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European Project for the development of HTR Technology - Materials for the HTR

3rd Year Scientific/Technical report for the HTR-M & M1 Projects

by

D.E. Buckthorpe

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Executive Summary

HTR-M & M1 projects have been launched within the 5th Framework Programme to consolidate and advance HTR materials technology in Europe. Work has been performed to highlight the materials requirements for HTR reactor pressure vessel, high temperature resistant alloys for the internal structures and turbine, and graphite for the reactor core. The programme of work includes intermediate creep testing of turbine materials and irradiation testing of graphite materials and involves two main areas of investigation covering review of past experience and development of database and testing of most promising materials under irradiated and non-irradiated conditions. The combined results up to the third year of the HTR-M project are summarised.

Technological survey and database for structural integrity

For the vessel materials that can operate at temperatures as high as 450°C, information on manufacturing, environment (neutron-irradiation fluence, operational temperature, and helium environment) plus welding have been assembled. Fabrication and the effects of irradiation/environment on the vessel weld behaviour are considered crucial issues in determining the structural viability of the material for future RPV application. The main safety and structural integrity concerns are at the vessel welds, at thicker sections, hot spots, the belt line and regions important to functionality. Fracture, fatigue and creep-fatigue (depending on temperature) are the main damage mechanisms and degradation mechanisms (irradiation, thermal ageing, temper-embrittlement and corrosion) the important environmental considerations. Issues to be addressed for the 'cold' vessel design concept (LWR steel) and 'warm' vessel concept (modified 9Cr 1Mo steel) have been assembled. The synthesis confirms the potential of both options and identifies important issues that need to be addressed in their development. For the database, formats have been agreed, information is being collated and the transfer of the information between partners is underway. The final housing of the data in Alloys-DB is being investigated.

Turbine materials operate at temperatures of 850-900°C (HTR) and 1000°C (VHTR). Metals and ceramic materials are being considered for the blades and disc and the control rod. The main concern is the ability of the material to withstand the temperatures and conditions in the direct cycle environment. Creep and effects of helium on properties and strength are crucial issues. For very high temperatures (VHTR), carbon-carbon composite materials offer increased thermal resistance and improved reactivity control during shutdown. The HTR-M study considers C-based materials for the control rod for which a review of published data was carried out. As with other carbon based materials significant changes in the physical and mechanical properties occur due to the effect of irradiation and temperature. The benefits of such materials over high strength metals have to be quantified through investigation and test. The work within HTR-M purposely involves setting up close links and co-operation with manufacturers and scientific experts to develop the technology base sufficiently to enable testing of irradiated material to be considered. For the turbine high temperatures and long-term endurance (60,000h) are key issues. Manufacturing considerations are especially important for the disc and blades. Candidate alloys for the disc must have good forging properties and proven thermal stability. For the blades, cast material is not sufficient, and so directionally solidified or single crystal alloys have to be considered. The review suggests that the need for cooling and possibly coatings for both disc and blades. Impurities in the helium at temperature can result in carburisation and/or de-carburisation of the super-alloys and lead to a worsening of material strength. For current disc materials temperature limits are in the region of 750°C, also a significant manufacturing development is required if a large diameter (1.5m) disc is to be adopted.

Graphite acts as a moderator and structural component and has important safety implications because of structural and property changes that occur when it is irradiated. Information on all these is crucial and design and material property data under HTR relevant conditions are required for candidate materials in relation to both normal and accident conditions. Given that almost all the graphites previously irradiated are no longer manufactured and that of the test results that do exist, most use irradiation temperatures are <550°C (so are not representative of HTR temperatures) this graphite work is recognised as an important first step in establishing suitable materials for the future. A review of past information and the basis for the development of a database have been established. It is considered that for the new HTRs and VHTRs a new graphite is required that will need significant development and confirmatory irradiation tests and because of the extent of the R&D needed, global sharing of information through international cooperation would seem to be a necessity.

Testing under irradiated and non-irradiated conditions

For the vessel, testing of welded joints of modified 9Cr 1Mo steel are planned for irradiated and non-irradiated conditions, the test pieces to be irradiated in the Petten High Flux Reactor before mechanical testing. Pilot test welds (150 mm thick) have been manufactured, inspected and machined to produce a range of test specimens suitable for tensle, creep and fracture tests. Round robin tests have been performed on suitable material to ensure testing practices, especially for fracture toughness testing, are kept similar between partner organisations. The irradiation experiment is based on poolside fluence irradiations within HFR using the LYRA Rig which allows accurate control of temperatures in inert atmospheres. Heaters in the rig will be used to maintain nominal temperature conditions (the difference in temperature between specimens maintained within 15°C). The neutron fluences will be monitored and the fluence gradients due to the flux buckling and self-shielding kept as low as possible. Pure helium will be used to provide the inert environment in the test. PIE will consist of tensile, fracture, toughness and impact plus creep tests within shielded facilities. The start of the irradiation in HFR is scheduled for the end of 2003. No previous tests have been performed under irradiated conditions for representative vessel welds manufactured with this material. It is expected that the testing will provide the first important information on the level of deterioration in properties under representative HTR conditions.

For the turbine components short and medium term testing under simulated HTR environments are underway covering fracture, creep and creep/fatigue conditions in air and simulated helium environments. The impact of corrosion is an important selection criterion since the coolant gas used in an HTR contains small levels of impurities (H_2, H_2O, CO, CO_2) at low partial pressures, and these can interact with the core graphite and metallic components and cause degradation of their properties. Udimet 720 was chosen for the disc material and CM 247 LC DS and IN 792 DS for the blade materials with tests conducted using four simulated environments.

- As received with ageing heat treatment – I.
- Decarburised – II.
- Carburised – III.
- After the carburisation or decarburisation heat treatment – I(mod).

With all tests carried out in air following treatment. Udimet 720 is considered to have a potential for operation up to 700°C (tests carried out up to 750^{o}C) and to be the best available using established manufacturing techniques for large disc applications. The blade tests are performed for periods of up to 10,000h and at temperatures up to 850^{o}C. Preliminary results in air show material strength data to be consistent with available published results.

For the graphite, tests are proposed to measure properties following exposure to irradiation and to investigate the effects of oxidation. The latter primarily for understanding graphite burning under severe air ingress accidents. In addition work has been performed to review the progress made on SiC coatings in Germany. This programme successfully achieved a very good corrosion resistant quality coating on IG 110 and V438 graphites with no damage or weight loss observed after being irradiated and corroded and dropped from a height of 50 cm.

For the graphite irradiation programme five graphites were selected (supplied by three manufacturers), two iso-molded and two extruded graphites, manufactured from pitch coke and petroleum coke plus IG100 (HTTR graphite). The iso-moulded graphites are made using a petroleum coke, to provide a reference A range of grain sizes (1mm down to 10 μm) was also selected. Additional 'piggy-back' samples are also included in the irradiation capsule to extend the knowledge of other grades. The irradiation test provides a number of data points within a single test run by using the flux (buckling) distribution available in the HFR capsule. It is not possible within the current project timescales to provide data for candidate graphites up to the peak end of life fluence. The current available locations suggest an irradiation for 1-1.5 years to EDN fluence ~6-7 1025 m-2 (approx. 8 dpa). Although this is not expected to yield properties up to turn round behaviour it will provide information on whether the graphite is useful or not and provide an opportunity of replacing those that are unsuitable with alternatives if required. This irradiation looks at physical property changes at one temperature (750^{o}C) and therefore requires continuation in the next framework programme not only to complete the information up to 'turn-round' but also to look at different temperatures. The Irradiation begins in 2004 with a first PIE in the spring/summer of 2005.

A series of graphite oxidation tests on structural and fuel graphite and promising carbon fibre composites from the Fusion programme have been completed. The tests look at the acquisition of data for models that simulate burn off dependent chemical reactivities (regime 1) which occur at lower temperatures and the consumption of oxidising gas within the pores (regime II) at intermediate temperatures. At high temperatures external mass transfer is important and the reaction is restricted to the geometrical surface only. The results suggest that un-graphitised binder material has to be treated as a separate component because of its remarkably different oxidation behaviour. The oxidation behaviour in regime 1 is characterised by a continuous rate verses burnoff curve with one (more or less) pronounced maximum and the shape of the rate vs. burn-off curve is roughly similar in air and steam although the rates are markedly different. Further tests are underway on the graphites chosen for the irradiation programme.

Overall programme

This report presents the current status of the work three quarters the way through the HTR-M programme which is largely on course to complete its work within the intended time frame. The main irradiation experiments are due to start shortly and the focus in this final stage will be on maintaining the high level of commitment and quality shown so far in order to reach a successful conclusion. The technical tasks addressed by the project are at the forefront of understanding and the results are expected to have an important impact on materials selection and development for the future HTR and VHTR modular designs.

1 Scientific and Technical Objectives

The need for reliable material property data is a key issue in the development of the HTR and VHTR technology. The development of advanced HTR concepts requires an understanding of material behaviour plus data information under representative reactor operating conditions for the main HTR components important to safety and feasibility. The HTR-M & M1 projects deal with the selection and development of materials for the vessel, high temperature components and the graphite core with specific experiments on selected materials, including irradiation tests on vessel weldments and graphite.

The main objectives are:

- to establish a materials platform on which to develop a future HTR technological basis within Europe;
- identify materials of interest and establish an HTR database of information and carry out testing of some of the most promising options for future application.

The scientific and technical objectives for the three main areas of work are summarised as follows.

Reactor Pressure Vessel (RPV)

- Review existing materials used in gas-cooled reactors and previous high temperature reactors.
- Set up a materials database on design properties (which will lead to identification of data omissions for future R& D testing).
- Carry out specific tests on welded joints under irradiated and non-irradiated conditions.

