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Materials for the High Temperature Reactor

by

D.E. Buckthorpe	NNC Ltd.	, UK
R. Couturier	CEA	, France
B. van der Schaaf	NRG	, Netherlands
B. Riou	FRA	, France
H. Rantala	JRC	, Netherlands
K. Kuehn	FZJ	, Germany
A. Buenaventura	EA	, Spain
B-C. Friedrich	FRA	, Germany

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MATERIALS FOR THE HIGH TEMPERATURE REACTOR

D. BUCKTHORPE
Reactor System & Advanced Technology, NNC Ltd., Knutsford, UK

R. COUTURIER
Commissariat à l' Energie Atomique (CEA.), Grenoble, France

B. VAN DER SCHAAF
Nuclear Research and Consultancy Group (NRG), Petten, Netherlands

B. RIOU
Framatome ANP, Lyon, France

H. RANTALA
European Commission, Joint Research Centre (JRC), Institute for Energy, Petten, Netherlands

K. KUEHN
Forschungszentrum Juelich GmbH (FZJ.ESS), Juelich, Germany

A. BUENAVENTURA
Empresarios Agrupados Internacional S A (EASA.MD), Madrid, Spain

B-C. FRIEDRICH
Framatome ANP (GmbH), Germany

ABSTRACT

Research activities are in progress on the modular HTR technology, supported by the European Commission within its 5th Euratom Framework Programme. Two Projects (HTR-M & HTR-M1) consider the selection and development of materials for the reactor vessel, high temperature control rod and turbine, and graphite core. They involve two areas of activity covering a literature review and database and materials testing including irradiation tests in the High Flux Reactor at Petten. This paper describes the activities of the combined HTR-M & M1 programme and some results from the first year. The paper also looks towards the planning of some key tests due to begin in 2002.

Introduction

High temperature and gas-cooled reactors have been developed within Europe over many decades and considerable expertise exists within the European Community countries in this technology. There has been renewed interest in modular and direct cycle concepts in recent years and European Community Framework Programmes have been launched to consolidate and advance this modular HTR technology in Europe [1]. Coordinated projects are underway to investigate important technological issues associated with HTR development in key some technical areas (HTR-F, HTR-M, HTR-N, HTR-E, HTR-L). They cover issues on Fuel, Materials, Nuclear Physics, Equipment, and Licensing. The projects involve partnerships between principal industrial and research organisations from different countries of the European Union, and have the individual aim to develop and consolidate European expertise and experience in each of the individual areas. A European HTR Technology Network (HTR-TN) has also been launched to coordinate and encourage work-shared actions within European companies and to serve as a channel for international collaboration [2]. This network involves over 20 organisations from industry, research and development, utilities and universities together with the Joint Research Centre

(JRC) of the European Commission. These companies have agreed to work collectively on longer-term issues to establish a firm basis for continued development of HTR's in Europe. The technological issues important to two recent industrial projects PBMR and GT-MHR are used as basis.

The project on materials (denoted HTR-M & M1) deals with the selection and development of materials for the reactor pressure vessel, high temperature resistant alloys for the internal structures and turbine and graphite for the reactor core. The work began in November 2000 and involves eight partners from five European countries. This paper gives an update of the progress of the work for the HTR2002 conference and discusses some of the initial results. The main elements of the programme have previously been reviewed [3,4] and are shown in Figure 1.

