A mini review on a study of Helium behavior for fusion reactor development

P. PRAPAN
Interdisciplinary Graduate School of Engineering Sciences, Kyushu University

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A mini review on a study of Helium behavior for fusion reactor development

P. PRAPAN[1*]
[1]Interdisciplinary Graduate School of Engineering Sciences, Kyushu University, 6-1 Kasuga-Koen Kasuga, Fukuoka 816-8580, Japan
*Corresponding author email: pdavin.99@gmail.com

Abstract: A fusion reactor is thought to offer the answer to the present global warming crisis. However, there were a few rooms that needed further examination. Helium research typically focuses on three key areas: 1) the transportation of helium from the plasma core as the D-T product; 2) the use of helium as the working gas for the removal of tritium, and 3) helium interaction with various plasma facing walls. The ITER plasma facing component is currently made entirely of metal, specifically beryllium and tungsten. Predicting the ITER plasma and the commercial stages of reactor development may depend on our ability to understand helium.

Keywords: Fusion; Helium; Plasma facing wall; Plasma material interaction

10. INTRODUCTION

Fusion energy potentially provides the solution to the cure to the issue of global warming and clean renewable energy. Helium is a special study in the fusion reactor's construction. The reactor's main fuel now is created by the fusion of deuterium and tritium. We must investigate the transportation of helium from the plasma core to the edge of plasma before being removed from the reactor by the exhaust system [1-3]. One of the important difficulties is plasma wall interaction (PWI). The biggest drawback of using tritium as a fusion fuel is the tritium retention on the vacuum vessel wall. The decision on the plasma facing wall will affect the reactor's long-term performance. Helium was also considered a tritium cleaning gas in various types of walls [4-11]. To determine the interactions and behavior of the helium inside the reactor, laboratory studies were required [12–16]. The limited availability of tritium makes it necessary to continue helium research to potentially replace it with helium-3 (He-3), which offers lesser neutron energy. The primary energy carried by charged particles and alpha particles leads to the direct creation of electricity. Helium may not be used in the first generation of fusion reactors. Nevertheless, a tritium bottleneck will arise later. Thus, the deuterium-helium3 fusion reactor may succeed the Deuterium-Tritium (D-T) type as the most effective fusion reactor available at the time.

2. Helium research in fusion reactor development

2.1 Helium exhaust

The deuterium-tritium reaction produces helium. It is essential to research helium transport inside reactors. Analyzing helium ash outflow is one way to learn more about the behavior of helium. The analysis of helium transport has been reported in E.J. Synakowsky et al. (1995) [2]. The pump-out helium ash from the reactor exhaust system can be measured by employing a helium gas puff system to study the transportation of helium from the core position. The outcomes of the simulation are compared with the experimental measurement. The results after helium was pushed into D-T plasma demonstrated that the model's prediction of the helium ash exhaust density is accurate. The comparison is displayed in Fig. 1 a) and b). Both demonstrate the helium's transport out of the core. The result of the no transport model indicates that if there was no transportation, the plasma density should have increased. Fig. 1. b) shows that the helium ash source near the wall position has a larger peak than the heating source like the core plasma, even if the measurement results coincide with the gas puff estimate in Fig. 1. a).