Fabrication and the effects of irradiation/environment on the vessel weld behaviour are considered crucial issues in determining the structural viability of the material for future RPV application.

The project resources allow one type of vessel steel to be irradiated. For the tests a recommendation is made, depending on operating conditions, data omissions and available property and fabrication details.

High Temperature Materials – (Control Rod plus turbine disc & blade)

- Review existing materials used in gas-cooled reactors and previous high temperature reactors.
- Set up a materials database on design properties (which covers a few potential materials for each component).
- Perform specific tests on selected materials (carbon/carbon (C/C) composites, high alloy steels) at temperature and under short and intermediate times in air, vacuum and helium.

The main concern is the ability of the material to withstand the temperatures and conditions in the direct cycle environment. Creep and effects of helium on properties and strength are crucial issues.

Graphite core

- Review of existing graphite materials irradiation data and formulate a database of available information.
- Identify which of the currently manufactured graphites would be suitable for a HTR.
- Study of irradiation testing of a few selected graphites.
- Graphite oxidation and the investigation of the consequences of severe air ingress with core burning, including the development of models and data, protective coatings and C-based materials.

The graphite programme is an important first step in establishing suitable materials for future HTRs given that almost all the graphites previously irradiated are no longer manufactured. Note that of the test results that do exist, most use irradiation temperatures <550^{o}C and so are not representative of HTR temperatures

2 Scientific and Technical Progress

The projects essentially consider two main areas of investigation: the review of past experience and assembly of available properties (including available design information) plus tests on key materials where necessary information is considered lacking or scarce. The following sections summarise the scientific and technical progress made up to the end of the 3rd Year of the HTR-M Project.

2.1 Work Package 1 of HTR-M - Reactor Pressure Vessel (RPV)

The specific tasks involved in this work package are as follows.

Task 1 Review of RPV materials and design
Task 2 Establish a Database
Task 3 Weld production and testing
Task 4 Irradiation and PIE
Task 5 Synthesis

The following sections review the progress of work in each of the above tasks.

(a) Task 1 - Review of RPV materials and design

The material review work has considered developing reactors (HTR-M 01/9 D.1.1.9) as well as experience from other Reactor systems.

The RPV provides part of the pressure boundary that contains the helium coolant during normal and abnormal conditions. It also provides structural support and alignment for the components within it (core and core support structures).

For the reactor pressure vessel assurance of safety is of key importance. HTR vessel materials will operate at temperatures higher than 450^{o}C and information on materials behaviour during manufacture, in the operating environment (neutron-irradiation fluence, operational temperature, and helium environment) and welding are needed for assurance of integrity. Two vessel options are considered in the technological review.

The 'cold' vessel design concept adopted for the PBMR reactor pressure vessel uses a Mn-Ni-Mo or C-Mn type steel with grades similar to SA 508 Class 2. The advantage for these steels is the large experience that can be taken from PWR vessel technology and design rules.

The 'warm' vessel concept adopted for the GT-MHR reactor pressure vessel leads to the choice of Cr-Mo steels with modified 9Cr 1Mo steel grades similar to ASME Grade 91 being the prime candidate. For the GT-MHR the operating conditions are around 400 – 460°C with potential temperature excursions up to 570°C. Thermal and mechanical loads, in combination with environmental conditions can enhance degradation and can have a significant effect on reliability.

The main safety and structural integrity concerns are at the vessel welds, at thicker sections, hot spots, the belt line and regions important to functionality. Fracture, fatigue and creep-fatigue (depending on temperature) are the main damage mechanisms and degradation mechanisms (irradiation, thermal ageing, temper-embrittlement and corrosion) the main environmental considerations. The effects of irradiation on the toughness of LWR steels have been studied extensively elsewhere. The major effect is an increase in the ductile brittle transition temperature (DBTT) due to the combined effects of matrix hardening and grain boundary weakening. This is often associated with the residual elements phosphorus (P) and copper (Cu). For LWR this threat is generally considered to be significant when irradiation causes a change from ductile to brittle fracture at temperatures within the operational envelope. For the warm vessel option, no data are available for modified 9Cr 1Mo steel irradiated under conditions expected for the HTR. Information has however been obtained at much higher doses (≥1dpa) and these show a strong effect of irradiation temperature in the range 250-450°C. Effects may be small on ΔDBTT at 400-440°C but measurable at 350°C. There is no information on the behaviour of modified 9Cr 1Mo steel welds.

On thermal ageing embrittlement (TAE) the base materials and weld metals of both cold and warm HTR vessel steels normally have very fine grain sizes and therefore the most likely embrittlement location is the coarse-grained heat affected zone. This is a narrow region adjacent to the fusion boundary where temperatures of 1100 to 1300°C are reached during welding. Cold vessels operate at temperatures below the embrittlement range (350-550°C) so thermal ageing embrittlement is only possible during transients involving exposure to higher temperatures for tens or hundreds of hours. Warm vessels on the other hand operate in the TAE temperature range throughout their lifetime and therefore TAE is of greater concern with regard to vessel integrity.

For corrosion, the main concerns are on metal loss/carburisation/decarburisation due to actions of the impurities in the helium coolant. For the HTR vessel, metal losses will be either by controlled carburisation in cold wall vessels, and decarburisation combined with internal oxidation in warm wall vessels. The kinetics of these processes determine the effective metal loss.

The Technical synthesis of information has been compiled in a series of documents and summarised in deliverables 1.2 & 1.3 of Table 1. The relevant supporting technical documents relating to this task are listed in table 2 and concern the following document numbers.

HTR-M 01/3 D 1.1.9	HTR-M 02/02 D.1.1.28	HTR-M 02/11 D 1 1 48
HTR-M 01/10 D 1.1.17	HTR-M 02/04 D 1.1.31	HTR-M-02/11 D 1 1 49

HTR-M 01/10 D 1.1.18	HTR-M 02/08 D 1 1 40	HTR-M 03/05 D 1 1 70
HTR-M 02/02 D.1.1.27	HTR-M 02/09 D 1 1 41	

(b) Task 2 - Establish a Database

An important objective of the HTR-M project is the development of a database of materials information for the key components which includes the Reactor Pressure Vessel. Several materials have been considered including C-Mn steels (as used in the UK Magnox and AGR types), SA 508 Grade 3 Class 1 (LWR) or its European equivalent, 2¼ Cr-1Mo steel as used on HTTR and modified 9Cr1 Mo. 12Cr steel has also been included although because of poor welding capabilities this material is unlikely to be a serious contender. Aspects such as composition, manufacturing information, test and design data plus environment are covered. The materials property database includes code data where available, and raw data for analysis. Significant amounts of data exist for all these steels and the main focus is on comparisons and assembling relevant design data.

The database jointly covers vessel steels, high temperature alloys and graphites. The overall layout of the database was discussed in a preliminary document outlining its aims and format. The overall layout of the spreadsheets to be used in the database covers material designation, provisions, products & parts, test data and design property information. Blank spreadsheets in excel sheet format for the Vessel steels have been described and issued in HTR-M 02/11 D 0 0 51. The design property information is one of the main outputs from the database and is expected to help drive future decisions on information needs and tests. Design needs plus omissions and shortage of test data is expected to be the basis on which the database will be built up and expanded.

The Alloys-DB system developed at JRC for high alloy steels is being considered as a future potential for housing the information on the web to allow remote partner access and to maintain secure transfer of information between the different partners and countries. A PC version of Alloys-DB has been provided to NNC for assessment of the programme and its suitability for this purpose and suitable agreements to allow access by all the HTR-M participants drawn up by JRC.

A document dealing with the management and transfer of the information provided by the HTR-M partners has been issued, also a series of datasheets have been compiled by NNC and loaded onto the SINTER website. The next steps involve finalising transfer and assembling the information for partner access. The relevant supporting technical documents relating to this task are listed in table 2 and concern the following document numbers.

HTR-M 01/3 D 0.0.1	HTR-M 01/11 D.0.0.22	HTR-M 03/08 D 0 0 75
HTR-M 01/3 D 0.0.2	HTR-M 02/11 D 0 0 50	
HTR-M 01/3 D.0.0.21	HTR-M 02/11 D 0 0 51	

(c) Task 3 - Weld production and testing

Modified 9Cr1 Mo steel showed the most uncertainty on data and fabrication experience with respect to HTR characteristics and potentially has the wider applicability hence this material was selected for the irradiation test programme.

For the tests, some representative thick section welds were manufactured by Framatome using narrow gap TIG with a double V weld preparation. A reduced phosphorus filler material was used purchased from the producer Kobe steel. This has been successfully used by Framatome in the past for the producing 30mm thick welded joints with a large range of welding parameters and considered the best available for the application.

The first pilot test welds (150 mm thick) showed a significant number of micro-cracks (24 over a area of 6600mm^2) in the weld metal. These cracks (< 0.3 mm) were not detected using the usual X-ray or ultrasonic examination but seen when performing a metallurgical examination. The shape of the cracks indicated hot cracking. This was not expected as the selected filler had the lowest tendency to hot cracking among the products tested by Framatome. The fabrications were repeated using a lower energy weld sequence and although there was a significant reduction the problem was not completely solved. The number of micro-cracks was reduced to 5 (see fig. 1). Changing the welding parameters further was not considered a viable option since the tests would need to mirror industrial practice as far as possible. Following discussion (Ref. 1) and recognising that procurement of a new filler material would take the best part of a year and that there would be no assurance of success or that a new filler would be available commercially, it was agreed to adopt the Kobe filler with reduced energy for the tests. It was expected that the HAZ would be defect free and the density of the cracks in the weld metal allow machining of defect free specimens to be achieved. A procedure for identifying cracks and machining and checking of test pieces was developed in co-operation with JRC and NRG.

