ELSEVIER

Journal of Nuclear Materials 271&272 (1999) 491–495

journal of nuclear materials

Evaluation of magnetic fields due to the ferromagnetic vacuum vessel and their influence on plasma discharge in tokamak devices

T. Nakayama [a,*], M. Abe [a], T. Tadokoro [a], M. Otsuka [b]

[a] *Power & Industrial Systems R&D Division, Hitachi, Ltd., 7-2-1 Ohmika-cho, Hitachi-shi, Ibaraki-ken 319-12, Japan*

[b] *Hitachi Works, Hitachi, Ltd., 1-1 Saiwai-cho, Hitachi-shi, Ibaraki-ken 317, Japan*

Abstract

We studied characteristics of the magnetic fields due to a ferromagnetic vacuum vessel (F-VV) experimentally and computationally to clarify whether plasma discharge is possible with the F-VV in tokamak devices. We made three kinds of evaluations using the Hitachi tokamak HT-2. One was a discharge test with error field coil. The second was a numerical analysis of the magnetic field induced by a ferritic first wall. The third was a discharge test with the ferritic first wall. Consequently, we confirmed that a normal plasma discharge could be obtained with a ferritic first wall in the HT-2. The strength of the localized magnetic field induced by the F-VV in the plasma region was smaller in tokamak devices with the size of the JFT-2M and ITER than in the HT-2. Therefore, the F-VV should be applicable to tokamak devices. © 1999 Elsevier Science B.V. All rights reserved.

1. Introduction

Stainless steels are used as structural components of existing tokamak devices. However, they are considered inappropriate for a fusion reactor vacuum vessel (VV) due to their high activation and swelling. Ferritic steels (e.g. F82H: 8%Cr–2%W–0.2%V–0.04%Ta–Fe [1]) are regarded as a better choice and a blanket case made of them was designed in the ITER R&D work [2]. Another advantage is ability to reduce toroidal magnetic field (TF) ripple. To study the influence of the F82H on the plasma in tokamak devices, the JFT-2M has plans to install F82H plates for TF ripple reduction and to replace the stainless steel VV (SS-VV) with ferritic steel VV (F-VV) [3]. The F82H starts to be saturated magnetically at an external magnetic field of 0.24 A/m. Its saturated magnetization is 1.96 T [4]. The magnetization curve used in this study was reported in Ref. [5].

Problems encountered when applying the F82H to the VV of tokamak devices are predicted to be generations of (1) unintentional magnetic fields, (2) magnetization force, and (3) impurities. The rigid support structure can eliminate problem (2). Problem (3) of impurities from the F82H was found not to degrade the vacuum quality in a tokamak device [6,7].

Then problem (1) of unintentional fields can be divided into two types of fields, a toroidally symmetric field and a localized magnetic field (LF). There are two generation paths. One is the increase of eddy current due to high permeability of the F82H. The other is leakage of magnetic flux from the F82H. Because this leakage is caused by discontinuity in the magnetic force line direction, the LF is generated around the port. The eddy current may delay the penetration time of the external poloidal magnetic field (PF). The LF may disturb plasma discharge [8]. In this paper, we discuss the influence of the magnetic fields on plasma discharge in the case of the Hitachi tokamak HT-2 [9] with F82H plates installed as a first wall.

---

[*] Corresponding author. Tel.: +81-294 53 3111/5214; fax: +81-294 53 4344; e-mail: nakayama@erl.hitachi.co.jp

0022-3115/99/$ – see front matter © 1999 Elsevier Science B.V. All rights reserved.
PII: S0022-3115(98)00806-X

2. Evaluation of magnetic fields due to the ferritic steel

2.1. Evaluation procedure

To clarify the influence of the magnetic fields on plasma discharge, we planned to install the F82H plates in the HT-2 and to attempt a plasma discharge test. Fig. 1 shows the poloidal cross-section of F82H plates in the HT-2. The VV is made of stainless steel (SS304). The F82H plates are divided into 12 parts along the toroidal direction, which is the same number as the number of TFCs. The gap between the plates is about 5 mm. The major radius of the torus surrounded by F82H plates is $R = 0.42$ m. There are two kinds of holes for the port. One is a round shape with 0.134 m diameter, the other is a rectangular shape with 120 mm (toroidal width) × 127 mm (vertical width) dimensions. Before installation of these plates, we evaluated the influence of magnetic fields on plasma discharge as follows:

1. First of all, we tested magnetic field response in the iron pipe to an externally applied magnetic fields, which simulated poloidal field response in the F-VV to the PF coil (PFC) current [5]. The measured poloidal field response was almost the same as the SS-VV case, because of magnetic saturation of the iron pipe. The results showed that PF controllability in the F-VV is expected to be almost identical to that in the SS-VV [5].
2. Then, we tested a plasma discharge with the error field coil (EFC) without F82H plates. Fig. 2 shows the arrangement of the EFC which generates the radial component of the LF (B_r^{EFC}) in the plasma region. A full width at half maximum of B_r^{EFC} distribution along the toroidal direction is 75° in toroidal angle. The discharge test was carried out with up to $B_r^{EFC} = 7.8$ mT at the center of the plasma region of $R = 0.41$ m. A start-up of plasma discharge was possible even with $B_r^{EFC} = 7.8$ mT at $R = 0.41$ m.
3. Finally, we computationally evaluated the strength of the LF induced by F82H plates in the HT-2, using the 3D non-linear magnetic field analysis code MAGFIC [10]. This is discussed in the next section in detail. Code accuracy when applied to the tokamak device was verified through comparison with the measured magnetic fields around the plasma region induced by F82H plates in the HT-2 [5].

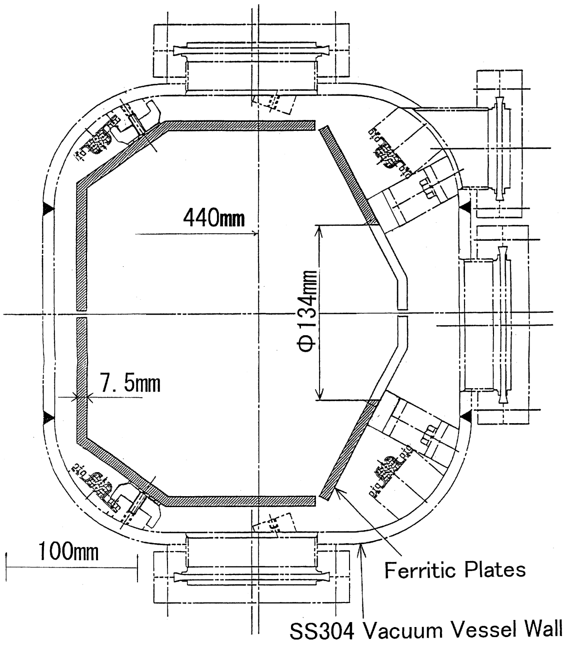

Fig. 1. The poloidal cross-section of F82H plates in the HT-2. The shaded parts are F82H plates.

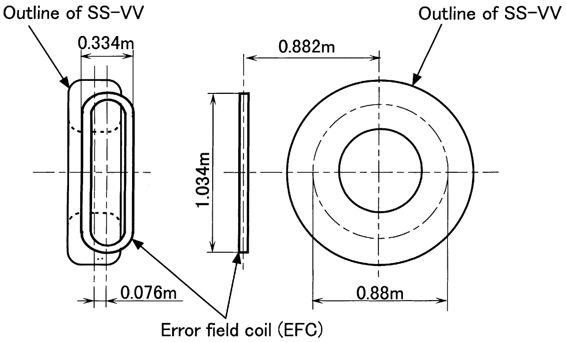

Fig. 2. The arrangement of the EFC in the HT-2. The EFC is installed outside the SS-VV of the HT-2 without F82H plates.

2.2. Evaluation of the LF induced by the F82H plates around the plasma region

The distribution of B_r^{F82H} around the plasma region was computationally evaluated in detail. Fig. 3(a) shows the computational model of F82H plates with a rectangular port, and the direction of the LF induced by the F82H plates. The region enclosed by the bold lines is the displayed region for Fig. 3(b), (c). The strength of external TF (B_t^{ext}) was 1.8 T at $R = 0.41$ m. This LF was induced by the port and the gap between the F82H plates. The LF due to the gap could be reduced by narrowing it. Fig. 3(b), (c) show the radial components of LF distributions induced by F82H plates and TFC, respectively. The B_r^{F82H} was about 2 mT at $R = 0.40$ m. The strength of the radial LF component (B_r^{TFC}) induced by TFC was about 2 mT at this same radius. The B_r^{F82H} was in the opposite direction of B_r^{TFC} and their absolute strength was the same order. Therefore, B_r^{TFC} in the plasma region could be reduced by the F82H plates. The major radius of the plasma region center (R_{ax}) is usually kept at $0.40\ \text{m} \leqslant R_{ax} \leqslant 0.44$ m in the HT-2. The maximum strength of B_r^{F82H} is 5 mT at $R = 0.44$ m. This value