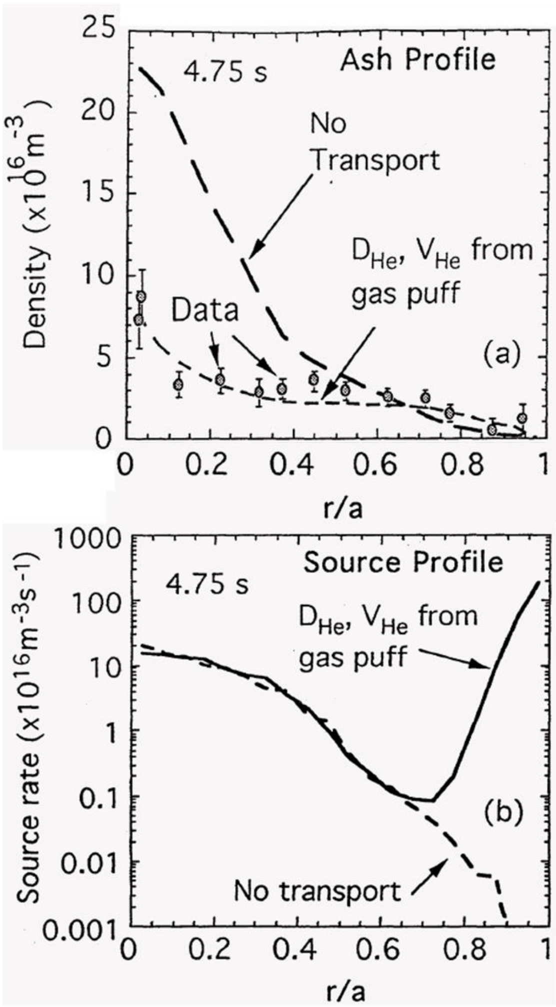

Fig. 1. Comparison of the gas puff rate in a) with the amount of helium ash measured from the exhaust system in (b) have been over-plotted with a computation of the plasma density without helium transportation from the core plasma (thick dashed line) in the radial density profile over the plasma minor radius (r/a) for both figures. [2]

With a carbon W-shaped divertor, the research on the helium ash exhaust was also reported in JT-60U [3]. Investigation of the impacts of pumping out helium ash is required. Results of the helium ash pumping out in JT-60U's ELMy H-mode operation, which is also expected to operate in ITER, show that the helium exhaust system from the W-shaped divertor single null is enough for detached plasma operation for ITER.

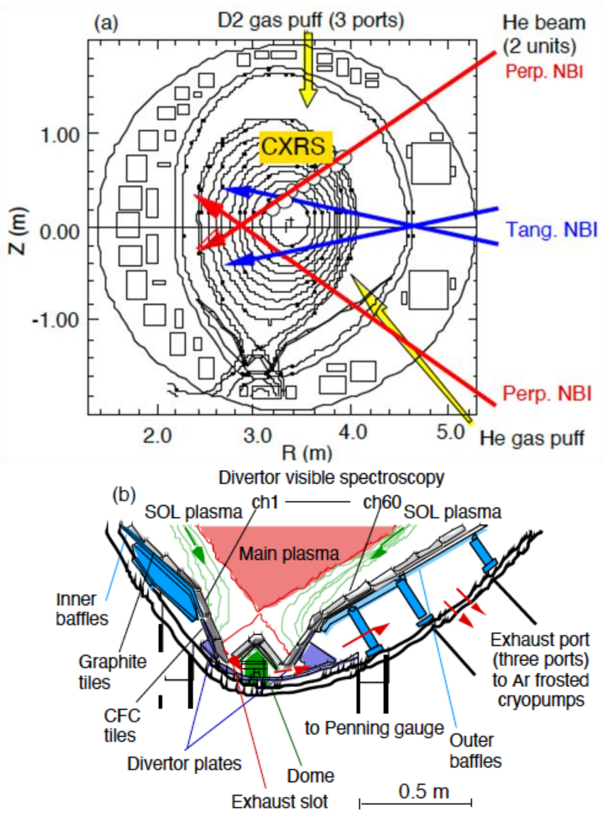

Fig. 2. a) The ELMy H-mode plasma setup in JT-60U with a single null divertor structure is depicted in figure a). It also includes helium beam injection with perpendicular and tangential directions, helium gas puff, and charge exchange recombination spectroscopy (CXRS) observation lines. The W-shape divertor's construction is depicted in figure b). Improved gas exhaust is the goal of the dome configuration [3].

Helium behavior in DEMO, which aims to be more compact than ITER for the commercial phase, is also investigated in the simulation study. The edge transportation modeling findings for DEMO ELMy H-mode plasma are presented in H. Pacher et al. (2007) [1]. With the helium neutral influx decreasing and the continuous power input to the divertor remaining constant, the helium density inside the separatrix structure stays constant.