This work represents the first trials in this material for such thick section vessel type welds and considers only one geometry. These tests are therefore a first step in confirming the fabrication parameters for such an application. Later developments in the fabrication procedure will include forged parts.

A test programme has been established consisting of tensile and creep tests. The tests will be performed by NRG (irradiated) and JRC (non-irradiated). Round robin tests have been performed on suitable material to ensure testing practices, especially for fracture toughness testing, are kept similar between NRG and JRC. The test temperature was chosen as 375°C. Some tensile tests will be performed at room temperature. Creep testing will be performed at around 500-550°C. The irradiation dose imposed on the material will be approximately 20mdpa

The presence of the hot cracks within the specimens necessitated the development of an (ultrasonic) inspection procedure to ensure only sound material was used for the irradiation tests. Unfortunately after inspection and machining of the specimens it was not possible to fill all the material locations available in the Lyra Rig. The small number of spaces not occupied by weld materials will be filled with P91 grade steel.

The relevant supporting technical documents relating to this task are listed in table 1 and concern the following document numbers.

HTR-M 03/02 D 1 4 63	HTR-M 03/08 D 1 4 78	

(d) Task 4 - Irradiation and PIE

The objective of the irradiation experiment is to achieve the End of Life (EOL) neutron dose with a relevant neutron spectrum for the HTR pressure vessel at the nominal operating temperature. The EOL neutron dose depends on the design and type of core. The conditions were estimated as follows:

- peak neutron fluence – $8x10^{22}$ n/m^2 (E>0.1 MeV);
- nominal irradiation temperature - 375°C +/- 10°C;
- irradiation to take place in the poolside of HFR.

The design of the irradiation experiment is based on previous pool side fluence irradiations performed at NRG using the Lyra Rig which allows accurate control of temperatures in inert atmospheres. Figure 2 shows details of a typical loading. Tungsten plates will shield the specimen holder from gamma radiations to avoid high gamma heating. Heaters in the rig will be used to maintain nominal temperature conditions. The difference in temperature between specimens will maintained within 15°C.

The actual temperature of the specimen will be monitored using thermocouples at different levels within the irradiation rig. The neutron fluences will be monitored using pieces of nickel-cobalt, titanium, iron and niobium which will be analysed after the irradiation. The fluence gradients due to the flux buckling and self shielding will be determined using neutron monitors. The specimen fluence gradients will be kept as low as possible (<5%) with the notch axis parallel to the HFR core box. Pure helium will be used to provide the inert environment in the test.

PIE will consist of tensile, fracture, toughness and impact plus creep tests within NRG shielded facilities. There are 3 shielded creep machines reserved for the irradiated testing covering 4 conditions (approx 100, 300, 1000, 3000h = ~ 4500h). It is expected that testing will take around 250 days commencing soon after the cool down period following the two cycles of poolside exposure.

The relevant supporting technical documents relating to this task are listed in table 2 and concern the following document numbers.

HTR-M 02/11 D 1 1 48	HTR-M 03/02 D 1 1 70	HTR-M 03/08D 1 4 78
HTR-M 03/08D 1 4 63		

2.2 Work Package 2 of HTR-M - High Temperature Materials

The results from tasks involved in this work package are as follows.

2.2.1 Work Package 2.1 – Control Rod

Task 1 Specifications for Control Rods
Task 2 Review and database
Task 3 Supply and simple tests on Carbon-Carbon Composite material.

(a) Task 1 - Design Specification for the Control Rod

A report on the design use and specifications for the control rod has been issued that contains information on the function, geometry (control rod plus core) and typical transient and operating conditions based on information available from the PBMR and GT-MHR. For the GT-MHR, the transients are separated into five categories and values that envelope temperatures in the helium gas in each of these categories have been defined. Information on maximum heat release in the structural material and rod absorbent is also given together with considerations for seismic, He impurity content and neutron fluence. For the PBMR the main information is on the system, function and geometry. As a first assessment the operating conditions of the GT-MHR were expected to bound the overall requirements and have been used. The information will be expanded as more detailed information on PBMR and GT-MHR becomes available.

HTR-M 01.07 D 2 1 14		

(b) Task 2 - Review and database

For non-graphite internal components (e.g. core support plates, grids, etc.) austenitic and some ferritic steels can be used. Such materials have an established capability at temperatures up to 550°C and have been used within the reactor block of other high temperature reactor projects (e.g. Advanced Gas Cooled Reactor in the UK, the European Fast Reactor Project and High Temperature Reactor Projects such as AVR). For higher temperatures however, nickel based and Fe-Cr-Ni Alloys need to be considered.

For the Control Rod, Alloy 800 has been adopted for past and current test reactors (e.g. HTTR and AVR). The control rod compensates for fuel burn-up and power variation reactivity effects and is used to control the reactor operation under fast normal operating modes and to shutdown the reactor system. Available metallic materials (such as alloy 800H) are considered to be at their operating limit. For the HTTR a two-stage insertion of the control rod has been necessary to avoid damage. As an alternative to Alloy 800, JAERI have considered the potential of Carbon-Carbon composite materials based on a carbon fibre weaving technique (which provides strength in the circumferential, axial and radial directions). The most recent located article suggests non-irradiated data on two grades of C/C composites have been evaluated at JAERI. Initial tests have shown a stable fracture behaviour and high strength compared with metallic alloys. Irradiation tests within JAERI are planned from 2005 on such materials.

For very high temperatures (VHTR), C/C composite materials offer increased thermal resistance and improved reactivity control during shutdown. As with other carbon based materials such as graphite significant changes in the physical and mechanical properties occur due to the effect of irradiation and temperature. A technical synthesis of published information for alloy 800 and 18 different C/C composite materials was produced that reviews the available published material properties in the non-irradiated and irradiated conditions. The findings show that the properties are largely an-isotropic and are severely degraded by neutron irradiation, the level of degradation being dependent on level of irradiation and temperature. From the references studied, only limited information on irradiation induced properties were available although some general trends in behaviour were seen in properties such as thermal conductivity and thermal resistivity. Also the high cost and limited supply of suitably manufactured material was prohibitive. The information gathered provides a basis for recording available published information in a database format and assessing likely needs and property requirements. To date a number of data sheets have been compiled for some of the commercially available materials.

HTR-M 01/09 D 2 1 11	HTR-M 02.06 D 2 1 36	HTR-M 03/08 D 0 0 76
HTR-M 01/09 D 2 0 13		

(c) Task 3 – Supply and testing of Carbon-Carbon composite material

For tests, a supply of C/C composite material was needed in a cylindrical form with dimensions representative of a typical HTR control rod cross section. Procurement of suitable material however proved difficult within the budget figure set. The use of C/C composite materials for control rods is not a proven technology. The material sits permanently in a neutron field accumulating fast fluences that change its structure and properties. There was therefore a clear need to define the properties needed. The selection of the material is very important and given its specialised (and expensive) manufacture and the significant influence of irradiation on properties, it was agreed (Refs. 2,3) to channel the remaining resources into material selection rather than performing any non-irradiated tests on material that may not be finally chosen for the application. This increased emphasis on selection would involve direct discussions with manufacturers both of our needs and for assessing the potential benefits of different available materials. A revised programme involving discussions with manufacturers and specific review of experience at FZJ was agreed at the HTR-M & M1 progress meeting held at FZJ (Ref. 4) and scheduled deliverables revised to record the resulting conclusions.

2.2.2 HTR-M Work Package 2.2 – Turbine Disc & blade materials

Task 1 Specifications for turbines.
Task 2 Review and database.
Task 3 Supply and short-term tests on selected materials at high temperature in air and simulated atmospheres.

(a) Design Specification for the Turbine

A report on the design use and specifications for the turbine has been issued that contains information on transient and operating conditions based on information available from the PBMR and GT-MHR. Information is sparce and as with the control rod specification document the content will be expanded to include more detailed information as it becomes available and as the project develops. As a first assessment the operating conditions of the GT-MHR were expected to bound the overall requirements and have been used.

HTR-M 01/10 2 1 19		

(b) Review and database

The review considered eight potential materials for the turbine disc, thirteen for the blade (including the Russian backup options) and three structural materials for the stator and ducting, etc. Creep resistance and high temperature strength were important requirements and expected to be a main selection criterion.

High temperatures (850-900°C) and long-term endurance (60,000h) are key issues. The main concerns are creep and the influence of the environment. Manufacturing considerations are especially important for the disc and blades. Candidate alloys for the disc must have good forging properties and proven thermal stability. For the blades, cast material is not sufficient, and so directionally solidified or single crystal alloys have to be considered. The review suggests that the need for cooling is an important issue for both disc and blades. For current disc materials temperature limits are in the region of 750^{o}C, suggesting that a cooled disc could be necessary for the design.

The impact of corrosion is also an important selection criterion since the coolant gas used in a HTR contains small levels of impurities (H_2, H_2O, CO, CO_2) at low partial pressures, and these can interact with the core graphite and metallic components and cause some degradation of their properties. Low concentrations of H_2O and H_2 are produced by leakage and/or desorption from both metal surfaces and graphite. CO, CO_2, and CH_4 may be produced by coolant/graphite reactions at high temperatures. Depending on the partial pressure of the impurities the resulting atmosphere can either be oxidising or carburising for the selected materials. Damage may be either at the surface or may diffuse to significant depths in the metallic matrices (internal oxidation or carburisation or decarburisation) resulting in a loss of mechanical properties over a significant thickness of the material. The presence of alloying elements such as cobalt (which is in most of the currently available turbine disc and blade materials) is a further issue as is the potential for plate-out and lift-off of particles and their activation and prevention of going through the core.

