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Next Generation Toroidal Devices

S. Yoshikawa

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NEXT GENERATION TOROIDAL DEVICES

Shoichi Yoshikawa
Princeton Plasma Physics Laboratory
Princeton University

Abstract

A general survey of the possible approach for the next generation toroidal devices was made. Either surprisingly or obviously (depending on one's view), the technical constraints along with the scientific considerations lead to a fairly limited set of systems for the most favorable approach for the next generation devices. Specifically if the magnetic field strength of 5 T or above is to be created by superconducting coils, it imposes minimum in the aspect ratio for the tokamak which is slightly higher than contemplated now for INTER design. The similar technical constraints make the minimum linear size of a stellarator large. Scientifically, it is indicated that a tokamak of 1.5 times in the linear dimension should be able to produce economically, especially if a hybrid reactor is allowed. For the next stellarator, it is strongly suggested that some kind of helical axis is necessary both for the (almost) absolute confinement of high energy particles and high stability and equilibrium beta limits. The author still favors a heliac most. Although it may not have been clearly stated in the main text, the stability afforded by the shearless layer may be exploited fully in a stellarator.

Keywords: Toroidal plasma confinements, tokamak, stellarator, heliac, toroidal stability, toroidal equilibrium, super conducting coils.

1. INTRODUCTION

Design of the next generation of a toroidal confinement device is one of the most crucial aspects of the fusion science. Here, we would like to discuss the next generation toroidal device whose minimum requirement is that the device should have high field, superconducting coils. By nature, it should be a large scale experiment comparable to or larger than the major toroidal experimental systems of today. For a tokamak, the scientific and engineering target should be the ignition or Q about 5 or more. For a stellarator, the target should be a type comparable to the present major tokamaks such as JT 60 upgrade. Obviously the experiments must be carried out in a large institute such as JAERI, NIFS etc.

With the present knowledge, we would like to examine the design optimization factors as well as possible. It is certainly true that we cannot precisely predict the outcome of a large scale experiment which ultimately would cost several billion 1997 dollars. However, we should be able to assess the probability of the success of the experiment. We must note that the large scale experiment for fusion science is rather unique in that the underlying scientific laws are not completely known. Thus, unlike a high-energy particle accelerator, the design choice is not determined by the technological input alone.

Before we go into the details, readers should be reminded that there is a third path which was favored by some groups. The third path is to investigate a stellarator and tokamak hybrid concept. This concept is to add the stellarator field (either straight or helical axis) to a tokamak. As we shall show later that the hybrid concept is also excellent. Perhaps an ultimate fusion reactor might eventually have a helical winding for a tokamak, or equivalently, a net toroidal current for a stellarator.

The following parts of this short paper are divided into four. Section 2 will be devoted to the next generation stellarator. Section 3 will be for the next generation tokamak. Section 4 will be devoted to stellarator and tokamak hybrids and other asymmetrical tokamaks.

2. NEXT STELLARATOR

A stellarator, compared with a tokamak, has some theoretical drawbacks as well as technological difficulties arising from the fact that the stellarator system tends to be harder to construct and to maintain as a fusion reactor. Recent findings, however, indicate that these drawbacks and the technological difficulties might have been exaggerated in view of the discoveries of theoretically better stellarator configurations and the narrowing of technological difficulties between a tokamak and a stellarator. We shall look into the new configurations and indicate a path to lead to better designs of stellarator based reactors. [1-5]

2a) Magnetic configuration of low beta stellarator.

The vacuum field of the stellarator configuration may be expressed in terms of a magnetic potential which satisfies $\nabla^2\Omega = 0$. Arbitrary magnetic function is expressed as superpositions of the toroidal potential functions.[6-8] Therefore, all the magnetic configurations can be tabulated.

There are two types of the magnetic configurations. This classification comes from the comparison with their equivalents in the straight stellarator configuration. (i.e. in the limit of the infinite aspect ratio.). One has a straight magnetic axis and another has a helical magnetic axis.

