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学位論文題目	Experimental and Analytical Studies of High Heat Flux Components for Fusion Experimental Reactors （核融合実験炉用高熱流束受熱機器に関する実験的及び解析的研究）
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論　文　内　容　要　旨

As the second stage of research and development (R&D) for fusion experimental reactors, recent three large tokamaks, i.e., JT-60 in japan, JET in the European Community, and TFTR in the United States of America, were designed and constructed. Break-even plasma condition which is defined that an output fusion power produced by D-T reaction is equal to an input power have been achieved in JET. Equivalent break-even plasma condition in hydrogen plasma operation has also been achieved in JT-60. These positive results elevate the importance of R&D efforts on the next stage of experimental reactors. These R&D effotrs are currently being carried out under the International Thermonuclear Experimental Reactor (ITER) program, which is a collaborative enterprise between Japan, the European Community, the Russian Republic, and the United States.

Since the ITER will be designed to the nuclear fusion power from the nuclear fusion of deuterium and tritium (D-T), reactor engineering R&D issues are very important for the ITER program. In particular, development of plasma facing components, of which the surfaces directly face the ITER plasma, is one of the key issues for the design of ITER. The plasma facing components will be subjected to more severe heat loads not only during steady state normal operation but also during off-normal events such as plasma disruptions than those for the existing fusion experimental machines. Especially, divertor plates as the most important plasma facing components. Each divertor plate will be subjected to the steady sate heat load of 15 to 30 MW/㎡ as radiation heat loads, high fluxes of energetic particles, and high energy neutrons from the plasma. During plasma disruption, the divertor plates will also be exposed to heat loads as high as 600 to 200,000 MW/㎡, corresponding to a deposited energy of around 20 MJ/㎡.

For high performance operation in ITER, it is necessary to use low-Z materials such as high performnce carbon-based materials with high thermal conductivity and strong mechanical properties as the armor tiles for plasma facing components. Furthermore, to facilitate the conduction of heat from the armor to the actively cooled heat sink structure, it is very important to develop the braze technology.

The ITER R&D program for the divertor plates is summarized in Fig. 1. R&D issues can be categorized into four developmental items: 1) the armor, 2) the heat removal structure, 3) the braze technology, and 4) the integrated performance.

With respect to developing the armor materials, the most important issue is erosion damage by the tokamak plasma.

This issue can be studied in parallel with other developmental issues because the erosion processes effect only the near surface region of the structure from the analysis.

The second experimental issue is development of heat removal structures for the divertor plates under one-sided heating condition. The following thermal charecteristics becames important;critical heat flux (CHF) of the cooling structure, critical heat flux (CHF) correlations, and heat transfer along the circumference. There are many experimental and analytical studies on critical heat flux under uniform heating condition, but only a limited amount of literature under one-sided heating condition. Since there is little literature for one-sided heating condition, systematic experimental evaluations based on experiments are necessary.

With respect to the third and the fourth experimental issuse, carbon-based materials with high thermal conductivities are brazed to a metal cooling structures because high thermal response of the divertor plate is required to handle the intense heat loads. However poor compatibility of thermomechanical properties between the carbon-based materials and the metal will lead to residual stresses. Also the adhesive properties of the bond strongly depend upon braze materials, surface treatment before brazing, and configuration of the bond.

Since the divertor plate experiences periodic high heat flux variation, an evaluation of the thermal fatigue of the divertor plate be important for the ITER design. In perticular, the thermal fatigue induced in the bonds of the divertor plate, which is strongly affected by brazing process, is critical and therefore some experimental study is required from the viewpoint of lifetime evaluation. It is also very important to investigate applications of special techniques, such as separatrix sweeping which can be used to reduce the surface heat flux of the divertor plate, from the lifetime point of view.

In view of the above statement, this thesis focuses on the following subjects :

(1) critical heat flux experiments of various cooling tubes under one-sided heating condition

(2) evaluation of critical heat flux correlations under one-sided heating condition,

(3) evaluation of the thermal response of the divertor plate under large numbers of cyclic high heat loads,

(4) analytical and experimental evaluation of sweeping effects on the reduction of the surface heat flux deposited to the divertor plate.

〈Critical Heat Flux Experiments〉

High heat flux components such as the divertor plate and beam dumps for NBI are subjected to particle and radiation heat loads under one-sided heating condition, which induces thermomechanical probleems such as bending of the tube and large thermal stress. Further, extremely high heat flux is loaded on striking points of the divertor plate. In order to develop the cooling system, therefore, high heat flux removal system will be needed. Among many heat removal concepts, a swirl tube is considered one of the most promising candidates for the cooling system of high heat flux components and an experimental study to develop high heat flux components for the next generation of fusion experimental reactors has been started, using an ion source developed for JT-60 neutral beam injector (NBI).