The review and database work identified some potential materials to be considered for the experiments. Investigation of turbine disc materials revealed Udimet 720 to be the best currently available candidate that would not require significant manufacturing development. For the perceived endurance this is thought to have a temperature ceiling of around 700-750°C and is likely to require cooling. Two DS grades of material were selected for the blade tests (CM 247 LC as an Al-oxide former and IN 792 as a Cr-oxide former). Because of corrosion considerations, the current view is that any blades may have to be coated for long life. Application of suitable coatings can arrest creep reduction tendencies, but their use requires an understanding of the potential for interfacial cracking at the coating layer to avoid the development of more significant cracking from the interface. Coating must however be of the same type (chromium oxide former or aluminium oxide former) as the base alloy. The choice of coating remains to be made and tested in a future Framework Programme. Adoption of a cobalt free material was not considered a priority.

HTR-M 01/03 D 2 0 23	HTR-M 02/11 D 2 0 52	HTR-M 03/01 D 2 0 62
HTR-M 02/04 D 2 2 32	HTR-M 02/12 D 2 0 54	HTR-M 03/08 D 0 0 75
HTR-M 02/07 D 2 1 39	HTR-M 03/01 D 2 2 60	

(c) Supply and short term tests on selected materials at high temperature in air and simulated atmospheres.

Supply of turbine disc material was supplied by Aubert & Duval. Three small blocks (125mm x 70 to 100mm thick) and a large pancake (320mm dia. X 800mm thick) of forged heat-treated material were obtained. Metallography, tensile properties (650-750°C) and creep strength data (750°C) in air were obtained for the material and shown to be consistent with published data (Fig. 4).

The Blade materials (CM 247 LC DS and IN 792 DS) were obtained from Doncasters and specimens produced (16 mm dia.X 150 mm length, FPI, X-rayed – heat treated in vacuum (1220°C/2h + 1240°C/2h + 1246°C/2h/80°C/min to RT+ 1080°C/4h/FGC + 900°C/8h/FGC).

The High temperature mechanical/creep tests to be carried out (on both the turbine disc and blade materials) cover four simulated environments.

- As received with ageing heat treatment – I
- Decarburised – II
- Carburised – III
- After the carburisation or decarburisation heat treatment – I(mod)

It was originally intended that tests for conditions I and I(mod) (to be done at CEA) should be carried out in air, and tests for conditions II and III, (to be done at JRC) should be carried out in argon. After some discussion it was agreed that all tests would be carried out in air. Ageing would be carried out in an argon atmosphere, using oversize blanks.

The test matrix and proposals for the test conditions will as far as possible bound the temperature and transient cases expected to be experienced by the turbine. Maximum temperatures for these series of tests is 850^{o}C.

For the disc materials Udimet 720 is considered to have a potential for operation up to 700°C and to be the best available using established manufacturing techniques for large disc applications. The tests on the disc material are carried out at 700 and 750°C (see figure 5).

Tensile Tests in air on as received heat-treated material (I) on base material CM 247 LC DS have also been performed at CEA (see figure 6). Delays in conditioning specimens at JRC have meant that further testing of specimens in the de-carburised (II), carburised (III) and (II) or (III) with heat treatment have had to be put back. The start of tests on these specimens is expected from early 2004.

HTR-M 02/06 D 2 2 35	HTR-M 03/07 M 2 0 72	

2.3 Work Package 1 of HTR-M1 - High Temperature Materials

2.3.1 Medium term creep tests

The medium term creep tests are performed on both blade alloys for periods of up to 10,000h at 850^{o}C. Tensile and short-term creep tests will also be performed on IN 792 DS aged at 850°C-10000h at temperatures up to 850^{o}C and creep periods of up to 300h.

Initially, specific mechanical tests were foreseen to study the damage evolution in blade and discs alloys (HTR-M1). Additional Creep tests on notched samples were planned, together with microstructural investigations of damage (crack lengths, orientations, location, etc.). Associated with these, it was planned to identify the parameters of usual damage models on our tests. With this programme only two tests environment for each grade (air and HTR helium containing impurities) were planned. Following discussions (held principally during the Julich meeting) the consortium agreed to abandon the "representative helium" condition and to test the specimens in 4 states as indicated in section 2.2.2c) above.

The proposed test matrix was adapted accordingly and the samples dedicated to damage analysis abandoned. Damage analysis on the tests are therefore only be performed as follows;

- observation of fracture surface by SEM;
- identification of damage type (intergranular, etc.) on longitudinal cuts;
- identification of damage parameters that would be suitable to fit the creep data using the usual creep damage models for super-alloys (i.e. MacLean and Dyson models).

The start of the medium term creep tests on the treated specimens is expected from early 2004.

HTR-M 03/07 M 2 0 72	HTR-M1 02/09 D 1 1 7	HTR-M1 03/03 D 1 0 12

2.4 Work Package 3 of HTR-M - Graphite Core

The specific tasks involved in this work package are as follows:

Task 1 Review/State of Art
Task 2 Database
Task 3 Graphite selection
Task 4 Graphite Oxidation
Task 5 Synthesis

The following sections review the progress of work in each of the above tasks.

(a) Review / State of Art

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 (see figure 7). Many of the graphites used in previous core designs are no longer available and there has been a serious decline in ability to manufacture nuclear grade graphite in large quantities, the main questions concern the availability of the coke and manufacturing procedure. Current HTGR projects - HTTR (Japan) and HTR-10 (China) - use a Japanese graphite (IG-110) which, with its high strength, is suitable for exchangeable core components where low fast neutron fluences and hence low total doses are applicable.

In the past, only a small number of irradiation programmes were aimed at determining the irradiation behaviour of graphites at high temperatures, i.e. >600^{o}C. Two of the most significant were for the European Dragon HTR programme, and the German HTR programme. Unfortunately most of the data obtained by other countries are confidential and cannot therefore be published in open literature. The work on the review and collection of graphite properties considers 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 data relevant to the irradiation temperature and neutron fluence domains for new HTRs will largely come from the new tests and built up for each graphite grade and sample orientation, and contain the appropriate details of the graphite i.e. grade, manufacturer, coke source, grain size and manufacturing method.

For nuclear applications, the graphite has to be as free as possible from impurities. Most of the impurities present will become activated during the operating life of the reactor, which will give rise to operational problems, as well as decommissioning and final disposal problems. Most impurities, however, are volatile and so disappear during graphitisation. To remove as much of the remainder as possible, halogens are added generally during graphitisation to aid the conversion of metal impurities and boron particularly, to their more volatile halides (extremely low boron levels are important from a reactor physics point of view, as it is a very strong neutron absorber).

HTR-M 02/06 D 3 1 37		

(b) Database

Most of the available irradiated materials data are obtained from various materials test reactors and often it is not possible to complete an irradiated materials property database for a particular graphite in advance of design and construction phases of a reactor. In such cases design databases can be used to arrive at the required information, developed from previously irradiated graphites of similar microstuctures, and an understanding of irradiation damage in the polycrystalline graphites. R & D activities have been undertaken in a number of countries to investigate graphite properties. The IAEA International Irradiated Graphite Database Technical Steering provides a valuable forum for the exchange of information and opinion on graphites and has recently held seminars to review the current situation.

Two of the most significant experimental programmes in the past were for the European Dragon HTR programme, and the German HTR programme. Unfortunately most of the data obtained by other countries are confidential and so far are not published in open literature. However, data from the German HTR programme have been made available for publication and for the project. These data are being collated onto spreadsheets in a set format for use within the project. It is planned to present the experimental data on the new graphites (section 2.4.2) in the form of recommended 'design curves' for each required material property. Blank data sheets have been prepared for this purpose.

HTR-M 02/06 D 3 1 38	HTR-M 03/08 D 0 0 77	

(c) Graphite selection

The final choice of graphite for the core should be based on a number of factors, although the most important will be the effects of fast neutron irradiation on it's properties up to the peak doses envisaged. Given the best graphite available, it is the task of the core designer to produce a design that will operate safely over the design life of the reactor. The most important considerations are component integrity and changes in core geometry, both of which are affected by the dimensional change.

Meetings and discussions have been held with graphite manufacturers and there has been much interest in providing materials for the test programme. Suitable graphites were made available by UCAR and SGL who have offered extruded and iso-moulded graphites for the irradiation experiment. There has also been interest from other graphite manufacturers to supply material for the experiment and JAERI have offered international collaboration through Toyo Tanso who have proposed to supply the graphites they use in the HTTR reactor. It is clear that further input to graphite selection will be strongly influenced by the outcome of the planned irradiation experiment discussed under item 2.4 2 below.

HTR-M 02/11-D-3.2.53		

(d) Graphite Oxidation

The graphite oxidation work involves two principal tasks relevant to safety analysis and licensing of HTRs for normal operation.

- The improvement of experimental database for advanced graphite oxidation models.
- Experimental investigation of innovative C-based materials with respect to their application on HTRs.

This experimental work uses the FZJ thermo-gravimetric facility THERA and the induction furnace facility INDEX (see Fig. 8). Graphite burning under severe air ingress accidents, is usually assessed using computer models, based on isothermally measured kinetic equations that consider in-pore diffusion and chemical reaction mechanisms. The data requirements for such models are burn off dependent chemical (regime 1) reactivities which occur at lower temperatures whereas for intermediate temperatures the consumption of oxidising gas within the pores becomes important (regime II) and the oxidation attack becomes smaller in the depth of the material. At high temperatures external mass transfer is important and the reaction is restricted to the geometrical surface only. THERA was used for the regime 1 and INDEX for regime II where higher flow rates are needed. Long-term experiments at low temperature in air were also performed in an annealing furnace.

