MATERIALS INPUT FOR THE ARIES-ST DESIGN PROGRAM*

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SUMMARY

The ARIES-ST design incorporates a number of materials for which data and design criteria are required. The materials input includes the present status of the materials database, as well as progress in fabrication techniques and materials development which can be reasonably achieved over the next 25-50 years. During the past year, input was provided for the copper alloys considered for the electrically-conducting center post, the aluminum alloys proposed for the toroidal field return outer shell, the ferritic steels (low-activation and ODS) to be used as first-wall and blanket structural material, the compatibility between the slow-flowing Pb-Li breeder and the SiC thermal barrier and the ferritic steel structural material, and the tungsten selected for both the plasma-facing divertor material and the divertor coolant pipes. Some guidance was also provided with regard to fabrication and joining techniques. The level of detail provided with regard to the database and design guidelines for the various components depended on the functional importance of the component and the status of the database.

PROGRESS AND STATUS

Introduction

The ARIES Team has completed its design of the ARIES-ST spherical tokamak reactor as part of its latest power plant study. Final presentations and documentation are due at the January 28-29, 1999 ARIES Team meeting at UCSD in San Diego. Detailed information regarding design parameters, operating conditions, and material properties can be found at the public web site http://aries.ucsd.edu/PUBLIC/ by clicking sequentially on "Design Study Descriptions", "ARIES-ST", and "Engineering Design Book".

Throughout the course of the ARIES-ST design study, questions about materials performance and limits have arisen which required input from the Fusion Materials Program, as well as design criteria work performed for ITER. Documents such as the ITER Material Properties Handbook (IMPH) and the ITER Interim Structural Design Criteria (IISDC), as well as material properties summaries written for APEX have been used to resole some of these questions. On-going research efforts, such as those published in these Semiannuals, in summaries of IEA meetings and in proceedings of materials conferences (e.g., ICFRMs), have also been communicated from the Fusion Materials Program to the ARIES-ST Design Team. This author has served as the interface between the materials and design programs to facilitate communication and exchange. In particular, both material properties and materials performance evaluations have been communicated to the ARIES Design Team. In addition, design limits have been established based on the material databases.

The particular materials of interest to the ARIES-ST Design Team are: DS and PH copper alloys for the toroidal field (TF) center post, weldable aluminum alloys for the TF outer shell (which also serves as the vacuum vessel), low-activation ferritic steels (FSs) for the first-wall and blanket structural material, low-thermal-conductivity SiC or SiC/SiC composites for the thermal insulation between the FS and the hot Pb-Li breeder material, and tungsten for the divertor coolant tubes, which also serve as the plasma-facing component.

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Results

The ARIES-ST Design Team selected a DS copper alloy (AL-15) as its primary candidate for the center post, with a PH alloy (Cu-Cr-Zr) as an alternate material. The operating temperature range for this water-cooled structure is 30-115°C. The selection was made early in the design process. Since then, two factors have emerged which may make Cu-Cr-Zr a better choice than Al-15. The first factor has to do with irradiation performance. Both the DS and PH copper alloys experience a significant decrease in uniform elongation with dpa in this low-temperature range. However, recent data (ICFRM-8) suggest that the fracture toughness of Cu-Cr-Zr is significantly higher than DS copper alloys both before and after low-temperature irradiation. The second factor has to do with fabrication methods planned for the construction of the center post. Mid-way though the project, a rapid-prototyping fabrication method was proposed for construction of the center post. This includes converting powder to molten material and computer-controlled deposition and shaping. This would be extremely difficult to do with DS copper as the Al2O3 particles would tend to redistribute during the melting process. However, it may be feasible with Cu-Cr-Zr if the material can be properly heat treated either during or after the deposition and shaping process.

The aluminum TF return shell also serves as the vacuum vessel for the ARIES-ST. It is water-cooled through stainless-steel coolant tubes embedded in the aluminum. The proposed fabrication includes: welding 13-mm aluminum plates to form upper, middle and lower shell segments of the vacuum vessel; using computer-controlled spray-casting of aluminum onto each shell segment to final thicknesses ranging from 0.5-0.7 m at the mid-plane to 2.5 m at the top; and assembling the three shell segments in the power core area. The top shell is built in place in the power core area. The bottom and middle shells are constructed outside the power core area. 5000 series aluminum alloys (e.g., 5002 or 5052) are considered as prime candidate materials because of their good electrical conductivity and weldability. The outer shell operates at relatively low temperature and stress. All but a very small portion of the shell experiences very low dpa rates. From a materials perspective, several questions were raised as to the homogeneity and tensile properties of the final product. For example, would the shell segments need an O-temper anneal to increase the homogeneity of the final product? Also, given the large mass (≈2700 metric tons) of the final product, what external structures would be used to support the weight of the TF return shell?