In this study, a modified swirl tube, namely an externally-finned swirl tube, is developed and compared with a simple smooth tube. The major dimensions of the externally-finned swirl tube are 10mm in outer-daimeter, 15mm in external fin width 700mm in length.

To evaluate quanitatively cooling capability of the swirl tubes, burnout heat flux is introduced which is defined at the tube outer surface in this study because of one-sided heating condition. The burnout heat flux, which also indicates incident critical heat flux, of 41 ± 1 MW/m² is achieved in the externally-finned swirl tube under the condition that water flow velocity, inlet pressure and temperature were 13 m/s, 0.9 MPa and 20℃, respectively. The burnout heat fluxes of the externally-finned swirl tube and the internally-finned tube increased linearly with increasing the flow velocity.

〈Critical Heat Flux Correlation〉

The ability predict the crirical heat flux (CHF) with highly subcooled flow boiling is one of the key issued in the design of plasma facing components for the ITER and FER. In particular, the divertor plate is subjected to severe heat loads under one sided heating conditions. Because there are no correlations predictimg CHF for highly subcooled flow with heating on one side, experimental data obtained under one sided heating condition have been compared to various existing CHF correlations for uniform circumferential heating conditions. Incident critical heat fluxes for various tubes were translated into the heat flux values at tube inner wall by applying heat transfer correlation with highly subcooled boiling.

Experimentally determined CHF data for straight tube show relatively good agreement with some correlations within an accuracy of $\pm$ 20%. On the other hand, all of the existing correlations systematically underpredict CHF values for over the range of the mass flow investigated in the present experimental conditions. The heat flux profile along the circumference of the rectangular faced tube inner wall is found to be similar to that for the straight tube. This is mainly attributable to the high degree of sensitivity of existing CHF corrrelations to the inner diameter.

For the externally-finned tube, no existing correlations are available for prediction of the CHF, although this tube geometry more closely approximates uniform circumferential heating conditions. Further experiments are necessary to evaluate the applicability of the existing CHF correlations under one sided heating conditions.

〈Thermal Cycling Experiments〉

From the engineering point of view, divertor mock-ups with different armor tile materials have been prepared in order to investigate their overall performance . In particular, the adhesive property between the armor tile and the heat sink metal was concentrated in this study. Thermal cycling tests of the divertor mock-ups have been carried out under ITER/FER relevant heat flux conditions in a particle beam engineering facility at japan Atomic Energy Research Institute (JAERI).

Results of these tests confirmed that bonded carbon-fiber-composite/copper (CFC/OFHC-Cu) divertor mock-ups have withstood 10.0MW/m^2 for one thousand cycles without cracks. Some bonded CFC/OFHC-Cu samples have withstood 12.5MW/m^2 for one thousand cycles without an increase of the surface temperture, although a small crack was observed at a corner of the bonded layer.

Residual stresses from brazing have also been determined using three dimensional models. The analytical results confirmed the results of the test sample manufacture efforts, that is, no cracks or detachments were observed. Since no thermal plastic fatigue evaluation codes are available for the prediction of the lifetime, experimental data were correlated with numerical thermal stresses during heating.

〈Evaluation of Separatrix Sweeping Effects〉

Magnitude of the heat flux on the surface of the divertor plate of the ITER is one of the most limiting constraints on its lifetime. A technique for sweeping the separatrix across the divertor surface will be applied to reduce surface heat fluxes and erosion damages due to intense fluxes over 15MW/m^2. As the first step for evaluation of the sweeping effects, thermal response of the divertor plate has been analyzed under ITER relevant heat flux conditions.

The analytical results show the application of the sweeping is very effective for reducing the surface temperature of the divertor plate. To realize these benefits for ITER, the divertor separatrix must be swept with a frequency of higher than 3.0 Hz over a distance of ±10 cm.

Based on the analytical results, thermal response experiments with a divertor mock-up are carried out using the JAERI Electron Beam Irradiation Stand (JEBIS). The condition for this experiment was a peak heat flux of 30 MW/m^2 with a sweeping frequency of 1.0 Hz over adistance of ± 10 cm for a 30 second long cycle. Experimental results show that the divertor mock-up has successfully endured for> 1000 major thermal cycles without increase of the surface temperature. Therefore, it has been experimentally demonstrated that application of the sweeping tecnique is very effective for improvement in the power handing capability of the divertor plate. Experimental results showed good agreement with analytical results.