2.2 Tritium removal

The principal fuel for the present version of the fusion reactor is tritium-deuterium. Because deuterium is abundant on Earth, there are various studies involving deuterium recovery from vacuum vessels. Throughout the decades, deuterium has been effectively recovered in a variety of ways, including using a pumping system during operations, a cleaning discharge to allow for deuterium retention, collecting deuterium from the vassal wall, and so on. However, because of tritium's rarity, only a few reactors have done tritium experiments. Tritium manufacture may be too expensive for the final commercial stage. Tritium recovery has emerged as a crucial issue for the evolution of fusion reactors. They discovered varying retention levels of tritium in the vessel wall from the few reactors that were constructed with various types of plasma facing wall (PFW) materials and have performed tritium-related experiments, such as the Joint European Torus (JET) [10], the Large Helical Device (LHD) [5], the Tokamak Fusion Test Reactor (TFTR) [2,8,9], and the Tokamak Experiment for Technology Oriented Research (TEXTOR) [4]. After investigating various techniques attempting to detach the tritium from the wall, they discovered that tritium was more difficult to remove than deuterium. Plasma discharge, the usage of ionized gas to recover tritium from a chemical or physical process, is one of the techniques for vessel wall cleaning.

TFTR with carbon component PFW in 1995 [8,9], it was discovered that 40% of the tritium input was trapped in the reactor wall and 50% was pumped out during the experimental operation. Due to the dynamic retention of tritium close to the material surface, tritium is easily combined with deuterium at the beginning of the discharge, which results in deuterium glow discharge cleaning (D-GDC) having the highest tritium removal rate. The recuperation rate starts to decline after a while. The graphite surface that could retain tritium was etched away in GDC made from helium-mixed oxygen (9:1 ratio) plasma, which has a lower removal rate but remains constant over time.

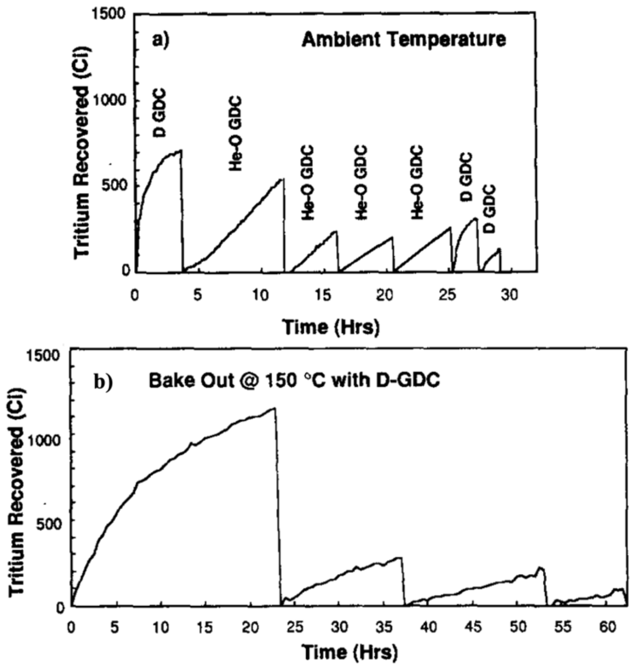

Fig. 3. The removal of tritium for almost 30 hours from the TFTR using D-GDC and He-O GDC displays in a) without wall baking, while b) is the result for over 60 hours from the cleaning process with wall baking. The strong T removal at the beginning of the discharge from D-GDC is depicted in (a), which thereafter substantially dropped at the 25-hour point. (b) demonstrates greater

tritium recovery from the start time before seeing a major decline at about 25 hours [9].

The amount of tritium extracted from the TRFT vessel alone from D-GDC and He-O GDC without wall bake (Fig. 3. a) and D-GDC with baking the wall at 150 °C (Fig. 3. b) reveal that tritium was eliminated from D-GDC cleaning process at a very high rate at the start of the discharge but dramatically decreased for the following procedures in both scenarios. Table 1 demonstrates that wall baking with D-GDC resulted in the greatest quantity of T elimination.