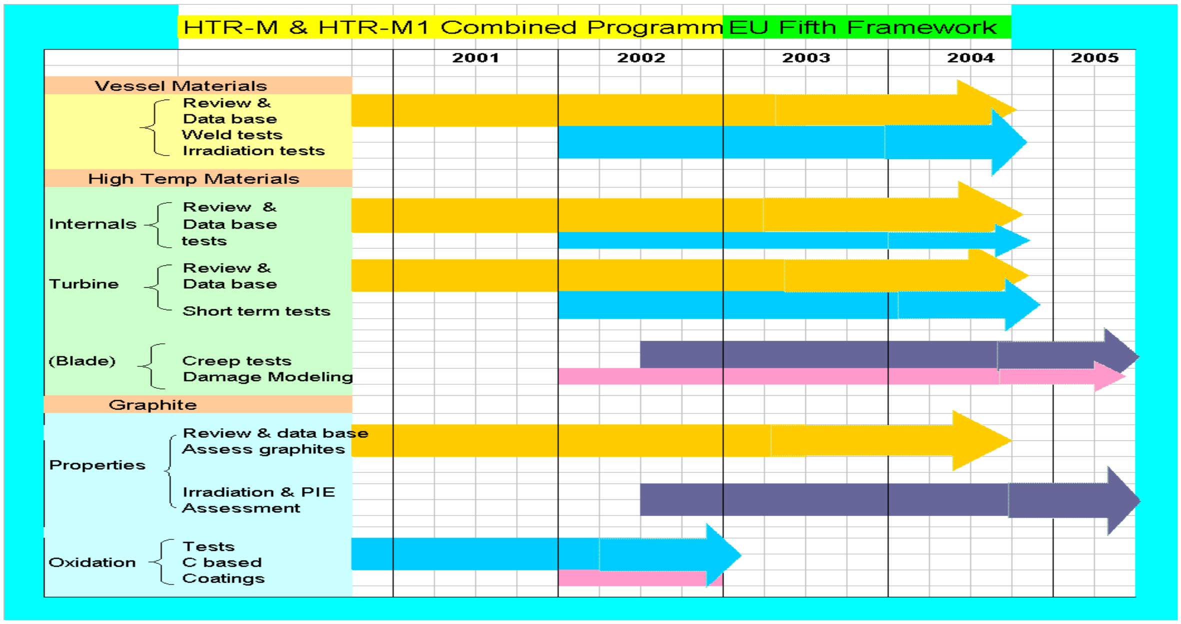

Fig. 1 Description of Overall HTR Materials Programme

HTR M & M1 Projects

The main objectives of the HTR Materials projects are to develop.

⇒ HTR vessel materials and data in the areas of design analysis, structural integrity analysis and materials properties under irradiated and non-irradiated conditions including testing on welds.

⇒ Materials for the high temperature regions of the HTR under simulated environments for both reactor internals and for the turbine

⇒ Graphite materials data identifying new graphites suitable for HTR and performing important testing and formulating a database of available information.

⇒ Graphite oxidation and consequences of severe air ingress with core burning including the development of HTR models and data, protective coatings and innovative C-based materials.

The project involves two main areas of investigation.

Review and Database

The vessel materials can operate at temperatures as high as 450^{o}C and information is required on manufacturing, environment (neutron-irradiation fluence, operational temperature, and helium environment) and welding. The primary circuit components operate at temperatures of 850-900°C and metals and ceramic materials are being considered for the reactor internals, core support, etc. with high strength alloys for the turbine. Graphite acts as a moderator and structural component and has important safety implications because of structural and other property changes that occur when it is irradiated. Information on all these components is crucial and design and material property data under HTR relevant conditions are to be reviewed and compiled for candidate materials of different HTR concepts, in relation to both normal and accident conditions.

Materials Testing

For the vessel some tests on a chosen grade of steel taken from typical welded vessel joints are to be carried out under irradiated and non-irradiated conditions. The test pieces will be irradiated in the Petten High Flux Reactor prior to performing the mechanical testing. For the high temperature control rod and the turbine components some short term and medium term testing of important material options under simulated HTR environments are to be carried out. The tests concern fracture, creep and creep fatigue conditions in air and helium environments. For graphite the tests will be carried out to determine properties and oxidation. Many existing graphites used in previous core designs are no longer manufactured commercially, and currently available grades are limited to a few specific sources of raw materials. Alternative graphite grades with a better combination of desirable properties will therefore be irradiated in the HFR and tested to determine variations in physical and mechanical properties up to low / medium doses. Also test work associated with graphite oxidation on the consequences of severe air ingress with core burning and advanced C-based options is to be performed