The second series of experiments looked at the performance of CFC materials under different temperatures in steam and air. Results in steam at 900°C are shown in figure 9. Oxidised gas flowed through the inner bore hole of the tube with flow rates sufficiently high to suppress the influence of boundary layer mass transfer on kinetics. Oxidation effects were measured by mass spectrometric analysis of the product gases and by weight loss of the sample. Experiments were also performed to determine properties of CFC materials irradiated and non-irradiated (up to 1dpa). These include measurements of strength, dimensional change, thermal diffusivity and heat capacity.

The main findings from this graphite oxidation work are summarised as follows.

- The un-graphitisied binder material has to be treated as a separate component because of its remarkably different oxidation behaviour.
- Despite its Si-content, the oxidation resistance of NS31 is not remarkably better than that of other examined materials (but a factor of 2 better than the case without Si).
- Both the amount of Si and its distribution within the material determines its ability to protect against oxidation.
- The oxidation behaviour of homogeneous materials in regime 1 is characterised by a continuous rate verses burnoff curve with one (more or less) pronounced maximum.
- The shape of the rate vs. burn-off curves are roughly similar in air and steam however the rates differ markedly. For regime 1: for a similar rate in air at 600 – 650 °C, temperatures are required to be 250°C higher in steam (900°C)
- Inhibition by hydrogen / CO remains to be measured for steam oxidation.

HTR-M 02/11-D-3.2.53	HTR-M 02/11 D 3 3 56	

(e) Synthesis of graphite work

An overall summary document on the results of Work Package 3 has been issued and placed on the SINTER web site.

HTR-M 02/12 D 3 0 58		

2.5 Work Package 2 of HTR-M1 – Graphite Irradiation Tests

The specific tasks involved in this work package are as follows.

Task 1 Graphite Irradiation tests
Task 2 Graphite with SiC coatings

(a) Graphite Irradiation tests

For the HTR-M1 programme three main graphite manufacturers were approached to see which graphites could be offered for the next generations of reactors. Five graphites were selected and identified for the programme, two iso-molded and two extruded graphites, manufactured from pitch coke and petroleum coke. A range of grain sizes (1mm down to 10 µm) was also selected. IG100 (HTTR graphite) was also chosen (iso-moulded graphite made using a petroleum coke) since this has previously been irradiated at high temperature, although only to a small/medium fluence. Nevertheless, the available data should provide a useful comparison with the data obtained from the new graphites.

The irradiation experiment uses a test rig that allows distinct temperature levels to be obtained. Irradiation temperatures are higher than 10 years ago and there are three temperatures of major interest (600^{o}C, 750^{o}C and 900^{o}C). A temperature of 750^{o}C, was considered to be most relevant for new designs and therefore adopted for the HTR-M1 tests. The rig design will require the stacking of up to 150 samples, 8mm diameter (either 6 or 12mm in height), covering two main directions (15 samples per grade per direction).

Characterisation of the graphites will be done at facilities at NRG. These are to cover dimensional change, CTE (coefficient of thermal expansion), dynamic Young's modulus and thermal diffusivity. For CTE a commercial dilatometer is to be used.

Although five (major) grades of graphite have been selected for full testing in the experiment, it was decided that it would be useful to have an early indication of the irradiation behaviour of some of the other proposed graphites. These could be included as 'piggy-back' samples in the irradiation capsule. The (minor) graphites selected are UCAR grade PPEA, SGL grade NBG-20 and Toyo Tanso grade IG-430.

The irradiation test provides a number of data points within a single test run by using the flux (buckling) distribution available in the HFR capsule. A third order polynomial can describe the behaviour and a sufficient number of points should be obtained from the test to describe the distribution.

Ideally the test conditions for the experiment are those that give data for candidate graphites up the peak end of life fluence. However this requires a significant effort over a long time scale, typically 10 years. The target flux in the current four-year programme is as high as possible. The current available locations in the HFR suggest an irradiation for 1-1.5 year to EDN fluence ~6-7 x 10^{25} m^{-2} (approx. 8 dpa). This is not expected to yield properties up to turn round behaviour but will provide information on whether the graphite is useful or not. The current framework therefore provides a first sorting and as the experiment progresses there is the possibility of replacing those that are unsuitable with alternatives. For the highest flux position in HFR a total of 3 years will be needed for the irradiation. The Irradiation is planned to start in the first part of 2004 with first PIE in the spring of 2005. The intention is to continue the test within the context of the 6th Framework programme and complete the target flux irradiation.

HTR-M1 03/03 D 2 1 9	HTR-M1 ¾ D 2 2 13	HTR-M1 03/09 D 2 5 14

(b) Graphite with SiC coating

Work carried out in Germany to develop SiC coatings for graphite materials for protecting the graphite against oxidation from air or steam ingress under severe accidents has been reported. Investigations were done using the matrix German graphite A3-3 to measure corrosion rate for temperatures up to 1200^{o}C, and on coated pebbles using different graphites. Most of the coating experiments used the Japanese graphite IG 110 adopted for the structures and fuel blocks of the HTTR. Corrosion tests, irradiation tests plus PIE were carried out to assess coating integrity and different coating methods (Chemical vapour deposition (CVD), Paste Siliconisation (PS)). A very good corrosion resistant quality was eventually achieved with IG 110 and V438 using CVD. No damage or weight loss was observed after being irradiated and corroded and dropped from a height of 50 cm. Although the original objective, to develop a coating for A3-3 graphite, has not yet been achieved, the results achieved so far are a very positive indicator and have attracted considerable interest Internationally (from USNRC).

HTR-M1 03/09 D 2 5 14		

3 Discussion

3.1 Overall Progress

Overall the project is progressing well and is expected to be largely completed within the scheduled programme finish dates. The main emphasis of the work is currently on starting the two irradiation experiements and chemical treatments which are an important first step before some of the last phase of the testing can begin.

<u>Vessel</u>

The review work is essentially finished and the database is currently being assembled. Initial delays in the procurement/fabrication of the vessel material have used up the available flexibility in the programme with respect to the testing. The start of the irradiation in HFR is now scheduled for December 2003 with PIE tests to commence from February 2004.

There are 3 shielded creep machines reserved for the irradiated testing which means that for tests at 4 conditions (approx 100, 300, 1000, 3000h = ~ 4500h) a testing period of 250 days will be needed. For a start of testing in February 2004 and everything running as predicted (rupture times not too long) then the test should finish before November 2004. However, given that the schedule will be very tight creep testing beyond November 2004 is possible. For the other tests: tensile, fracture, these toughness and impact (except full-size charpy) will finish before November 2004.

<u>Turbine & Control Rod</u>

Again the review work is largely finished and the database is being assembled. The tests on turbine materials in air have been completed and are being reported. Delays in the conditioning of the turbine materials has prevented the start of some of the mechanical tests on the disc and blade materials. The short-term tests within HTR-M programme should however be accommodated within the overall programme envelope. The intermediate creep tests planned within the HTR-M1 programme may extend up to the end of the programme. Any extension beyond the scheduled finish of the HTR-M1 programme will become clearer following the completion of the carburisation & decarburisation treatments. These treatments are due to start at the end of 2003.

<u>Graphite</u>

The graphite review and oxidation experiments of HTR-M have concluded as planned. The characterisation of the graphite specimens for the irradiation tests of HTR-M1 have been completed. Commencement of the experiments is expected to take place as soon as the test facilities are available. The irradiation of the graphite test pieces in the HFR is expected to begin towards the end of December 2003 with PIE in spring 2005.

3.2 Overall Scientific value

The technical tasks addressed by the project are on key components and deal with issues that are at the forefront of understanding with respect to materials behaviour and HTR application. It is expected that in all three areas, the findings from this work will have an important impact on the final materials selection and development for future HTR and VHTR modular designs.

<u>Vessel</u>

The need for reliable material property information is a key issue in the development of any innovative reactor technology. For the HTR it's especially important in view of the strong impact the environment has on materials behaviour. The review and database activity forms an important foundation of information for future developments of the HTR and VHTR modular vessel designs. The synthesis focuses on the needs of the two vessel options being considered for the latest industrial prototypes: the 'cold' (cooled) vessel design concept adopted for the PBMR (LWR type steels); the 'warm' (non-cooled) vessel concept adopted for the GT-MHR (modified 9Cr1Mo steel). The synthesis confirms the potential of both options and identifies important issues that need to be addressed in their development.

The testing looks at short-term materials strength values for a modified 9Cr 1Mo steel vessel weld manufactured with the narrow gap TIG process being considered for the HTR and VHTR vessel. These manufactured vessel weld samples are the first trials in this material (at such thicknesses) and therefore break new ground with respect to manufacturing and research and development in this area. The tests are performed with and without irradiation. No previous tests have been performed under irradiated conditions for representative vessel welds manufactured with this material. It is expected that the testing will provide the first important information on the level of deterioration in properties under representative HTR conditions.

<u>Turbine & Control Rod</u>

The review and database work carried out for the control rod materials (largely carbon-carbon (C-C) composite options) and the turbine (disc and blade) underlines the importance of achieving the required creep strength at temperature and making due allowance for environmental impact. For the control rod the advantages of using carbon based materials over metallic materials in the high temperature and high irradiation environment needs substantiation. This first phase of work on C-C composites purposely involves setting up close links and co-operation with manufacturers and scientific experts to develop this technology further. The turbine review work confirms that there is no suitable material available that can be used for a non-cooled disc and that a significant manufacturing development is required if a large diameter (1.5m) disc is to be adopted (i.e. for GTMHR). For blade materials cooling may also be necessary plus a coating to protect against corrosion and damage.