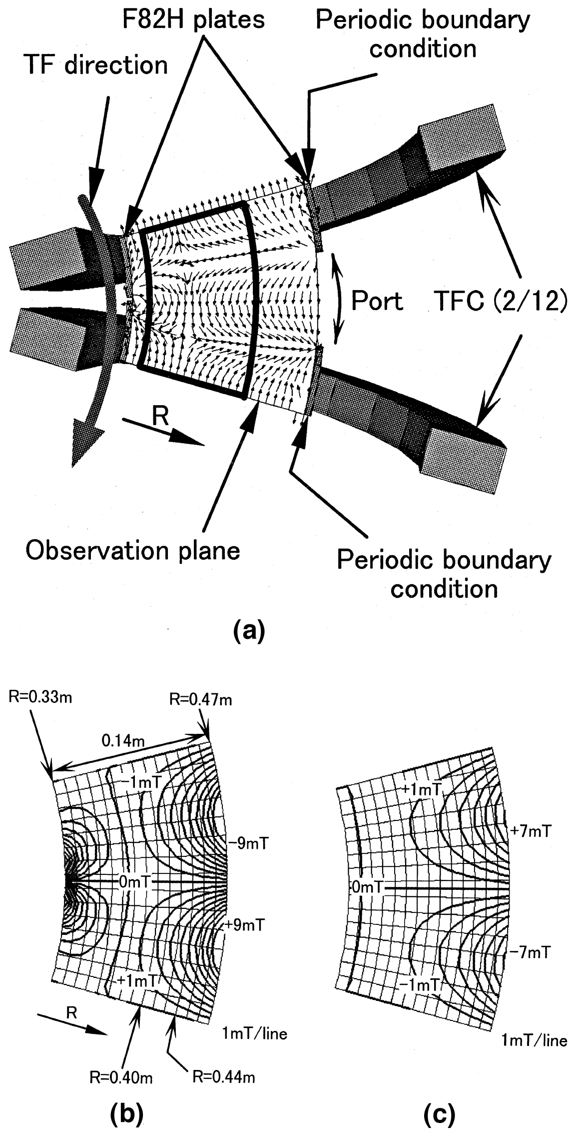

Fig. 3. The distribution of magnetic field in the plasma region on the midplane. (a) Cross-section on the midplane of the model. The direction of the field induced by the F82H plates is shown on the observation plane. The region enclosed by the bold lines is the displayed region for (b), (c). (b), (c) the radial field distributions induced by the F82H plates and the TFC, respectively.

is less than $B_r^{EFC} = 7.8$ mT. Therefore, plasma discharge should be possible with F82H plates in the HT-2.

2.3. Reduction technique of the LF induced by the F82H plates around the plasma region

We proposed a reduction technique of the LF induced by F82H plates by adjusting the F82H port neck size and shape. Using this technique, it is possible to adjust the TF ripple. The basic idea is that magnetic force line in the F82H plates pass through the F82H port neck without passing through the plasma region. Fig. 4(a) shows the upper half of the computational model of the F82H plates for MAGFIC. We calculated the poloidal component of the LF (B_p^{F82H}) induced by the F82H plates on the torus surface (observation surface). Fig. 4(b) shows the distribution of B_p^{F82H} on the surface for three different port neck thicknesses (t_p). The B_p^{F82H} was the weakest when t_p was about 15 mm, which is the thickness needed to conserve the toroidal magnetic flux in the F82H plates.

3. Discharge test and extrapolation to large tokamak devices

3.1. Plasma discharge test with the F82H plates

We made a plasma discharge test with the F82H plates in the HT-2 [7]. In this test, the F82H port neck was not used. Fig. 5 shows the reconstructed PF distribution of typical results of plasma discharge at the flattop of plasma current, after 1500 discharge cleanings. This distribution is drawn based on measured magnetic field data and considering magnetization of F82H plates [7]. A normal plasma discharge was obtained with the same control systems as those without F82H plates. The plasma current was 15 kA at flattop, and the discharge duration was 35 min. Plasma was controlled almost at the center of the F82H plates cross-section as pre-programmed. We concluded that there is no influence from the F82H plates on plasma control.