Table 1. The outcomes of GDC operations with and without 150 °C wall baking [9].

Activity	T removed (Ci)	T retention amount (Ci) before activity	T retention amount (Ci) after activity
After D-T campaign	0	16399	16399
1st D-GDC (No wall baking)	687	16399	15713
He-O GDC	1249	15713	14463
D-GDC (No wall Baking)	495	14463	13968
150°C with D-GDC	1609	11867	10258

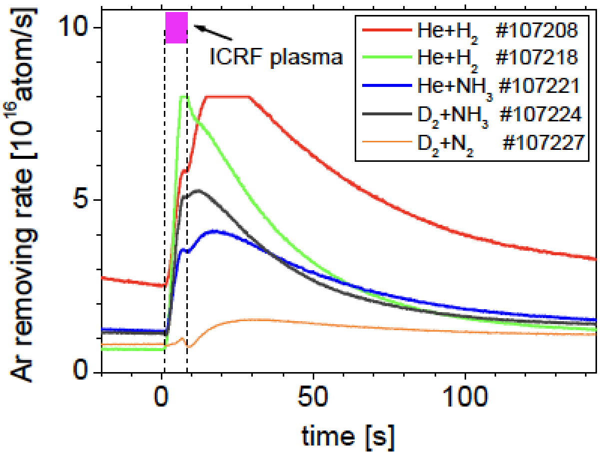

Fig. 4. Argon removal by the different gas combinations used during ICWC operation to investigate wall cleaning ability for TEXTOR. [4]

Additional to the investigation of the interaction between carbon material and helium, CX-2002U [12] has been exposed to D-T plasma first then follow by helium plasma to replicate the tritium removal procedure using helium plasma. But when introduced to air plasma, the tritium elimination was more efficient. While oxygen in the air reacts chemically with carbon as seen in the results of TFTR GDC with He-O mixed plasma, helium eliminates tritium with simple physical sputtering. The issue that needs to be considered is the impurity from trapped oxygen in the sample. The trapped oxygen in the PFW may be released back into the plasma during the operation of the fusion phase after T has been removed. It might reduce the plasma core's temperature when a high-temperature core was required to produce a fusion reaction.

In the Japan Torus-60 upgrade, additional research on the interaction of carbon-type walls was conducted (JT-60U). Tritium(deuterium) elimination was studied by H. Nakamura et al. (2004) using both hydrogen discharge and helium discharge. According to the data, He-GDC predominates due to the physical sputtering process on the carbon wall whereas H2-GDC has a higher removal rate due to the chemical reaction, which is consistent with the TRFT results employing He-O GDC. This indicates that employing D-/H2-GDC with or without wall bake will be able to remove tritium retention in the carbon wall fusion device through the isotope exchange process or chemical procedure.

A superconducting fusion reactor called the Wendelstein 7-X also uses a carbon divertor to operate. He-GDC was employed to study the wall condition. Due to the large hydrogen inventory of the carbon wall, He-GDC was required between operating days, according to the findings of T. Wauters et al. (2018) [11]. He-GDC eliminates impurities like methane (CH_4), carbon monoxide (CO), and carbon dioxide (CO_2) in addition to offsetting hydrogen. Although the metal portion of the wall, such as stainless steel, retains helium from the cleaning shots, the carbon component is properly clean.