Review and Database

Vessel Steels

Most designs of future plant make use of the either a LWR type steel or modified 9Cr 1Mo. These materials have similar strength levels at temperatures up to 370°C. Modified 9Cr 1Mo steel allows higher temperatures with only a gradual reduction in design strength at temperatures up to 450°C and above 450°C allowable stresses for all materials fall off rapidly. The main structural integrity concerns are at the vessel welds, at thicker sections, hot spots and regions important to functionality. Fracture, fatigue and creep-fatigue (depending on temperature) are the main damage mechanisms with compatibility aspects (metal loss and carburisation) unlikely to have an impact on material choice. The database is being assembled to include such materials as C-Mn steels (as used in the UK Magnox and AGR types), SA 508 Grade 3 Class 1 (LWR) or its European equivalent, 2¼Cr 1Mo steel as used on HTTR and modified 9Cr 1Mo. Aspects such as provisions, products & parts, test and design data and environment are to be covered. Significant amounts of a data exist for these steels and the main focus is on comparisons and assembling relevant design data. Modified 9Cr 1Mo steel showed the most uncertainty on data and fabrication experience with respect to HTR characteristics. It also potentially has the wider applicability. Modified 9Cr 1Mo steel was therefore selected for the test programme in the High Flux Reactor (HFR).

High Temperature materials

Materials for the control rod and turbine (blades and discs) are being investigated. Carbon-carbon (C/C) composites offer advantages in strength, ductility and increased thermal resistance over metal alloys, giving potential for improved reactivity control during shutdown and for allowing the normal operating temperatures of future reactors to be increased [5]. For the turbine high temperatures and long term endurance are key issues. Criteria for material selection are typically to be based on a safe operation period of up to 60,000 h at 3000 rpm (50 Hz), with upper temperature limits in the range 850 to 950^{o}C. The main criteria to be considered are creep and the influence of environment. Manufacturing considerations are also a major factor for the disc. Candidate alloys for the disc must be capable of production of large defect free ingots with good forging properties and proven thermal stability. The need for cooling is another major issue since the review has shown temperature limits for suitable disc materials to be below 750^{o}C. For blades corrosion is a concern. Helium atmospheres from decomposition of methane under extremely low oxygen partial pressures are heavily carburising and can cause a significant shortening of the material creep life and accelerated creep crack growth rates. The presence of alloying elements such as cobalt (which is in most of the currently available turbine disc and blade materials) may be difficult to avoid. The main issues are the potential for plate out and lift off of particles and their activation and prevention of them flowing through the core. The review and database work has identified a few potential materials and these are being considered for the experiments.

Graphite

The graphite core is a key component that affects safety and operability of the reactor. It provides structural support, coolant channels, moderation, and shielding while operating in a high temperature helium environment. Its performance is critically dependent on the graphite properties, which are irradiation dependent. The most important considerations are component integrity and changes in core geometry, both of which are affected by the dimensional change. Many of the graphites used in previous core designs are no longer available. The main questions are on the availability of the coke and manufacturing procedure. There are only three manufacturers left in the world. The decline in the ability to manufacture nuclear grade graphite in large quantities in the UK is an example of this. Two large plants, one owned by Anglo Great Lakes (AGL now SGL) and the other by British Acheson Electrodes Ltd (BAEL, later known as Union Carbide and presently called UCAR) manufactured all the Gilsocarbon graphite used in the UK AGRs. Neither plant now exists. Today's HTGR projects - HTTR (Japan) and HTR-10 (China) use a Japanese graphite (IG-110). This graphite, with its high strength, should be taken into account for exchangeable core components where low fast neutron fluences and hence low total doses apply. The proposal is to approach each of the three remaining manufacturers to see what graphites they can offer. The work on the review and collection of graphite properties is considering the accessible information on the IAEA database, internal information and published information at seminars and conferences. The IAEA database was established to help the development of International programmes on graphite moderated reactors, assist safety authorities in assessment of safety aspects and serve as a source of scientific information for nuclear and non-nuclear technology. The database is considered to have data that is relevant to the irradiation temperature and neutron fluence domains for new HTRs.

