enough to meet ITER requirements on a safe side, and development of novel insulators has been carried out in the recent years. Cyanate esters (CEs) offer enhanced temperature and radiation resistance as well as high mechanical strength.[11] No technological challenges in the coil fabrication are needed, since CE resins can be treated like epoxies and are compatible with the vacuum pressure impregnation (VPI) process, which is foreseen for the fusion coils due to their complexity. These materials are being considered also for replacement of insulators in ITER. Some recent publications[10] show the excellent radiation resistance properties of these materials. Several mixtures of CE and epoxy resins were investigated, and the pure CE composite, named T1, maintained about 75% of its mechanical strength, compared to the unirradiated state, even at a fluence of 5 10^{22} m^{-2}, which corresponds to a dose of 250 MGy. In conclusion, Cyanate esters could be the best candidate for Ignitor too. Recent literature shows their excellent characteristics,[10] and the product is available on the high-tech materials market.[11]

A scoping study was carried out in order to determine the best shielding material if no modification is made to the insulator. Three advanced materials were selected, based on their favourable physical properties for radiation shields, i.e., Solid lithium hydride (LiH), powdered zirconium dihydride (ZrH2), and powdered zirconium borohydride (Zr(BH4)4). The more traditional DPE (High-density Polyethilene), Borotron ® HD050 and B4C have been considered too. To assess their nuclear performance, the total radiation doses to the TF coils and neutron heat deposition were obtained with coupled MCNP-FISPACT-II calculations. The three advanced shielding materials all turned out to be better than the traditional materials. In particular, ZrH2, with its high mass density, high hydrogen content, and high average atomic number, was the best performing one. Advanced neutron shield materials (Zr- and Li-hydrides) form high performance neutron/gamma shields, within possible Ignitor design modifications. In particular, a 10-cm shield of Powdered zirconium dihydride (ZrH_2) can reduce the dose on the insulator by a factor 100:6 (if the dose is 100 in arbitrary units without shield, it becomes 6 with it). Higher shield thickness is not recommendable from the design viewpoint.

As far as radiation damage on vessel materials, a scoping study has been performed for several candidate structural and plasma-facing materials. Results are shown in figure 1. Due to the low foreseen fluence of Ignitor, it turns out that radiation damage for those components is not an issue, while their shielding capability is. Inconel 625 has the best shielding performance, while – as far as first wall tiles are concerned – C, due to its high moderation characteristics, has to be preferred to Mo: even if absorbing less neutrons, it enhances neutron capture in the vessel by slowing down neutrons.