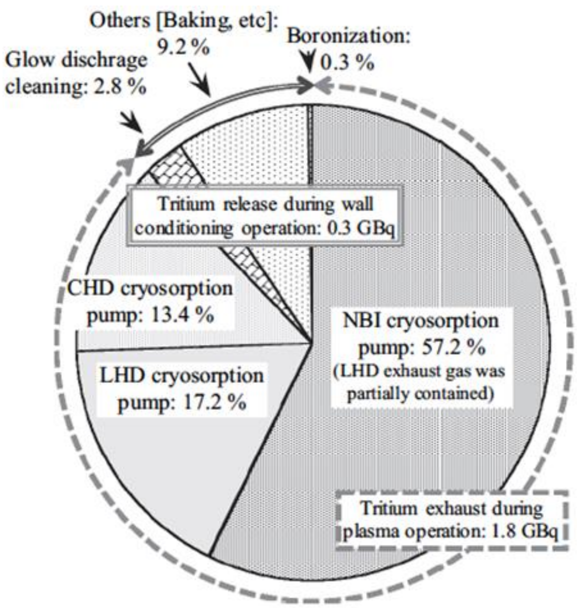

Fig. 5. The tritium exhaust from the LHD during the 2017 deuterium plasma experiment, which produced 6.4 GBq, displayed in percentage for various operations. The remaining 4.6 GBq was trapped in the PFW, while 1.8 GBq was pumped out during the experiment. Only 0.3 GBq of energy was released from the plasma facing component during the wall conditioning procedure [5].

With the largest reactor today, JET has successfully performed D-T plasma with a carbon component for the ITER previous plan. The study of tritium removal in JET is significant to demonstrate the wall cleaning system in ITER. In P. Andrew et al. (1999) [10], tritium content in the JET carbon vessel is too high for safe operation, which agrees with the previous deuterium campaign. The

results from GDC show no significant T removal. Similar results also happened for the electron cyclotron resonance heating (ECRH) in the attempt at tritium removal. For the carbon component, the addition of oxygen might be necessary due to its active erosion on the carbon surface.
Metals like tungsten and carbon components were both studied in the case of T retention. High heat resistance is necessary since plasma has a very high temperature. Due to its strong heat resistance, low atomic number, and lack of discernible effects in plasma, carbon is one of the most likely possibilities. However, another material was explored because of the high tritium retention of carbon. ITER will operate differently from the original proposal, using a tungsten divertor and a beryllium first wall. It is necessary to analyze T retention in Be and W.

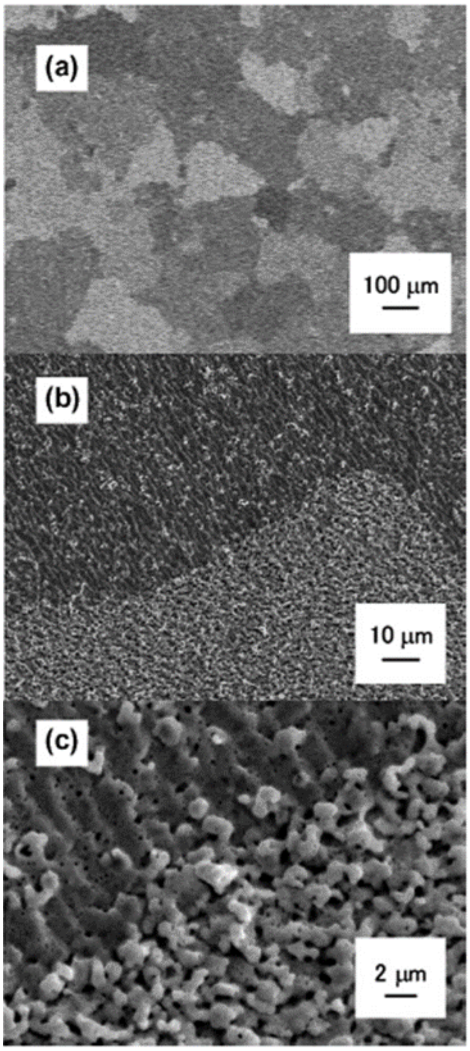

Fig. 6. The SEM results of subjecting an ITER grade tungsten sample to the helium plasma are shown. b) and c) are larger copies of a). Helium bubbles' nonuniformity is demonstrated [15].

The examination of appropriate wall conditioning for tritium removal in TEXTOR with boronization for wall coating was reported by V. Philipps et al. (2008). Ion cyclotron wall condition (ICWC) was carried out with pre-argon retention into the TEXTOR wall using various gas combination ratios. Unlike GDC, which cannot be carried out with the magnetic coil turned on, ICWC can be operated in a magnetic field. In shot number 107218 with 0.2T of ammonia (NH3), helium, and deuterium, more argon was removed than in shot number 107208 with 2.25T of helium, deuterium, and ammonia, according to figure 4.