It must be noted that, if there is a helical symmetry, in a straight stellarator configuration, both classes possess the absolute single particle confinement property. However, here we extend the stellarator definition to include the helically asymmetric stellarator. For example, the magnetic field configuration of the following form is included. The configuration may be understood in terms of a straight magnetic field and the helical field whose axis's do not coincide.

$$\mathbf{B} = \sum_{l} \nabla\Omega\,(\,kr, l\theta + kz\,) + \mathbf{B}_0 \qquad (1)$$

$$\mathbf{B}_0 = (\,B_x, B_y, B_z\,) = (\,0,\ B_y, B_z\,) = \text{const.} \qquad (2)$$

For convenience, in an achievable toroidal stellarator (i.e. the aspect ratio, A, is finite.), we call the two classes as straight-axis stellarator and helical-axis stellarator respectively. The former has a circular magnetic axis.

2b) Straight-axis stellarator

First we deal with a straight-axis stellarator (finite A) which has a helical symmetry in corresponding straight stellarator. For this, we know relatively well. There is no absolute confinement of a single particle. The maximum equilibrium plasma pressure is less than the quantity proportional to square of the ratio of the rotational transform to A. In a straight stellarator (i.e. A is infinite) of straight-axis type, maximum stability beta is at most 2 to 3 %: this number is less than that of the equilibrium beta.[9] In a straight-axis stellarator (i.e. A is finite), the stability is provided by the toroidal curvature just as in a tokamak. Unfortunately a straight-axis stellarator less than approximately 3 with a rotational transform comparable to a tokamak cannot be designed.[8] The equilibrium beta limit becomes less than the stability limit around A = 4 to 6 depending upon how we choose the coefficients of toroidal potential functions. It follows the optimum straight-axis stellarator has A around 6, as the lower A system is harder to construct. It is not surprising then that the superconducting stellarator of NIFS (which is essentially the straight-axis stellarator) has the present aspect ratio of 6-8. For the second type, that is a straight-axis stellarator without the corresponding helical symmetry in a straight stellarator is not known at present. However, it is possible in future, that a stellarator whose axis is arbitrarily close to a circular axis (in toroidal geometry) may be found to be superior.

2c) Helical-axis stellarator; heliac

The most simple helical-axis stellarator is heliac.[2] The simplest of the heliac group has a center straight conductor at the cylindrical axis in corresponding straight stellarator. With the outside helical conductor ($l = 1$ is often used.), the closed magnetic surfaces encircle the central conductor.. The center of the magnetic surfaces is a line, that is the helical magnetic axis. The central conductor needs not be a straight conductor located at the center. Indeed any helical conductors near the cylindrical axis in straight stellarator (which let us call inside conductors) along with the outside helical conductors form the magnetic surfaces. The whole system is of course can be made helically symmetric in straight stellarator configurations.

This heliac configuration in a straight stellarator (i.e. infinite A) has an inherent advantage over the straight-axis stellarator. The rotational transform and shear per pitch of the heliac are higher than those of an ordinary stellarator; this improves equilibrium and stability. The probably more important advantage lies in the fact that the heliac configuration has a magnetic well. Therefore the MHD stability of a straight stellarator comes both from the well and the shear. Also the plasma stability improves (although at the cost of the narrowing of the plasma radius) with the plasma pressure, as the radius of the magnetic axis increases with the pressure just like a tokamak. Hence the stability beta of a straight heliac is in the order of tens of percent.

A toroidal heliac of a large aspect ratio (between 10 to 20) can have in principle have a stability and equilibrium (averaged) beta of more than 5 to 10 %. The system then should be suitable for a fusion reactor. Also because of the large aspect ratio, a helically symmetric heliac turned into a torus can have a better single particle confinement. It follows then that a heliac could be able to compete with a tokamak as a fusion reactor. As we shall see later, a low aspect ratio tokamak (which has beta higher than 10 %) is technically difficult to make.

2d) Helical axis stellarator; improved heliac

Actually a standard or classical heliac can be improved in two ways. First, if a vertical field is added to a heliac, the magnetic axis of a heliac are pushed towards the increasing toroidal radius in relation to the outermost closed magnetic surface. The equivalent of the straight heliac is that of Eqs. (1) and (2). In this case, the magnetic field lines stay longer in the favorable curvature regime where the toroidal field gradient is good for the plasma stability than the unfavorable regime. The optimum magnetic field must be determined balancing the increase in the plasma beta vs. the decreases in the single particle confinement property and the radius of the plasma radius.