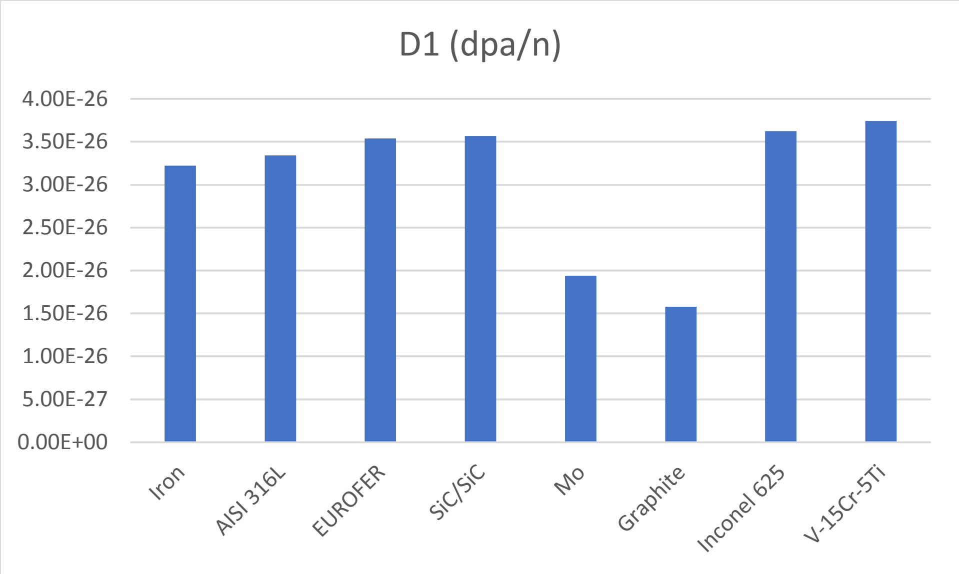

Figure 1 – Radiation damage in vessel and first wall Ignitor materials

4. Detailed studies for the Affordable Robust Compact (ARC) tokamak

ARC is a tokamak designed by MIT scientists. It has been conceived for a FNSF (Fusion Nuclear Science Facility) and pilot plant for producing power,[1] therefore it will be used to collect data about over-energy breakeven reactors and to demonstrate the feasibility of a nuclear fusion power plant connected to an electric grid.

Magnets are the main novelty of ARC reactor: TF coils and PF coils rely on High Temperature Superconductor (HTS) technology, now available on commercial scale. HTS are alloys based on Rare Earths (more specifically Yttrium) Barium and Copper Oxide (REBCO or YBCO) and their critical temperature is as high as 80K. The adoption of the HTS technology for the magnets allows obtaining more compact and cheaper tokamaks than those using traditional technologies based on 4K superconductors. The downside of this technology is that REBCO tapes are very sensitive to neutron damage.[12,13] Also, since ARC's TF coils are demountable, PF coils can be placed inside them in order to achieve a better plasma control.[2] This approach was followed by many tokamaks, such as DIII-D[14] and Alcator C-Mod.[15] However this may raise concerns about neutron damage that would be not only on TF coils, like other tokamaks, but also, and even with a higher rate, on PF coils.

ARC is Deuterium-Tritium fueled. A D-T fusion plasma is a huge source of fast neutrons and, more specifically, ARC's plasma, supposed to achieve 525 MW of fusion power, will emit about 2.2×10^{20} n/s (Ref. 1). Therefore, ARC is a relatively small device, with a high density of complex components near its center, and a strong neutron source at the core. This surely requires careful neutronics studies and effective shield design proposals.

The ARC shield can be split into two main components. Starting from the plasma core, the first neutron shield is the Nickel-superalloy-based vacuum vessel (VV); its first aim is to provide a vacuum and pure volume for plasma and not being an effective neutron shield, however shielding properties of VV materials will be addressed in this scoping study. The actual neutron shield of this device, however, is the blanket: in fact, the vacuum chamber is immersed in a tank full of a LiF BeF_2 (FLiBe) molten salt. This fluid works as a neutron multiplier, tritium breeder, and coolant, while the bulk tank is a quite effective shield for PF and TF coils.

MCNP simulations[1] have shown that an additional shield was necessary in order to safeguard TF coils, and a TiH_2 layer has been placed all the way around the FliBe tank. TiH_2 has been chosen as it confirmed – like in the case of Ignitor – its excellent moderation and absorption characteristics. Further simulations demonstrated that this configuration is actually able to guarantee a lifetime of 9 FPY (Full Power Years) to magnets.[1]

A second ARC design session lead to some design modifications: long leg, double null divertors, able to effectively exhaust huge heat fluxes, have been added to the machine.[3] The neutron flux in the divertor area (at the outer surface of the VV at the midplane of the divertor foot feature) is a factor of thirty lower than at the outer midplane, while the neutron spectrum is significantly softened due to the moderation provided by the FLiBe.[2] The reduction of fast neutron flux drastically reduces neutron damage rates, in terms of both displacements per atom (dpa) and helium production from neutron-alpha interactions. A lower damage rate in the divertor represents an advantage for those high heat flux components. However, the new divertors raise the complexity of the required magnetic configuration, asking for additional coils. Furthermore, as previously mentioned, PF coils have been moved inside the TF ones and FLiBe tank has been reshaped.[2] An improved Monte Carlo model for the new configuration highlighted the necessity of more effective shielding for the PF coils in their new location. It has been chosen therefore to add 25 cm thick ZrH_2 plates directly inside the FLiBe tank, in order to shield the magnets.[2] In this case, the role of the 5 cm-thick vessel gives a small but relevant contribution in reducing the neutron flux on the PFC, even if most absorptions take place in the 1 m thick FLiBe blanket: the additional shield, with its high hydrogen density, softens the neutron spectrum and enhances neutron absorptions in the blanket. Altogether, those structures can reduce the neutron load on PFC of roughly two orders of magnitude: more specifically, total flux is around to 3×10^{18} n/cm^2, with a soft energy spectrum, and that guarantees to the TFC a lifetime of 10 FPY at least.[3]