Table 2. Comparison of tritium exhaust rate between JT-60U and LHD [5]

Device	Plasma facing wall material (1st wall, divertor)	T exhaust rate	T production from the experimental campaign
JT-60U	(C, C)	28%	31 GBq (1991-1993)
LHD	(SUS, C)	32.8%	6.4 GBq (2017)

Some reactors choose to operate in both low atomic numbers, carbon for the divertor, and high atomic numbers, stainless steel that contains the iron component. The study of tritium removal in large helical devices (LHD) by M. Tanaka et al. (2022) [5], shows in Fig. 5. that around 28% of the tritium product during the first deuterium plasma campaign was pumped out during the operation. while almost 70% was trapped in the plasma facing component of the reactor. After performing various cleaning methods, including He-GDC, only 7% of tritium retention has been released from the wall. Compared with JT-60U results, as shown in Table 2., they found a higher tritium exhaust rate in LHD due to the mixing of the metal wall in the vessel.

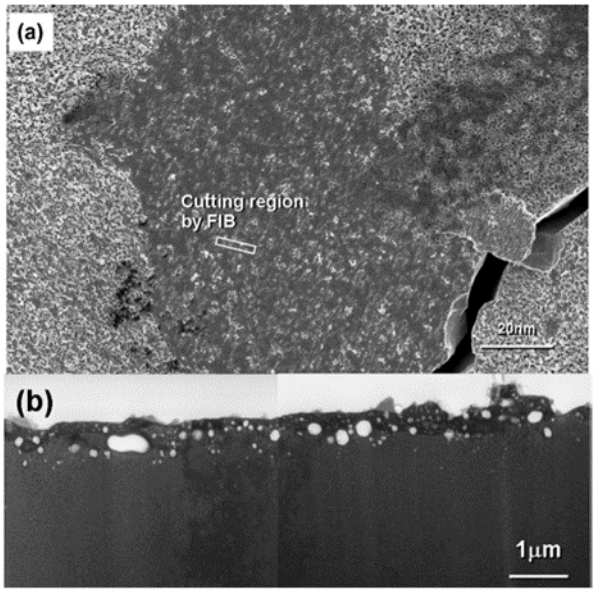

Fig. 7. The SIM depicts non-uniform bubble development on the tungsten surface that resembles the ITER in (a). A non-uniform bubble structure can be seen in the sample's cross-section surface when it was cut using the Focused-Ion beam (FIB) milling technique [15].

2.3 Effect on the material

As previously mentioned, it is essential to research helium on a material surface in a scientific setting. The findings of T. Todokoro et al. (1998) for carbon reveal that air successfully eliminated tritium compared to

helium plasma exposure because of the oxygen component in the air plasma. But given that the ITER PFW is constructed of metal, it is important to look at how helium affects the metal. The actions of helium on the carbon and metal components produce noticeably distinct outcomes. Due to the development of helium bubbles, helium is retained more in the metal surface. Several studies of helium exposure to metal were conducted on a laboratory scale to precisely determine the conditions under which the bubble is generated and trapped. Several laboratory-scale studies of helium exposure to metal were carried out to better understand the circumstances under which the bubble is produced and trapped.
According to P.D. Edmondson et al. (2013) [14]'s findings on the bubble distribution over ferritic alloy, the size of helium bubbles grew as the influx of helium on the 14YWT nanostructure ferritic alloy rose. And the results of Atom Probe Tomography (APT) and Transmission Electron Microscopy (TEM) revealed the combination of the different locations: 4.4 percent are found on coarse precipitate, 12.2 percent are found at alloy dislocations, 14.4 percent are found at the grain boundary, and 48.6 percent are found on nanocluster. The findings showed that the alloy's nanostructure was where helium bubbles were mainly induced.
A study of bubble formation on tungsten, particularly tungsten of the ITER grade, is presented by M. Yamagiwa et al. (2018). Fig. 6. and 7. illustrate the findings from scanning ion microscope (SIM), scanning electron microscope (SEM), and transmission electron microscope (TEM) analyses of tungsten of ITER grade, which reveal non-uniform bobble distribution in the sample's top view and cross section.