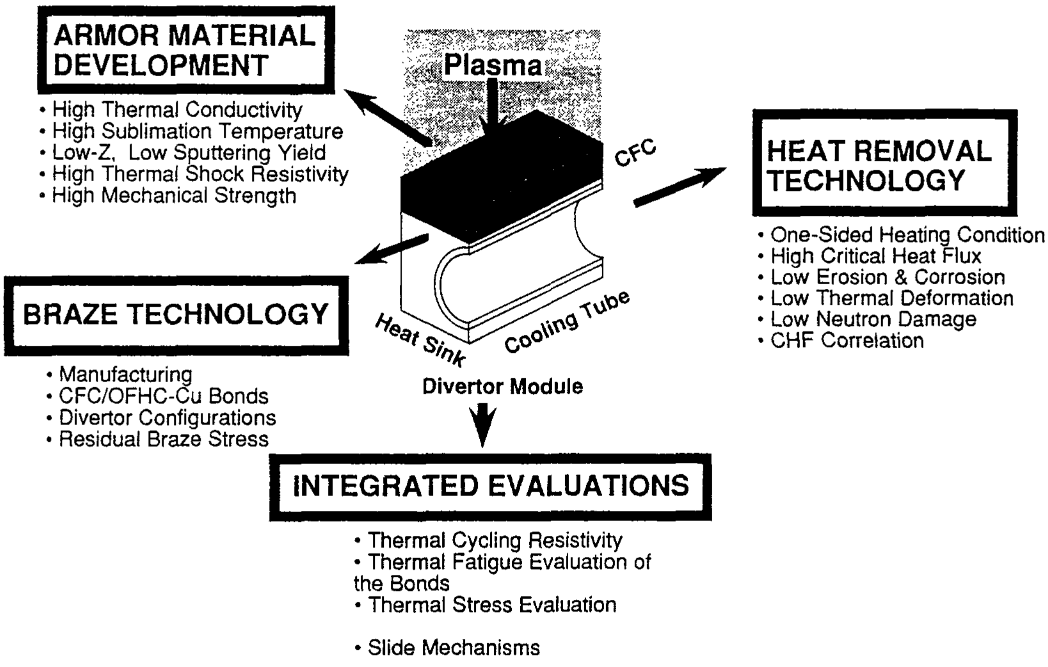

Fig. 1 R&D Issues of Divertor Plate for ITER

審 査 結 果 の 要 旨

国際熱核融合実験炉（ＩＴＥＲ；International Thermonuclear Experimental Reactor）を実現するためには，プラズマに対向して高い熱，粒子負荷を受ける高熱流束受熱機器のうち，特に定常時に最高30MW／㎡の熱負荷，プラズマ異状消滅時に受熱表面材料の損傷，損耗を引き起こすさらに高い熱負荷の作用するダイバータ板の開発が必要である。本論文は，プラズマの性能維持の観点から低原子番号の炭素系材料を表面に持つ異種材料接合冷却構造体であるダイバータ板について，その高性能冷却管の開発，異種材料の接合技術，及びダイバータ板表面熱流束の低減技術の実験的および解析的研究を行い，それらを綜合することにより新しい有用なダイバータ技術を確立したもので，全編６章よりなる。

第１章は序論である。

第２章では，ダイバータ板用高性能冷却管として外部フィン付スワール管を提案し，スワールテープによる視回流により限界熱流束を増大させ，外部フィンにより片面加熱条件下の温度分布の不均一を低減して熱変形を抑制し，さらにこれらの相乗り効果により40MW／㎡以上にも達する高い熱負荷冷却特性が得られることを，水素イオンビームを加熱源とする実験により実証している。これは極めて有用な技術である。

第３章では，片面加熱条件下にあるダイバータ板冷却管の限界熱流束を前章と同じ装置により調べ，その結果を既存の均一全周加熱条件下の予測式と比較している。円管では比較的良い一致を示すが，外部フィン付管では予測値は過小となり適用できず新たな予測式の必要を指摘している。

第４章では，プラズマ対向機器表面材料候補材の炭素系材料と冷却構造体との接合について，接合部熱負荷実験と炭素系材の異方性を考慮した３次元応力解析を行い，３軸応力成分による新しい評価法を導入してＩＴＥＲ設計条件を満たすダイバータ異種材料接合冷却構造体の開発に世界で初めて成功している。これは極めて有用な成果である。

第５章では，ダイバータ板上の熱負荷を低減するダイバータセパラトリックススイーピングに関し，炭素系材料の熱，機械的異方性の考慮による３次元熱解析と熱サイクル特性の実験により，熱負荷を大幅に軽減できるとともに温度振幅に対して十分に健全であることを実証している。

第６章は結論である。

以上要するに，本論文は核融合実験炉ＩＴＥＲに必要な炭素系材料を表面に持つ異種材料接合冷却構造体であるダイバータ板を開発するため，片面加熱を受ける高性能冷却管のフィン付スワール管の高熱流束冷却特性，繰り返し熱負荷を受ける異種材料の接合部の健全性，及びダイバータ板表面熱流束の低減技術であるダイバータセパトリックススイーピングの実験的及び解析的研究を行い，それらを綜合して新しい有用なダイバータ技術を確立したものであり，原子核工学，特に核融合炉工学の発展に寄与するところが少なくない。

よって，本論文は博士（工学）の学位論文として合格と認める。