However, the mentioned previsions are supposed to be quite pessimistic. In fact, 3×10^{18} n/cm^2 is the limit set for NB_3Sn superconductors, used for ITER,[16] while REBCO tapes are supposed to have less constraining limits.[12] Furthermore, limits were not set at the failure point of the material, but rather at a point at which the critical current begins to degrade.[2] In addition, recent literature[13] has shown evidence that it is possible to mitigate critical current degradation, due to fast neutrons, decreasing operating temperature (i.e. for operating temperature of 30 K and roughly 3.2×10^{18} n/cm^2, YBCO critical current seems to be still very similar to its initial, not irradiated one). Low energy neutrons – on the other hand - have hardly any effect on the superconducting properties of YBCO tapes.[13] Even if ARC's magnets will be running with a significant margin to their critical current, design modifications could affect these shortcomings, like for instance the substitution of the VV material, or an optimization of the ZrH_2 shield thickness.

Concerning radiation damage in the vessel material, a study has been performed in order to verify the susceptibility to radiation damage of the proposed low-activation substitute of Inconel718, that is, the V-15Cr-5Ti alloy. By means of FISPACT-II, with the already mentioned new library, results available in Figure 2 have been obtained, where radiation damage in terms of dpa/FPY (Full Power Year) are shown. Dpa rate in V-15Cr-5Ti is just less than 5% higher than the same figure for Inconel718. While it is actually unknown the onset threshold for some radiation damage effects in Inconel718 or V-15Cr-5Ti, they should reach a lifetime irradiation lower than 100 dpa before being replaced for other reasons.

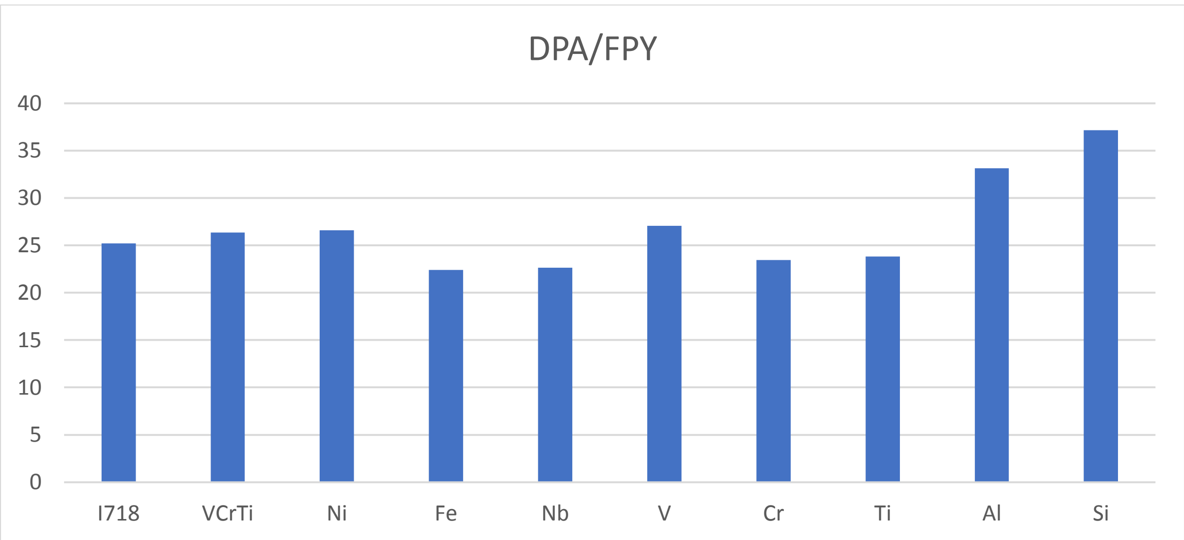

Figure 2 – Radiation damage (DpA per Full Power Year) in Inconel 718 (I718) and V-15Cr-5Ti (VCrTi), and in their main alloying elements.