3.2. Extrapolation to other tokamak devices

We attempted to extrapolate the results about the LF in the HT-2 to other tokamak devices such as JFT-2M and ITER, when replacing the SS-VV with the F-VV made of F82H. The strength of the LF induced by the F-VV can be discussed using a similarity law. The maximum strength of the LF induced by the F-VV ($B_{r,max}^{F-VV}$) at the R_{ax} can be written as [5]

$$B_{r,max}^{F-VV} = \frac{\mu_0}{4\pi} M_s \left(\left(\left(\frac{w}{d} \right)^2 + 1 \right)^{-3/2} - 1 \right) \frac{t}{d} \frac{h}{d}, \qquad (1)$$

where M_s is the saturated magnetization, w the toroidal port width, h the vertical port width, d the difference between the major radius of the port position and the reference position and t is the thickness of the F-VV. Eq. (1) indicates that the field induced by the F-VV at the center of the discharge region is the same order in tokamak devices, because each ratio (w/d, h/d and t/d) has roughly the same value, where d is regarded as the

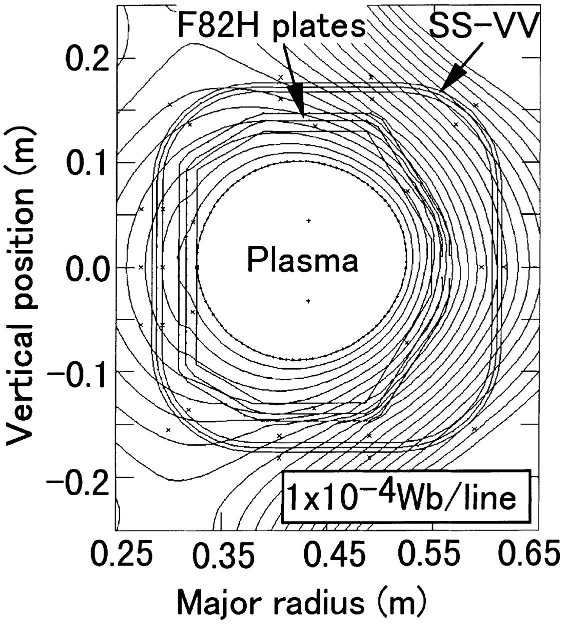

Fig. 5. Reconstructed PF distribution of typical experimental results of plasma discharge at the flat-top of the plasma current with F82H plates in the HT-2.

minor radius of the F-VV. Table 1 shows the size of the F82H plates in the HT-2 and the VV in the JFT-2M [11] and ITER [12], and $B_{\mathrm{r,max}}^{\mathrm{F\text{-}VV}}$ extrapolated from the results in the HT-2. Minor radius is defined as half the horizontal width of the VV. We expect plasma discharge is able to occur in the JFT-2M and ITER because the $B_{\mathrm{r}}^{\mathrm{F82H}}$ is less than that in the HT-2.

4. Conclusion

We studied characteristics of the magnetic fields induced by the F-VV experimentally and computationally, in order to clarify the possibility of plasma discharge with the F-VV in tokamak devices. Poloidal field controllability in the F-VV was found to be almost identical to that in the SS-VV. The maximum strength of the LF at the center of the discharge region induced by F82H plates (5 mT) was less than the allowable strength of the field to discharge plasma in the HT-2. The radial component of the LF in the plasma region due to TFC could be reduced by the F-VV. The radial component of the