Second, as shown in the author's paper in Nuclear Fusion[8], the outside helical coil can be replaced by a set of (vertically aligned) toroidal coils. These coils can even be placed in the same horizontal plane with alternating radial position. In a fusion reactor, the access to the plasma and its surroundings (vacuum vessel and blanket) must be facilitated as much as possible. Discrete toroidal coils will make the easy access and simple fabrication of toroidal coils. Obviously the optimum design should be attempted in future, when more scientific and technical facts become known.

2e) Stellarator with superconducting coils

Perhaps it is better to digress into discussing the design of a high field superconducting stellarator here.

A superconducting coil of presently available material has the heat shield, size of which is approximately 15 cm. In addition, a future fusion reactor coil must have a neutron shield. Because there is an upper limit on the current density of the (stabilized) superconducting wire, the width of the coil must be proportional to the magnetic field strength. For the magnetic field strength of 3 T or more,. the total average distance from the center of the coil to the plasma surface cannot be less than 50 to 60 cm. This consideration makes a small-size, high-field stellarator construction difficult. As was pointed previously, the size of LHD is therefore perhaps as small as can be made for a superconducting stellarator.

Furthermore, if we are to require that the coils are to be made demountable like those of the future German stellarator[5], the distance from the magnetic axis of the plasma to the center of the coil must be increased over that of the helical winding coils of LHD. Then the minimum major radius becomes approximately 5 to 6 m. We shall show that the similar consideration is important for the design of the next generation tokamak experiment (INTOR type).

2f) Helias

If, in a straight stellarator, the magnetic function is a combination of $f(kz + \theta)$ and $g(kz + 2\theta)$, along with the constant B field in the z direction, the magnetic surfaces are no longer helically symmetric. However, it has a mirror symmetry, say, along the x axis. The magnetic axis is helical. The shape of the magnetic surface is a function of theta of the cylindrical coordinates. The absolute magnitude of the magnetic field is no longer helically symmetric. We can choose the coordinate such that the shape is round at $\theta = \pi$ and crescent like as in a heliac at $\theta = 0$. In another words, the shape is round for the negative x and crescent like for the positive x. We will identify the positive x-axis with the positive (major) radial axis of a torus.

Now as this system is made into a finite aspect ratio stellarator, the system is called a Helias. Because in the equivalent, straight stellarator, the magnetic field strength is not helically symmetric, the total field strength which includes the variation of the toroidal field strength due to the change in the major radius can be made almost constant on a magnetic surface. If the magnetic surface coincides with the constant B surface, the magnetic field gradient drifts of charged particles are restrained on the magnetic surface: hence no single particle loss. This is the principle of Helias. Obviously, it is impossible to design an ideal Helias. However, Nuehrenberg and his group have found a configuration which cuts down the single particle loss significantly.[10] That is the basis for the next generation German stellarator, W 7-X. Their configurational studies show that the high energy particles can be contained over the ion ion collision time commonly found in high temperature plasmas. This effectively makes the single particle loss problem in a stellarator disappear.

As for the MHD plasma pressure limit, the equilibrium pressure limit is high. The stability limit is somewhat less than that of a Heliac; but as it has a helical axis, the pressure limit is higher than that of a straight or circular axis stellarator.

2g) Other helical axis stellarator designs.

Once the constraint of a straight or circular magnetic axis is removed, several alternative designs are possible. Several of those designs were reported at 1977 Toki Conference. Among others, Zarnstoff (PPPL)[11], Garabedian (NYU)[12], Okamura (NIFS)[13] have proposed new configurations. It must be pointed out that those configurations that emphasize low aspect ratio are bound to have difficulties in future, when a superconducting device is to be built. Then the space limitation probably make the construction of the device impractical.

2h) Considerations of the other energy loss mechanism

For the next generation experiments such as LHD, there are two energy loss mechanisms which we must concern ourselves. As is well known, one is the charge exchange loss of ion energy. The another is anomalous energy loss of electrons, if the experience of tokamaks can be applied to stellarators. The new configurations discussed above should not be so much different from an ordinary stellarators. If the helical winding pitch is optimum, microinstabilities resulting from the ballooning type instabilities should not be neither greater or smaller than those of a tokamak. It is of interesting to see whether there is a difference in the energy transport due to i) electric field and ii) change in the magnitude and the sign of the magnetic shear.