Impurities in the helium at temperature can result in carburisation and/or de-carburisation of the super-alloys and lead to a worsening of material strength. The best disc material (U720) plus two blade materials (Al_20_3 former and Cr0 former) were chosen for the test work as the precise atmosphere was not known. This short and medium term test work following exposure to treated atmospheres should provide an overall indication of the relative bounds on strength such carburising and decarburising effects can have. If shown to be significant there may be important implications with respect to environmental control and acceptable levels of impurity levels in the reactor helium gas.

Graphite

Graphite behaviour has important safety implications because of structural and property changes that occur when it is irradiated. Information on behaviour is crucial and design/material property data under HTR relevant conditions is needed for candidate materials for both normal and accident conditions. Many of the graphites used in previous core designs are no longer available and there has been a serious decline in the ability to manufacture nuclear grade graphite in large quantities. In the past, only a small number of irradiation programmes have looked at the effects of irradiation on the behaviour of graphites at high temperatures, i.e. >600°C. The result is that for the new HTRs and VHTRs a new graphite is required that will need significant development (irradiation tests) and require global sharing of information through international cooperation.

A test programme has been initiated on candidate graphites in the High Flux Reactor at Petten. The materials tested have been chosen in full co-operation with graphite manufacturers and are therefore considered the best potential materials available. This irradiation looks at physical property changes at one temperature (750^{o}C) and requires continuation in the next framework programme to complete the information up to 'turn-round'. More information is needed (at higher temperature) also model development initiatives towards developing a predictive methodology. The first stage PIE after 1-1.5 years will enable some initial sorting of a possible candidate to be done, however the major impact will come at the end of the next framework when the full curve behaviour can be described.

4 Conclusions

The objective of the HTR-M project is to improve knowledge of materials for future HTRs for the reactor pressure vessel, the control rod and turbine (blades & discs) and the reactor graphite. The scientific and technical progress for the HTR-M project has been described and issues that are likely to influence the technical decisions on the next stage of the programme reviewed. The review and database work is largely completed and testing is in progress with the main test on the vessel and graphite irradiations set to commence at the end of 2003.

The vessel review work has focused on materials used in new and developing plants and identified modified 9Cr 1Mo steel for the irradiation tests. The database templates are prepared and data collection is underway for the nominated steels. The pre and post irradiation test programmes are largely finalised and fabrication and machining of the test pieces largely complete. The specimens are to be loaded into the Lyra Test rig in the next few weeks for commencing irradiation in the pool area of HFR.

The review and database work on high temperature materials relevant to the control rod and turbine disc and blade materials are well advanced. The database templates are prepared and data collection is underway for the nominated steels. The review and database activity show C-C Composite materials as a higher temperature alternative to metals for the control rod. For the turbine (disc and blades) following discussions involving Turbine materials manufacturer Aubert and Duval candidate alloys have been selected for the test work. Material has been procured and machined and a number of specimens are to be exposed to carburising and decarburising atmospheres before testing. This pre-treatment of the specimens is expected to be completed by the end of 2003. Short term and medium term tests in air have been carried out. Intermediate creep testing of material is planned to start in 2004 under the HRT-M1 project.

The work on the graphite property side has concluded with a series of reports produced consisting of a review and database, graphite selection, and oxidation tests on graphite and C-C composite materials carried out at FZJ using hot air and steam. The oxidation test work was concluded at the end of 2002. Experiments to determine the thermal conductivity of CFCs have also been made in non-irradiated material. Three principle graphite manufacturers and suppliers have collaborated closely with the HTR-M1 project and supplied new graphites for irradiation testing in HFR. Following selection, procurement and machining of the graphite samples, assembly of the test rig and mounting of the specimens is underway for placing in the core of HFR with the irradiation test expected to begin at the beginning of 2004 and PIE in spring 2005.

Overall the project is still within its overall timescale but because of delays in procurement, fabrication and environmental treatment some of the creep tests are expected to extend up to the end of the project. The main emphasis for the last year of the HTR-M project and 18 months of the HTR-M1 project is to assemble the remaining database information, conclude the experimental work on the vessel steel and turbine materials and for HTR-M1 to ensure the start of the graphite irradiation tests and intermediate creep tests as early as practical.

5 References

Ref	Title
1	D. E Buckthorpe European Project for the development of HTR Technology Minutes of HTR-M plus HTR-M1 Progress Meeting Report No. HTR-M-02/5-M-0-0-33
2	D. E Buckthorpe European Project for the development of HTR Technology Minutes of HTR-M plus HTR-M1 Mid Term Meeting Report No. HTR-M-02/11-M-0-0-47
3	D.E. Buckthorpe, et al. European Project for the development of HTR Technology Minutes of HTR-M plus HTR-M1 Progress Meeting Report No. HTR-M-03/7-M-0-0-74
4	D. E Buckthorpe European Project for the development of HTR Technology Minutes of HTR-M plus HTR-M1 Progress Meeting Report No. HTR-M-03/12-M-0-0-83

Table 1 HTR-M Deliverables list

Table 1	**HTR-M Deliverables list (WP1)**				
Deliverable No	**Deliverable title**	**Month due**	**Current planned**	**Date Achieved**	**Main Organisation**
1.1	Database on pressure vessel steel plate and weld material for reactor pressure vessel (RPV)	24	01/11/2002	24	NNC
1.2	State of the art on HTR vessel materials	24	01/11/2002	25	NNC
1.3	Recommendations for RPV design, choice of material, defect assessment and leak before break methodology	24	01/11/2002	25	CEA /EA
1.4	Evaluation of factors for defect tolerance of RPV weld	42	01/05/2004	-	FRA
1.5	Complementary irradiation data on base material and weld	42	01/05/2004	-	NRG
1.6	Summary report on vessel materials	48	01/11/2004	-	NRG

Table 1.	**HTR-M Deliverables list (WP2)**				
Deliverable No	**Deliverable title**	**Delivery date**	**Current planned**	**Date Achieved**	**Main Organisation**
2.1	***Materials for core and internals*** : Report on design specifications and operating conditions by partner 2	9	01/07/2001	9	FRA
2.2	***Materials for core and internals***: Synthesis of materials database	18	01/05/2002	19	NNC
2.3	***Report on past experience from FZJ (revised title)***	37 (revised)	22/12/2003		CEA
2.4	***Report on discussions with manufacturers (revised title)***	41 (revised)	01/04/2004	-	CEA/JRC/ NNC
2.5	***Materials for core and internals***: Final report on material grade and processing, synthesis of further R&D efforts needed.	48	01/11/2004	-	CEA
2.6	***Materials for turbine*** : Report on design specifications and operating conditions	9	01/07/2001	11	FRA
2.7	***Materials for turbine*** : Synthesis of materials database	18	01/05/2002	19	NNC
2.8	***Materials for turbine*** : Definition of complementary testing experiments	18	01/05/2002	19	CEA/JRC
2.9	***Materials for turbine*** : Report on preliminary material testing and progress on material cyclic tests	36	01/11/2003	37	CEA/JRC
2.10	***Materials for turbine***: Final report on material grade and processing and material tests, synthesis of further R&D efforts needed.	48	01/11/2004	-	CEA

Table 1.	HTR-M Deliverables list (WP3)				
Deliverable No	**Deliverable title**	**Delivery date**	**Current planned**	**Date Achieved**	**Main Organisation**
3.1	State of the art review of graphite materials and behaviour	18	01/05/2002	19	NNC/FZJ
3.2	Report on database of properties of graphite suitable for use by the designer	18	01/05/2002	19	NNC
3.3	Assess new potential graphites and needs of HTR	24	01/11/2002	25	NNC/FZJ
3.4	Report on chemical kinetic data for advanced oxidation models	24	01/11/2002	25	FZJ
3.5	Report on oxidation resistance of advanced C-based materials	24	01/11/2002	25	FZJ
3.6	Final Report Summary	24	01/11/2002	25	NNC

Table 1.	HTR-M1 Deliverables list (WP1)				
Deliverable No	**Deliverable title**	**Delivery date**	**Current planned**	**Date Achieved**	**Main Organisation**
D1	Report on selection of materials & testing requirements	12	01/10/2002	11	CEA
D2	Report on specimen fabrication and procurement & testing requirements	18	01/04/2003	17	CEA
D3	Report on material testing	42	01/04/2005	-	CEA
D4	Damage analysis and modelling	42	01/04/2005	-	CEA
D5	Review of results and updating of database	42	01/04/2005	-	CEA
D6	Final Report summary	42	01/04/2005	-	FRA

Table 1.	HTR-M1 Deliverables list (WP2)				
Deliverable No	**Deliverable title**	**Delivery date**	**Current planned**	**Date Achieved**	**Main Organisation**
D7	Report on selection of graphites and testing requirements	12	01/10/2002	14	NNC
D8	Report on specimen fabrication and procurement	18	01/04/2003	18	NRG
D9	Report on irradiation and post irradiation tests	42	01/04/2005	-	NRG
D10	Review of results and updating of database	42	01/04/2005	-	NRG
D11	Assessment of results and suitability	42	01/04/2005	-	FZJ
D12	Final Report Summary	42	01/04/2005	-	NNC