Materials testing

Vessel Steel

The test programme for the vessel steel will concentrate on qualification of modified 9Cr 1 Mo. For this purpose an irradiation test of thick welded joints is being prepared. The tests involve the use of the Lyra rig, giving accurate control of the temperatures and use of a helium atmosphere. The heaters in the rig will be used to correct the temperature levels obtained from gamma heating. Post irradiation tests will include tensile, creep, Charpy and CT toughness tests. The fabrication and characterisation of the welded specimens involves tests on 40 and 150mm thick plate. The thinner plate specimens provide a means for trials using different weld bead lengths and different weld parameters before producing the final weld specimens with the thicker plate. The conditions used will simulate the accumulated dose at the end of life. For the tests the lowest temperature condition is being considered as higher temperatures are expected to act to anneal out any irradiation damage effects. Work has started in defining the tests for post-irradiation and reference properties and final test selections and boundary conditions. The procurement of the material and fabrication of the thick section welded test pieces will be done following completion of the characterisation tests. The test programme details are expected to be finalised by mid 2002 to allow the actual test programme to start towards the end of 2002.

High Temperature Materials

Experiments on these materials will involve short and intermediate term tests. High temperature short-term mechanical / creep tests are planned for the control rod material. Short term tensile / creep tests (air, helium) up to 850^{o}C are planned for the blade materials (2 grades) and fatigue and creep / fatigue tests up to 750^{o}C in air and helium for the disc material (1 grade). The duration of the intermediate creep tests is expected to be >5000 h. The tests on the blade material will include aged and notched samples. The test matrix and proposals for the test conditions will as far as possible bound the temperature and transient cases experienced by the turbine. The details of the test programme and necessary heat treatments will be finalised during 2002. Possible candidate materials for the blade and disc tests are cast IN738 and IN792 DS which give a good creep resistance. For the disc U720 has a potential for operation up to 700^{o}C and IN706 at 650^{o}C. Discussions are underway with suppliers to confirm initial choices and availability.

Graphite

The irradiation experiment will use a test rig that allows two distinct temperature levels to be obtained. Measurements will be on dimensional changes, Young's modulus, CTE and thermal conductivity. The final conditions of the experiment have yet to be confirmed. The current programme must use currently available graphites since the development of a new material would take 3-5 years. Ideally the test conditions for the experiment would be those that give data for candidate graphites up to the peak end of life fluence. This requires a significant effort over a long time scale, typically 10 years. The target flux in the current four year programme will be as high as possible.

The graphite oxidation work involves three experimental tasks relevant to safety analysis and licensing. Tests on graphite oxidation using THERA and INDEX facilities are well advanced. Measurements have been taken on the kinetics of graphite in oxygen for temperatures between 550 and 750^{o}C and partial pressures of between 2 and 20kPa. Oxidation resistance of three selected Carbon-Fibre-Composites (CFCs) developed for the Fusion Reactor have also been measured on THERA in air and oxygen (800^{o}C). A 3D CFC containing approximately 9% of silicon showed reactivity reduced by a factor of 2 compared with similar but non-silicon containing material. Oxidation attack concentrates on the matrix instead of the fibers. Further tests in steam are planned for 2002. Results from the high flow rate test rig INDEX are expected in mid 2002. Experiments are also planned for irradiated and non-irradiated CFC materials (up to 1dpa) that will look at strength, dimensional change, thermal diffusivity and heat capacity. Experiments to determine the thermal conductivity of such CFC materials have been made in non-irradiated material with tests on irradiated samples to be completed in 2002.

Summary and Conclusions

This paper reviews research activities on the HTR-Materials projects in support of the modular HTR technology within Europe. Overall the project is progressing to programme with review and database investigations in hand and planning for experiments started. Graphite oxidation experiments are well underway and results available. The next phase of the work will involve decisions on materials and the setting up of the tests on the vessel material, the disc and blade high temperature alloys, and for the graphite irradiation tests to enable them to start as early as possible in the programme.

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