3. NEXT GENERATION TOKAMAK

Works done about the tokamaks in these 40 years have shown beyond doubt that the scientific break even condition for the tokamak fusion plasmas was attained. The empirical scaling law established by experiments appears to show that ignition tokamak plasmas are possible, if we are to aim solely to design a tokamak for achieving the scientific demonstration of self ignition. That type of the experiments would utilize high field copper coils with minimum of neutron shield for the structure. The cost and time needed for that project, however, appear to dissuade people from embarking on the experiment. Instead, a superconducting coil tokamak with the capability of being converted into the engineering prototype is deemed advisable. For that purpose, the ITER project was started in the middle of 1980's. Unfortunately, the advance of the project is slowed by the political and financial realities. The slow down, however, can be partly attributed to the technological problems. The technological design that simultaneously satisfies both scientific and engineering needs is difficult. In a sentence, it is the conflict between the scientists' desire to have a low aspect ratio tokamak and the engineers' desire to have as large as space possible to design the system around the main plasmas We shall review first the facts about tokamak designs. Then we shall discuss the possible improvements which were presented previously but not widely known.

3a) Fundamental design constraints of a tokamak

Tokamak designs are scientifically constrained by the need for having a toroidal plasma in MHD equilibrium and (MHD) stability. (a) Since the rotational transform is provided by the plasma current, the inverse of the rotational transform, q, must be greater than 1. (b) The positional stability requires either a metal conductor surrounding the plasma or the vertical magnetic field with the suitable curvature. (c) For high interchange stability beta limit, the low aspect ratio is necessary. Also elongation in the vertical direction is called for. (d) As is well known, the elongation jeopardizes the positional stability: hence the feedback stabilization against the vertical positional instability becomes necessary. (e) The divertor is needed to dispose the energy flow from the plasma core to the periphery. These requirements mean that the plasma aspect ratio must be low, preferably around 3 to 3.5. The aspect ratio is defined as the major diameter of the toroidal plasma divided by the horizontal plasma minor width.

Engineering design criteria require just the opposite. The space need requires a higher aspect ratio. First, let us determine the major radius of a toroidal plasma. If the toroidal magnetic field at the inner side of the plasma to be 20 % over the average magnetic field of 5 T, then at the inner surface the field is 6 T. The distance from the plasma to surface of the neutron shield (within, there are the scrape off layer, vacuum vessel wall and possibly poloidal coils) is to be .35 m, the thickness of the neutron shield is to be .2 m, and the thickness of the thermal insulation along with the structure support of the toroidal coil is to be .2 m. The the total distance between the plasma surface and the coil surface is then .75 m. If the maximum field strength of the toroidal field coil is to be 8 T, the aspect ratio, A, is calculated from

$$A = 1 + 3 / a \qquad (3)$$

where a is the half the plasma horizontal width, w/2, in meters. The above equation essentially determines the size, a, of a tokamak, if the aspect ratio is determined from the scientific consideration. Once the plasma size is established, the ignition condition automatically determines the plasma power output. Thus the experimental cost is roughly estimated.

The above formula gives the minimum aspect ratio for a given minor radius, a. Other considerations, such as remote handling, the deterioration of the vacuum wall under the neutron bombardment, access to the heating equipments and so forth make the engineering very hard, if the space is at premium. In principle, by increasing the cost of the R & D projects and the fabrication cost, these problems could be solved. However, we definitely end up a very complicated experimental apparatus. Then the reliability becomes in doubt. Often, the major breakdown during the experiment could be fixed. But in reality, the major breakdown is fixed with a large amount of expense and the loss of time. Eventually, the question of starting a project is resolved by the will of people.

Could there be any scientific or technical fix to improve the success of the experiment without substantially increasing the cost? The answer cannot be given too easily. We know a number of occasions where large experimental endeavors or satellite launches have failed, because of the supposed improvements are turned out to cause the unexpected failures. It is usually safe to assume that any possible improvement has actually, in close examination, defects which are worse than the cures. Nonetheless, in the case of the next generation tokamak, some minor change in the design may actually improve the probability. We shall look into those in the following.

3b) Open tokamak

If we increase the major radius of the return legs of the toroidal coil, the access to the tokamak is improved. Also under that circumstance, the number of the toroidal coils can be reduced, further facilitating the access. If the radius becomes large enough, the toroidal field can be provided by a single coil. Obviously, the cost of the toroidal coil increases and the cost for the failure of the coil will become huge. However, in view of the difficulty encountered for the designing a tokamak system around a conventional toroidal coil system, this new type of the design needs the attention. Let us call this system as an "open tokamak".