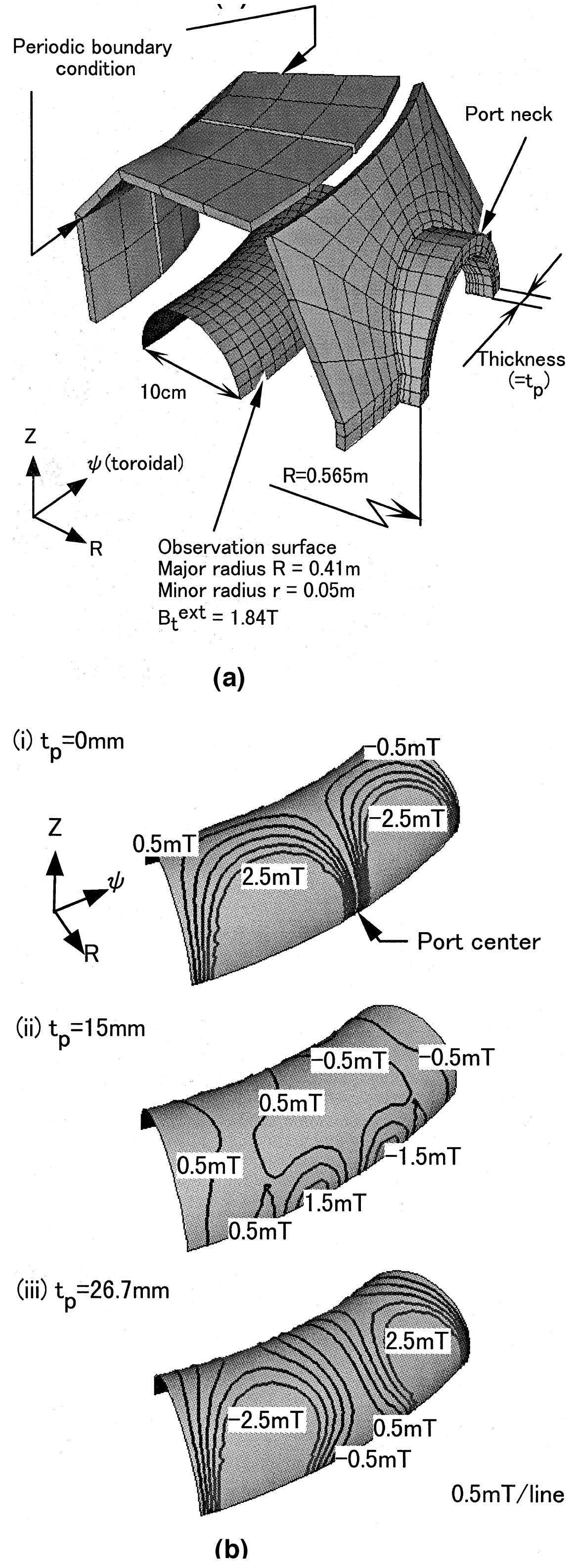

Fig. 4. Evaluation of the reduction technique for the LF. t_{p} is the thickness of port neck. (a) Evaluation model with the observation surface. (b) The distribution of B_p^{F82H} on the observation surface in the HT-2 with the F82H port neck.

Table 1
The size of the VV and the maximum strength of the LF extrapolated from the results in the HT-2 at the center of the discharge region induced by F-VV

Model	Minor radius d (m)	Toroidal port width w (m)	Vertical port width h (m)	Thickness t (m)	w/d	h/d	t/d	$B^{F-VV}_{\mathrm{r,max}}$ at R_{ax} (mT)
F-VV (JFT-2M) [11]	0.42	0.40	0.48	0.026	0.95	1.1	0.062	5.0
F-VV (ITER) [12]	4.4	1.8	3.0	0.10	0.41	0.68	0.023	0.4
F82H plates (HT-2)	0.12	0.12	0.13	0.0075	1.0	1.1	0.063	5.0

LF induced by the port could be reduced by the F82H port neck. A normal plasma discharge could be obtained with F82H plates in the HT-2. The strength of the LF induced by the F-VV at the center of the discharge region was smaller in tokamak devices with the size of the JFT-2M and ITER than in the HT-2. Plasma discharge should be possible with the F-VV in tokamak devices.

Acknowledgements

The authors gratefully acknowledge informative discussions with members of the JFT-2M group, and they thank Drs. A. Hishinuma and K. Shiba for their advice on steels' properties of the ferritic material F82H.

References

[1] N. Yamanouchi, M. Tamura, H. Hayakawa, A. Hishinuma, T. Kondo, J. Nucl. Mater. 191–194 (1992) 822.

[2] Y. Gohar, ITER Joint Central Team, ITER Home teams and participants of the test blanket working group, Proc. 16th Int. Conf. Plasma Phys. Cont. Nucl. Fusion Res., Montreal, 1996, IAEA-CN-64/FP-10.

[3] M. Sato, Y. Miura, S. Takeji, J. Nucl. Mater. 258–263 (1998) 1253.

[4] K. Shiba, A. Hishinuma, A. Tohyama, K. Masamura, JAERI-Tech 97-038, 1997 (in Japanese).

[5] T. Nakayama, M. Abe, T. Tadokoro, Y. Miura, N. Suzuki, M. Sato, S. Sengoku, J. Plasma Fusion Res. (in Japanese) 74 (1998) 274.

[6] K. Odaka, O. Sato, M. Otsuka, T. Abe, S. Hara, H. Takatsu, M. Enoeda, J. Plasma Fusion Res. (in Japanese) 73 (1997) 1001.

[7] M. Abe, T. Nakayama, Proc. 17th Symp. on Fusion Eng., California, 1997, P-14,3.

[8] B. Lloyd, G.L. Jackson, T.S. Taylor, E.A. Lazaurs, T.C. Luce, R. Prater, Nucl. Fusion 331 (1991) 2031.

[9] M. Abe, K. Takeuchi, Fusion Technol. 29 (1996) 277.

[10] M. Koizumi, Y. Higuchi, IEEE Trans. Magn. 31 (1995) 1516.

[11] T. Shoji, et al., JAERI-M 83-194, 1983 (in Japanese).

[12] ITER Design Description Document, 1.5 Vacuum Vessel, ITER, 1996.