The first wall is a He-cooled box-like structure constructed of low-activation ferritic steel. In the early designs, the He entered the first wall at 350°C and exited to the blanket at a temperature low enough to keep the FS maximum temperature ≤500°C. For thermodynamic efficiency reasons, the design was later modified to have the He enter at 300°C and exit to the blanket at 420°C, while allowing the peak FS temperature to be ≤600°C. Considerable effort was directed toward establishing design criteria for the ferritic steel (which is not included in the IMPH). The tensile and thermal creep data for the Japanese F-82H and the IEA heat F-82H (mod) were compared to the data for the ASME and RCC-MR code-qualified modified Fe-9Cr-1MoVNb. Even with the irradiation-induced softening of ultimate tensile and yield strengths of the F82-H above ≈400°C, the tensile properties of F82-H remained within the scatter band of those for Fe-9Cr-1MoVNb. Thus, the established design limit stresses and criteria for the code-qualified material could be used. The short-term tensile properties resulted in a design stress intensity of 148 MPa at 500°C, 128 MPa at 550°C and 103 MPa at 600°C. However, thermal creep for 3 full-power-years of operation limited these stresses to 110 MPa at 550°C and 61 MPa at 600°C. The design stress intensity directly limits the average (through-wall) primary stress and indirectly (though multiplicative factors) the primary bending and secondary thermal stresses. Because of the low thermal creep strength in the temperature range of 550-600°C, the decision was made to use an Oxide-Dispersion-Strengthened (ODS) ferritic steel. Because of the limited data on such steels, the tensile and creep properties of F82-H were shifted (upward) by 50°C which may be reasonably conservative. However, the change from 350°C to 300°C minimum temperature and the change to ODS FS has introduced some uncertainties into the design because of the uncertainties in the

databases for both FS and ODS-FS. The simultaneous effects of neutron damage and He transmutation on the DBTT of FS are not well established. While a minimum FS temperature of 300°C may be reasonable based on neutron irradiations without significant He production, some question remains regarding the effects of He on the DBTT of ferritic steels such as F82-H. With regard to the switch to ODS-FS, the uncertainties in the database increase significantly. Also, special joining techniques are required for ODS-FS in order to retain the creep-resistance of these alloys. Questions also remain regarding the lower temperature limit (based on DBTT) and the upper temperature limit (based on possible irradiation-induced softening).

The ARIES-ST blanket consists of a box-like structure of ODS-FS cooled internally by He exiting the first wall, a thin layer of stagnant or slow-flowing Pb-Li, a SiC thermal barrier, and slow-flowing Pb-Li breeder. The Pb-Li exits the blanket at 700°C. The database for Pb-Li/FS is reasonably well established for faster flow rates. However, the database for the compatibility of Pb-Li/SiC is quite poor (static Pb-Li with poor quality SiC). More R&D is required in this area to support the design. Also, no data were found regarding the compatibility of ODS-FS with Pb-Li.

Tungsten was chosen as the divertor material. The divertor consists of Porous Metal Heat Exchanger Tubes which are best described through the figures in the "Engineering Design Book." While the fabrication of these tubes is quite challenging, the focus of this section is the material properties of the final product. The decision to use unalloyed W rather than more-easily-fabricated W-Re alloys was based on reducing material costs. This issue will have to be revisited after a final fabrication cost estimate is made for the unalloyed W. It is specified that the W pipes will be formed by powder-metallurgy hot-isostatic-pressing, cold-worked, and stress-relieved to give optimum tensile strength and ductility. The anticipated temperature range is 250-800°C with the coolant being He. The main structural issues with the stress-relieved W are related to ductility and fracture toughness both before and after irradiation. The nominal DBTT for stress-relieved W is given as 100-200°C. However, tensile properties show very low ductility (uniform elongation, total elongation, and reduction-in-area) in the range of RT-500°C. In general, there are few data points in this range and there is considerable scatter in the data for cold-worked/stress-relieved W depending on impurities, amount of cold-work, and time-temperature of the stress relief. Whatever the DBTT of unirradiated W, the DBTT is expected to increase with neutron irradiation. It is possible that the whole divertor will eventually be in the "brittle" regime with regard to design rules. Thus, the fracture toughness and/or the critical stress intensity become extremely important design parameters. With the database for the fracture toughness of unirradiated W relatively limited, one can only expect that the database for irradiated W is even more limited. Thus, there remains a material's issue with regard to the performance of the W divertor. An additional design feature which may require optimization and demonstration is the joining of the W divertor to the FS He manifolds in the back of the divertor. Brazing has been specified with Ti-25Cr-3Be as a possible braze material.

Discussion

While a number of fabrication and materials performance issues regarding the ARIES-ST design remain unresolved, the design study concerns a power reactor with about a 50 year lead time. In evaluating the design, it is important to consider what improvements in materials performance and fabrication can be reasonably achieved over the next 25-50 years, rather than rely on the limitations of the existing database.

FUTURE WORK

The materials input to the ARIES-ST Design Team will be documented for incorporation into the final report, as well as journal articles.