The idea is not new. This concept was discussed at the start of the JT-60 design in 1970's. The concept of OCLATOR (one coil low aspect tokamak reactor) is one of them. For OCLATOR, a toroidal coil system is replaced by a single coil of a large radius.[14]

In an open tokamak, the toroidal field correction coils are necessary. However, in view of the recent advance in the study of particle behavior in asymmetric field, the requirement of the correction coil system may be less stringent than previously expected.

As for the reactor, a single coil tokamak reactor has its own merit. An almost circular large toroidal coil (say radius of 30 to 50 meters.) could accommodate many tokamak systems. Hence the cost of construction and maintenance of the toroidal coil is divided by many power generating tokamaks. As we know, the startup power and heating system requires a heavy cost as well, concentrating several tokamaks in one place makes an economic use of these.

3c) Addition of a weak helical field for positional stability

As is noted, the positional stability is important for a design of a large tokamak. The stability can be increased by applying a relatively small amount of externally applied rotational transform. For this purpose, it is advisable to use $l = 2$ helical windings. The externally applied rotational transform is proportional to the current rotational transform at the plasma surface (that is $1/q$) and inversely proportional to the square of the aspect ratio for a circular cross section tokamak. The proportionality constant is of the order of unity. (This fact was recognized in 1960's in connection with the C stellarator experiments.)[15]

For a modern large radius tokamak, the externally applied rotational transform is estimated to be 15 to 20 degrees for doubling the positional stability. The helical winding to generate this could be fitted without too much design problem, although the design is not straight forward, especially the tokamak cross section is not circular. Presumably the helical windings can be replaced by the series of alternating quadrupole coils similar to those in particle accelerators: it is not known at present which is preferable.

Both open tokamak and addition of external rotational transform are relatively minor modifications of the scientific tokamak properties, Because of the recent developments in the improved stellarator designs, we should re-examine the tokamak stellarator hybrid. That will be done in the next section.

4. TOKAMAK STELLARATOR HYBRID

Both tokamak and stellarator could become a reactor, especially if we can breed a slight amount of fissile material in the reactor. As explained in the previous sections, however, to build a prototype reactor is expensive; for a tokamak reactor, minor radius is large and for a stellarator reactor, major radius is large.

Would the combination of good parts of tokamak and stellarator make the R and D cost for the prototype reactor less expensive? New ideas can be found easily and indeed several have been proposed. We must be, however, very careful about the new ideas, as there are many requirements for a new concept to be useful for a reactor.

A lower aspect ratio tokamak ($A < 3$) appears to be not a good candidate for a prototype. It is an excellent research device, for the cost of the proof of principle experiment should be approximately proportional to the square of the aspect ratio. However for a reactor, the constraint for the superconducting coils would make it impractical, even if the plasma stability pressure limit can be increased.

Likewise a very large aspect ratio stellarator tokamak (say $A > 15$) is not economical, as a helical-axis stellarator should be as effective.

Thus the aspect ratio of a stellarator tokamak must be between approximately 3 to 14. Within this category, we can see two types of stellarator tokamaks. One category is a tokamak with a helical winding to be discussed in section 4a. Another one is an asymmetric tokamak to be discussed in section 4b.

4a) Tokamak with a helical winding.

One of natural advantages of a tokamak is its ability to heat the plasma by its ohmic heating. Clearly a stellarator with a toroidal current, or equivalently a tokamak with a moderate to high external rotational transform has an advantage over a simple stellarator. The startup of a large radius stellarator without ohmic heating would probably be found more difficult than presently assumed.[16]

Secondly a tokamak without the external rotational transform is subject to the current disruption. In principle, if the vacuum chamber wall is designed properly, one or two major current disruption in the life of the vacuum chamber wall should not matter. A moderate or high external rotational transform even at the cost of increase of the aspect ratio (not necessarily reducing the critical MHD stability, if the design of the helical winding is right.) would stabilize the plasma against the major current disruption. The reason is the same as explained in section 3c; the positional stability is improved to the point that the plasma will not move easily across the magnetic field.[15]

It is possible that more elegant improvements can be found in future, if a combination of a helical axis configuration and a toroidal current is tried. However, there is always the fear that the complication arising the combination may cancel the advantage.