Table 2 HTR-M Project document list

Table 2 HTR-M Project document list				
HTR No.	**Status**	**Final issue date**	**Title**	**Author / Org**
HTR-M-01/3-D-0-0-1	Final	March 2001	HTR-M Release Data	T.Austin / JRC
HTR-M-01/3-D-0-0-2	Draft		On-line Alloys-DB Access Agreement	T. Austin / JRC
HTR-M-01/3-D-0-0-3	Mid Term Draft		Technological Implementation Plan	D. Buckthorpe / NNC
HTR-M-01/3-M-0-0-4	Final	April 2001	Minutes of HTR-M Kick-off Meeting, held on 21/22 November 2000 at NNC Ltd., UK	K.Peers/ NNC
HTR-M-01/5-M-0-0-5	Final	Dec 2001	Minutes of HTR-M First Progress Meeting held on 18/19 April 2001 at NRG, Petten, Netherlands	R.D.W.Bestwick/ NNC
HTR-M-01/5-D-0-0-6	Final	May 2001	Materials for the High Temperature Reactor, SMiRT 16 paper No. 1941, Div S. Washington, U.S.A, Aug. 2001	D. Buckthorpe/ NNC et. al.
HTR-M-01/5-P-0-0-7	Final	July 2001	HTR-M Six Month Management Report	D. Buckthorpe/ NNC
HTR-M-01/5-D-0-0-8	Final	July 2001	Summary of technical work done over first Six Months of HTR-M	D. Buckthorpe/ NNC; et. al.
HTR-M-01/9-D-1-1-9	Draft		HTR-M project - Preliminary review of RPV materials used on HTR and other gas reactors	R.D.W.Bestwick / NNC
HTR-M-01/9-D-2-0-10	Draft		HTR-M Work Package 2 - Detailed description of tasks	R. Couturier / CEA
HTR-M-01/9-D-2-1-11	Final	Inserted Sept. 2001	Development of carbon/carbon composite control rod for HTTR (II) - Concept, specification and mechanical test of materials	JAERI Research 98-003/ Japan
HTR-M-01/9-M-2-0-12	Draft		HTR-M WP2, Milestone 2 Materials review and database	R. Couturier/ CEA
HTR-M-01/9-M-2-0-13	Draft		Bibliography of papers for HTR-M Work Package 2	R. Couturier/ CEA
HTR-M-01/7-D-2-1-14	Draft		HTR-M Design use and Specification for Internals	B. Riou/ Framatome
HTR-M-01/7-D-1-0-15	Draft		HTR-M Work Package 1 - Detailed description of tasks	H. Hegeman & B van der Schaaf/NRG
HTR-M-01/10-D-0-0-16	Final	Oct. 2001	Investigation of High Temperature Reactor (HTR) Materials, OECD paper, 2nd Information exchange meeting of High Temperature Engineering, Paris, 11-12 Oct 2001	D. Buckthorpe/ NNC Ltd. et. al.

Table 2 (cont)			HTR-M Project document list	
HTR No.	**Status**	**Final issue date**	**Title**	**Author / Org**
HTR-M -01/10-D-1-1-17	Draft		Material selection for HTR Pressure Vessel	A. Buenaventura/ EA
HTR-M 01/11 D.1.1.18	Draft		Report on Damage Mechanisms	M.T.Cabrillat/CEA
HTR-M 01/10 D-2-1-19	Draft		HTR-M Design use and Specification for Turbine	B. Riou/ Framatome
HTR-M 01/10 D-3-0-20	Draft		HTR-M Work Package 3 - Description of Tasks	D. Buckthorpe/ NNC
HTR-M 01/3 D.0.0.21	Draft		HTR-M Materials Database - scope and proposed stages of development	D. Buckthope NNC
HTR-M 01/11 D.0.0.22	Draft		Scheme for HTR-M Materials data collection	D. Buckthorpe NNC
HTR-M 01/3 D.2.0.23	Draft		Information needs on "Turbine design and operating conditions" for the choice of high temperature materials and experimental test conditions	R. Couturier CEA
HTR-M 01/11 P-0-0-24	Final	Dec 2001	HTR-M First Annual Report	D. Buckthorpe/ NNC
HTR-M 01/11 P-0-0-25	Final	Dec 2001	HTR-M Twelve month management report	D. Buckthorpe/ NNC
HTR-M 02/01 M.0.0.26	Final	April 2002	Minutes of the HTR-M1 Kick-off plus 2nd HTR-M Progress Meeting on 5-7 December 2001, Grenoble, France	D. E. Buckthorpe / NNC
HTR-M 02/02 D.1.1.27	Draft		HTR RPV Materials Damage Mechanisms	A. Buenaventura EA
HTR-M 02/02 D.1.1.28	Draft		HTR RPV Sensitive Zones	A. Buenaventura EA
HTR-M 02/03 D.0.0.29	Final	April 2002	Materials for the High Temperature Reactor, HTR2002 Conference, Petten, 22-24th April 2002 (poster paper)	D. E. Buckthorpe /NNC Et.al.
HTR-M 02/03 D.0.0.30	Final	Oct 2002	High Temperature Reactor (HTR) Materials, Liege Conference, 30th Sept. to 2nd Oct., 2002 (poster paper)	D. E. Buckthorpe / NNC Et.al.
HTR-M 02/04 D 1.1.31	Draft		HTR - Identification and review of Pressure vessel candidate materials	C. Escaravage / Framatome-ANP
HTR-M 02/04 D 2.2.32	Draft		HTR: Identification and review of Turbine Candidate materials	C. Escaravage / Framatome-ANP
HTR-M 02/5 M 0.0.33	Final	Nov 2002	Minutes of the HTR-M / M1 Progress Meeting on 15-17 April; 2002, Julich, Germany	D. E. Buckthorpe NNC Ltd.
HTR-M-02/5-P-0-0-34	Final	July 2002	18month HTR-M plus 6 month HTR-M1 joint management report	D. E. Buckthorpe NNC Ltd.
HTR-M 02/6 D 2 2 35	Draft		Materials for Turbine: Definition of Complementary testing experiments	R. Couturier / CEA

Table 2 (cont) HTR-M Project document list

HTR No.	Status	Final issue date	Title	Author / Org
HTR-M 02/6 D 2 1 36	Draft		Materials for the Core and Internals: Synthesis of Materials Database	D. E. Buckthorpe & C. J. Earl/ NNC
HTR-M 02/6 D 3 1 37	Draft		State of Art review of graphite materials behaviour	M.W. Davies / NNC G Haag / FZJ
HTR-M 02/6 D 3 1 38	Draft		Report on database of properties suitable for use by designer	M.W. Davies / NNC
HTR-M 02/7 D 2 1 39	Draft		Materials for turbine: synthesis of materials database	D.E.Buckthorpe & C.J.Earl/ NNC
HTR-M 02/8 D 1 1 40	Draft		Recommendations for RPV design, choice of material, defect assessment and leak before break methodology	M.T.Cabrillat / CEA A. Buenaventura / EA
HTR-M 02/9 D 1 1 41	Draft		HTR Compared properties of Pressure Vessel candidate materials	C. Escaravage / Framatome ANP
HTR-M 02/10 D 0 0 42	Final	Oct. 2002	Design Criteria for High Temperature Metallic and Ceramic Components as well as Pre-Stressed Concrete Reactor Pressure Vessels for Future HTR Plants - Volume I, Part A: Safety technological boundary conditions.	(In German) Provided for the Project by FZJ
HTR-M 02/10 D 0 0 43	Final	Oct. 2002	Design Criteria for High Temperature Metallic and Ceramic Components as well as Pre-Stressed Concrete Reactor Pressure Vessels for Future HTR Plants - Volume IIa, Part B: Metallic Components.(B1, B2, B10)	(In German) Provided for the Project by FZJ
HTR-M 02/10 D 0 0 44	Final	Oct. 2002	Design Criteria for High Temperature Metallic and Ceramic Components as well as Pre-Stressed Concrete Reactor Pressure Vessels for Future HTR Plants - Volume IIb, Part B: Metallic Components.(B3 to B9)	(In German) Provided for the Project by FZJ
HTR-M 02/10 D 0 0 45	Final	Oct. 2002	Design Criteria for High Temperature Metallic and Ceramic Components as well as Pre-Stressed Concrete Reactor Pressure Vessels for Future HTR Plants – Volume IV, Part D: Graphite Reactor Internals.	(In German) Provided for the Project by FZJ

Table 2 (cont)			HTR-M Project document list	
HTR No.	**Status**	**Final issue date**	**Title**	**Author / Org**
HTR-M 02/10 D 0.0 46	Final	Oct 2002	European Project on Materials for Modular HTRs, ENC2002 Conference, Lille, France, 7-9th Oct., 2002	D. E. Buckthorpe / NNC Et.al.
HTR-M 02/11 P 0.0 47	Final	Jan 2003	Mid-term report for the HTR-M & M1 Projects	D. E. Buckthorpe NNC Ltd
HTR-M 02/11 D 1 1 48	Draft		Review of irradiation and thermal ageing - effects in modified 9Cr 1Mo steel	A. Braithwaite / NNC Ltd.
HTR-M 02/11 D 1 1 49	Draft		State of the art review of HTR materials	R.D.W.Bestwick/ NNC Ltd.
HTR-M 02/11 D 0 0 50	Draft		The HTR-M vessel materials database	A. Braithwaite / NNC Ltd
HTR-M 02/11 D 0 0 51	Draft		HTR-M vessel materials database Nov 02	A. Braithwaite / NNC Ltd
HTR-M 02/11 D 2 0 52	Draft		WP6 Review of the gas/Ni based alloy interactions in HTR impure Helium	P. Combrade/ Framatome.ANP SAS
HTR-M 02/11 D 3 2 53	Draft		Assess new potential graphites and needs of HTR	M. W. Davies/ NNC Ltd.
HTR-M 02/12 D 2 0 54	Draft		HTR-E WP1 Turbine – HTR recommendations of cobalt content in turbine materials	C. Mauget / Framatome.ANP
HTR-M 02/12 D 3 3 55	Draft		Oxidation resistance of Carbon based materials	K. Kuhn, H-K Hinssen, R. Moormann / FZJ
HTR-M 02/12 D 3 3 56	Draft		Chemical kinetic data for advanced oxidation models	K. Kuhn, H-K Hinssen, R. Moormann / FZJ
HTR-M 02/12 P 0 0 57	Draft		24 Month HTR-M plus 12 month HTR-M1 Management Report	D. E. Buckthorpe NNC Ltd
HTR-M 02/12 D 3 0 58	Draft		Summary report – work performed on graphite within the HTR-M project	D. E. Buckthorpe NNC Ltd
HTR-M 02/12 P 0 0 59	Draft		24 Month HTR-M plus 12 month HTR-M1 Management Report	D. E. Buckthorpe NNC Ltd
HTR-M 03/01 D 2 2 60	Final		Information on Udimet 720	Special Metal Corporation, 1988