The shielding capability of the vessel material can be a relevant question, given the above-mentioned necessity to shield the magnets. A good parameter to estimate the shielding performance of a material exposed to neutron flux is the KERMA rate (kW/cm^3). Figure 3 shows results for Inconel718, V-15Cr-5Ti and their main alloying elements. It turns out that the KERMA rate for V-15Cr-5Ti is significantly (-35%) lower than for Inconel718, mainly due to the lack of Nickel. This fact will have to be taken into account if the new vessel material is adopted for ARC. Not only the shielding capability of the new material is lower, but also the effect of Inconel718 substitution on the neutron spectrum and then on the neutron absorptions in the FLiBe blanket will have to be evaluated.

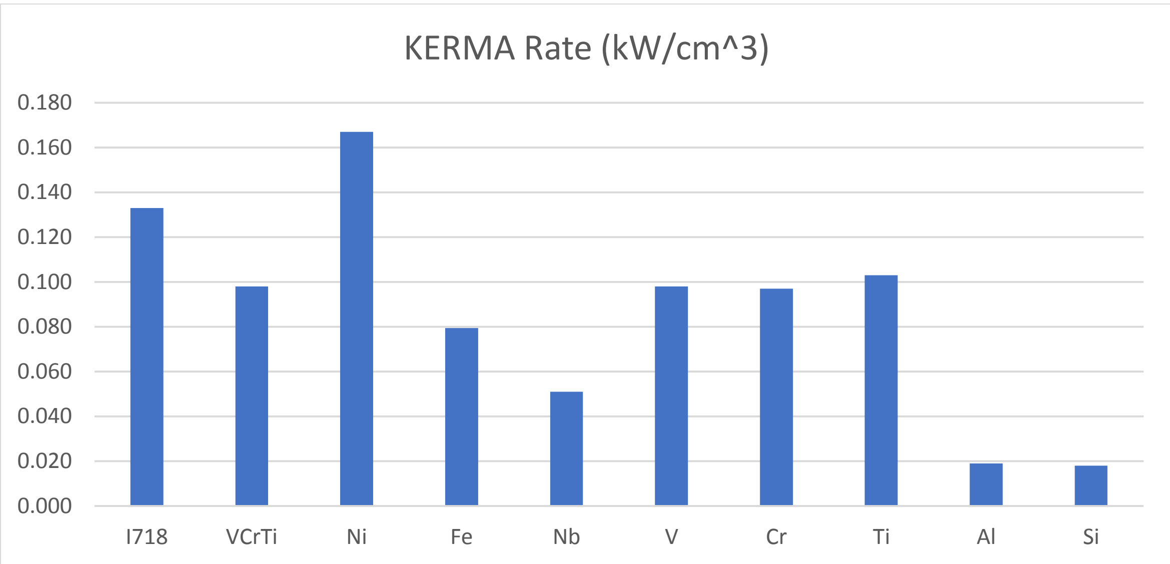

Figure 3 – KERMA Rates in Inconel 718 (I718) and V-15Cr-5Ti (VCrTi), and in their main alloying elements.

5. Conclusions

Neutronics scoping studies have been carried out for two different compact high-magnetic field tokamaks: Ignitor and ARC. A general-purpose macroscopic model has been set up, to estimate the radiation damage on selected machine components for the two cases.

Solutions to solve the problem of radiation damage of the Poloidal and Toroidal Field Coils materials have been explored. In particular, for ARC, it has been found that ARC's magnets will be running with a significant margin to their critical current, however the effects of the substitution of Inconel 718 with V-15Cr-5Ti as vessel material will have to be taken into account.

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