The toroidal current is generated spontaneously by the bootstrap current. The bootstrap current is augmented by the linked transformer. One intriguing possibility is to (in time) modulate the primary current. The plasma heating may be more efficient, due to the skin effect. Without the helical winding, the toroidal equilibrium would be locally destroyed, if the primary current is modulated.

4b) Asymmetrical tokamak

The toroidal coils cannot be placed in a perfect azimuthally symmetry, because the auxiliary heating is needed for the ignition. We know that the asymmetry, thus introduced, leads to the trapping of the high energy particles ending in the escape of the energy. The individual toroidal coil can be arranged helically to produce a weak rotational transform.

Admittedly there is a complication involved in arranging toroidal coils helically. However, it is possible that a helical axis stellarator field configuration which reduces the high energy particle loss is found advantageous enough to use for the future tokamak reactor. (See ref. 2)

Conclusions

The above covers the conventional medium beta fusion devices. It is probable that a reader finds this paper unsatisfactory as it does not contain an "advanced" configuration that he or she favors. The writer strongly warns about those advanced configurations, because they lack one of several conditions which is needed for it to be useful for a fusion device. Simply because a system is stable against the interchange instability is no guarantee that the plasma is stable against most devastating type of microinstabilities.

There is one exception, which may be worth for consideration. We know that starting from the works of Kimitaka and Sanae Itohs, the radial electric field influences the confinement.
The $E \times B$ rotation of the plasma apparently provides the differential rotational speed, which acts just like magnetic shear. (See the work of Dr. Yasuaki Kishimoto of JAERI).[17] Unfortunately, left to its own device, the potential difference between the surface and the center of the plasma cannot exceed the plasma temperature. (Except after the ignition, then the potential difference could theoretically become a fraction of fusion product energy.).

Now, we do not know whether there exists the anomalous conductivity across the magnetic field in a tokamak plasma.[18] Suppose that it does not. Then it follows that the classical conductivity (arising from the classical viscosity) is calculated to be very small. Thus injecting partially ionized high energy ions (say oxygen or carbon ions), we may be able to establish large potential difference between center and surface. This means that the large shear in the rotational velocity could be created. Adding this to a stellarator or a tokamak may improve the plasma confinement. If the potential difference can be changed in this way with minimum disturbance to the overall plasma characteristics, we can use this system to evaluate the effect of the shear to the plasma confinement.

It is hard to select a right approach to the fusion experiment. Because it affects many people's career as well as a large amount of resources, to design a right machine should be utmost importance beyond things such as institutional and personal preferences.[5]

Aknowledgement

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References

[1] For general descriptions: see K. Miyamoto, *Plasma Physics and Nuclear Fusion* (M.I.T. Press, Cambridge, Mass, 1987)

[2] S.Yoshikawa; *Nuclear Fusion* **23** 667 (1983)

[3] J.Harris and the H-1 Team; *Journal of Plasma and Fusion Research SERIES* **vol. 1** (Proceedings of Toki conference on stellarators, ISC - 11), 37 (1998)

[4] C.Alejaldre and the TJ-II Team; *J. Plasma Fusion Reseach SERIES* **vol. 1** 34 (1998)

[5] G.Grieger and the W-7-X Team; *J. Plasma Fusion Reseach SERIES* **vol. 1** 53 (1998)

[6] H.Bateman; *Partial Differential Equations of Mathematical Physics*, Chapter X (Cambridge University Press, UK, 1959)

[7] K.Miyamoto; *Nucl. Fusion* **18** 243 (1978)

[8] S.Yoshikawa; *PPPL-2038* (Princeton Plasma Physics Laboratory, 1983)

[9] S.Yoshikawa; *PPPL-1748* (Princeton Plasma Physics Laboratory, 1981)

[10] S.Gori, C.Nuehrenberg, J.Nuehrenberg, and R.Zille; *J. Plasma Fusion Reseach SERIES* **vol. 1** 62 (1998)

[11] A.Reiman et al.; *J. Plasma Fusion Reseach SERIES*, **vol. 1** 429 (1998)

[12] P.R.Garabedian; *J. Plasma Fusion Reseach SERIES* **vol. 1** 148 (1998)

[13] S.Okamura et. al.; *J. Plasma Fusion Reseach SERIES* **vol. 1** 164 (1998)

[14] S.Yoshikawa; *J.Fusion Energy* **1** 143 (1981)

[15] W.L.Harries, S.Yoshikawa, and R.M.Sinclair; *Phys. Fluids* **6** 1591 (1963)