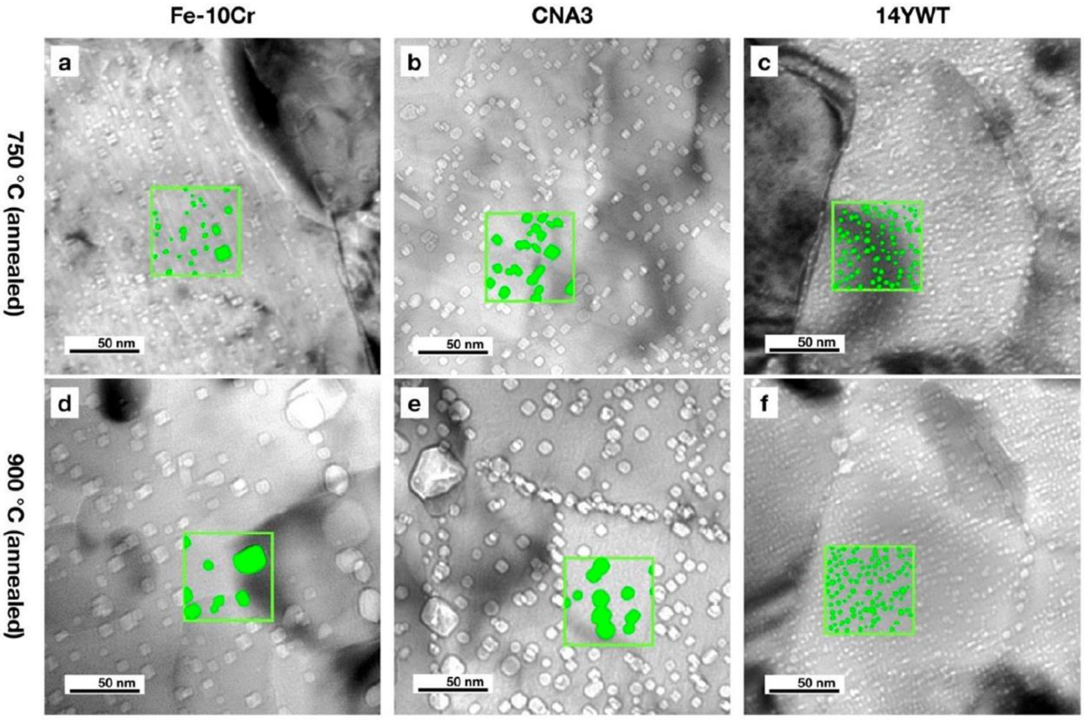

Fig. 8. Helium bubbles are produced on Fe-9/10Cr in (a), CNA3 in (b), and 14YWT in (c). The results for the sample with helium implanted at 600 and 900 degrees Celsius are shown in the top and bottom panels, respectively. [13].

Y.R. Lin et al. (2021) [11] used a scanning transmission electron microscope (STEM) and TEM to study the genesis of helium bubbles. Fig. 8. from the data demonstrates how the number of nanoparticles in the sample affected the size of the helium bubbles. In comparison to CNA3 nanostructure ferritic alloy (which has a medium number of nanoparticles), the 14YWT sample has the smallest helium bubble size, while Fe-9/10Cr (no nanoparticles) exhibits the biggest size. However, the densest bubble structure did develop on the 14YWT sample, followed by CNA3 and Fe-9/10Cr, at the same temperature and helium influence quantity.

3. Future of helium in fusion

The tendency toward reactor first walls that are increasingly likely to be metal, not just W, but maybe all-metal, has increased the importance of helium research. Helium causes the metal to react violently. It is crucial to research wall interaction and helium transport in the reactor scenario.

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