[16] S.Yoshikawa; *Nucl. Fusion* **13** 433 (1973)

[17] Y. Kishimoto (JAERI), private communication

[18] M.Okabayashi and S.Yoshikawa; *Phys. Rev. Letters* **29** 1725 (1972)

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NIFS-547 R. Kanno, N. Nakajima and M. Okamoto,
Electron Heat Transport in a Self-Similar Structure of Magnetic Islands; Apr. 1998

NIFS-548 J.E. Rice,
Simulated Impurity Transport in LHD from MIST; May 1998

NIFS-549 M.M. Skoric, T. Sato, A.M. Maluckov and M.S. Jovanovic,
On Kinetic Complexity in a Three-Wave Interaction; June 1998

NIFS-550 S. Goto and S. Kida,
Passive Saclar Spectrum in Isotropic Turbulence: Prediction by the Lagrangian Direct-interaction Approximation; June 1998

NIFS-551 T. Kuroda, H. Sugama, R. Kanno, M. Okamoto and W. Horton,
Initial Value Problem of the Toroidal Ion Temperature Gradient Mode ; June 1998

NIFS-552 T. Mutoh, R. Kumazawa, T. Seki, F. Simpo, G. Nomura, T. Ido and T. Watari,
Steady State Tests of High Voltage Ceramic Feedthoughs and Co-Axial Transmission Line of ICRF Heating System for the Large Helical Device ; June 1998

NIFS-553 N. Noda, K. Tsuzuki, A. Sagara, N. Inoue, T. Muroga,
oronaization in Future Devices -Protecting Layer against Tritium and Energetic Neutrals-: July 1998

NIFS-554 S. Murakami and H. Saleem,
Electromagnetic Effects on Rippling Instability and Tokamak Edge Fluctuations; July 1998

NIFS-555 H. Nakamura , K. Ikeda and S. Yamaguchi,
Physical Model of Nernst Element; Aug. 1998

NIFS-556 H. Okumura, S. Yamaguchi, H. Nakamura, K. Ikeda and K. Sawada,
Numerical Computation of Thermoelectric and Thermomagnetic Effects; Aug. 1998

NIFS-557 T. Yasuhiko, O. Masaki, T. Katsuyoshi, O. Yoshihide, K. Osamu, A. Eiji, K. Toshikazu, A. Ryuichi and T. Masanobu,
Development of a High-Current Hydrogen-Negative Ion Source for LHD-NBI System; Aug.1998

NIFS-558 M. Tanaka, A. Yu Grosberg and T. Tanaka,
Molecular Dynamics of Structure Organization of Polyampholytes; Sep. 1998

NIFS-559 R. Horiuchi, K. Nishimura and T. Watanabe,
Kinetic Stabilization of Tilt Disruption in Field-Reversed Configurations; Sep. 1998
(IAEA-CN-69/THP1/11)

NIFS-560 S. Sudo, K. Kholopenkov, K. Matsuoka, S. Okamura, C. Takahashi, R. Akiyama, A. Fujisawa, K. Ida, H. Idei, H. Iguchi, M. Isobe, S. Kado, K. Kondo, S. Kubo, H. Kuramoto, T. Minami, S. Morita, S. Nishimura, M. Osakabe, M. Sasao, B. Peterson, K. Tanaka, K. Toi and Y. Yoshimura,
Particle Transport Study with Tracer-Encapsulated Solid Pellet Injection; Oct. 1998
(IAEA-CN-69/EXP1/18)

NIFS-561 A. Fujisawa, H. Iguchi, S. Lee, K. Tanaka, T. Minami, Y. Yoshimura, M. Osakabe, K. Matsuoka, S. Okamura, H. Idei, S. Kubo, S. Ohdachi, S. Morita, R. Akiyama, K. Toi, H. Sanuki, K. Itoh, K. Ida, A. Shimizu, S. Takagi, C. Takahashi, M. Kojima, S. Hidekuma, S. Nishimura, M. Isobe, A. Ejiri, N. Inoue, R. Sakamoto, Y. Hamada and M. Fujiwara,
Dynamic Behavior Associated with Electric Field Transitions in CHS Heliotron/Torsatron; Oct. 1998
(IAEA-CN-69/EX5/1)

NIFS-562 S. Yoshikawa,
Next Generation Toroidal Devices; Oct. 1998