Table 2 (cont)			HTR-M Project document list	
HTR No.	**Status**	**Final issue date**	**Title**	**Author / Org**
HTR-M 03/01 D 2 2 61		Oct 2002	Materials Selection for HTR Power Turbine	A. Buenaventura/ EA
HTR-M 03/01 D 2 0 62	Draft		Testing and Investigations of Materials for Turbocompressor High-Temperature Components	Project Manager – Romantsov Aleksey Alekseyevich / OKBM
HTR-M 03/02 D 1 4 63	Draft		Specification of the test matrix for the RPV test programme	J. Hegeman / NRG
HTR-M 03/04 D 0 0 64	Final	May 2003	Research and Development Programme on HTR Materials. (Paper 3174). - ICAPP 03 Conference, Cordoba, Spain, 4-7 May, 2003.	D.E. Buckthorpe, et al
HTR-M 03/04 D 0 0 65	Final	May 2003	HTR RPV Materials. (Paper 3355). ICAPP 03 Conference, Cordoba, Spain, 4-7 May, 2003	A. Buenaventura/ EA
HTR-M 03/04 D 0 0 66	Final	May 2003	Behaviour of C-Based Materials in contact to oxidising gases. (Paper 3031). - ICAPP 03 Conference, Cordoba, Spain, 4-7 May, 2003	K. Kuhn, H. Hinssen, R. Moormann/ FZJ
HTR-M 03/04 D 0 0 67	Final	August 2003	Progress of HTR-M projects for the HTR. (Paper S443). - 17th SMiRT Conference, Prague, Czech Republic, 12-17th August 2003	D.E. Buckthorpe, et al.
HTR-M 03/04 D 0 0 68	Final		HTR Materials & Technologies, Review of Metallic Materials used in HTRs - HTR Eurocourse, CEA, Cadarache, 4-8 November 2002.	D.E. Buckthorpe/ NNC
HTR-M 03/04 D 0 0 69	Draft		Thermal Conductivity of Graphite Irradiated with Medium and High Neutron Fluences	G. Haag/ FZJ
HTR-M 03/05 D 1 1 70	Draft		Effect of Environment on the material of GT-MHR Vessel	M. Cabrillat/ CEA
HTR-M 03/05 P 0 0 71	Final	July 2003	30 Month HTR-M plus 18 month HTR-M1 Management Report	D. Buckthorpe/ NNC
HTR-M 03/07 M 2 0 72	Final		Notes of a HTR-M & M1 Specialist Meeting on Corrosion & Carburisation	H. Rantala /JRC
HTR-M 03/07 M 0 0 73	Draft		Minutes of 2nd Technical Meeting at Petten on HTR-M & M1 Irradiation experiments	D. Buckthorpe/ NNC

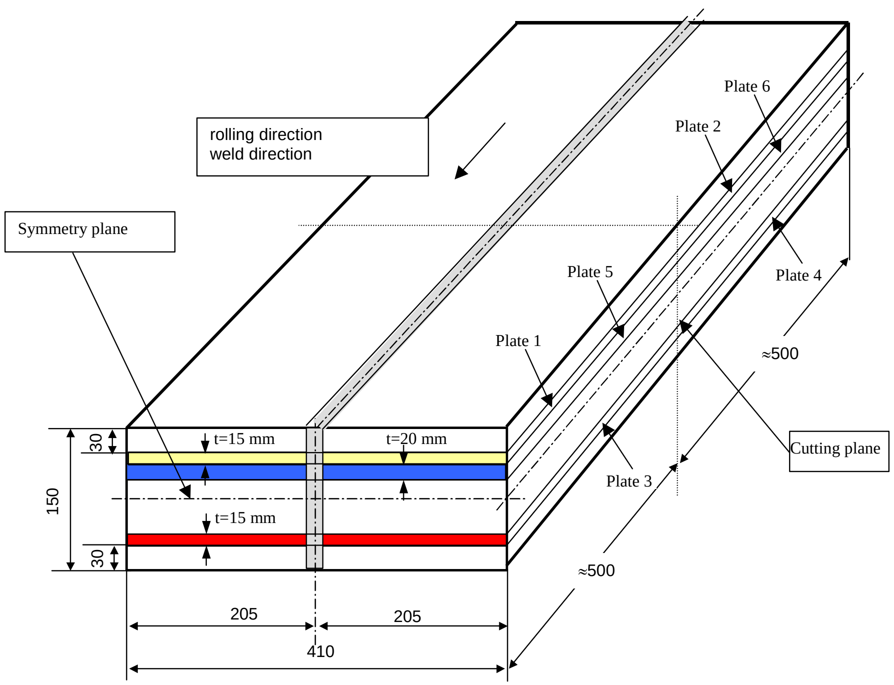
Table 2 (cont)			HTR-M Project document list	
HTR No.	**Status**	**Final issue date**	**Title**	**Author / Org**
HTR-M 03/07 M 0 0 74	Final	Nov 2003	Minutes of the 5th HTR-M plus M1 Progress Meeting	D. Buckthorpe, R. Bestwick, A. A. Braithwaite / NNC
HTR-M 03/08 D 0 0 75	Draft		HTR-M metal materials database (BLANK)	A.A. Braithewaite / NNC Ltd
HTR-M 03/08 D 0 0 76	Draft		HTR-M C-Based materials database (BLANK)	A. Keenan/ NNC Ltd
HTR-M 03/08 D 0 0 77	Draft		HTR-M Graphite materials database (BLANK)	M. Davies/ NNC Ltd
HTR-M 03/08 D 1 4 78	Draft		Proposed creep rupture test conditions for the HTR-M irradiation experiment on a modified 9Cr 1Mo steel weld	R. D. W. Bestwick/ NNC Ltd
HTR-M 03/10 D 0 0 79	Final	Sept. 2003	High Temperature Reactor Materials: Progress of HTR-M Projects, NEA/OECD, 3rd Information Exchange meeting on High Temperature Engineering, JAERI, Ibaraki-ken, Japan, 11-12 September 2003	D.E. Buckthorpe / NNC, et al.
HTR-M 03/11 P 0 0 80	Draft		36 Month HTR-M plus 24 month HTR-M1 Management Report	D. E. Buckthorpe/ NNC
HTR-M 03/11 P 0 0 81	Draft		Third Annual Technical Report for the HTR-M project	D.E. Buckthorpe / NNC, et al.

Table 2 (cont)		HTR-M1 Project document list		
HTR No.	**Status**	**Final issue date**	**Title**	**Author / Org**
HTR-M1 02/1 M 2.0.1	Final	April 2002	Minutes of 1st Technical meeting on graphite held on 13th Nov. 2001 at NRG Petten, Netherlands	J. van der Laan/ NRG Petten
HTR-M1 02/1 M.0.0.2	Final	April 2002	Minutes of the HTR-M1 Kick-off plus 2nd HTR-M Progress Meeting on 5-7 December 2001, Grenoble	D.E. Buckthorpe/ NNC Ltd.
HTR-M1 02/5 M 2.0.3	Final	July 2002	Minutes of meeting with graphite manufacturer UCAR on 26th. March 2002, Notre Dame, France	M. W. Davies/ NNC Ltd
HTR-M1 02/5 M 2.0.4	Final	July 2002	Minutes of meeting with graphite manufacturer SGL on 27th March 2002, Chedde, France	M. W. Davies/ NNC Ltd
HTR-M1 02/5 M 0.0.5	Final	Nov 2002	Minutes of the HTR-M / M1 Progress Meeting on 15-17 April 2002, Julich, Germany	D.E. Buckthorpe/ NNC Ltd.
HTR-M1 02/5 P 0.0.6	Final	July 2002	18month HTR-M plus 6 month HTR-M1 joint management report	D. E. Buckthorpe NNC Ltd.
HTR-M1 02/9 D 1 1 7	Draft		HTR-M1: Report on selection of materials & testing requirements for the Turbine	R. Couturier / CEA
HTR-M1 02/11 P 0 0 8	Final		Mid-term report for the HTR-M & M1 Projects	D. E. Buckthorpe NNC Ltd
HTR-M1 02/11 D 2 1 9	Draft		Report on selection of graphites and testing requirements	M. W. Davies/ NNC Ltd
HTR-M1 02/12 M 0 0 10	Final		Minutes of the HTR-M plus M1 Mid Term Meeting on 28-29 November; 2002, Framatome, Paris	D. E. Buckthorpe NNC Ltd
HTR-M1 02/12 P 0 0 11	Final		24 Month HTR-M plus 12 month HTR-M1 Management Report	D. E. Buckthorpe NNC Ltd
HTR-M1 03/03 D 1 0 12	Draft		Report on specimen fabrication, procurement & testing requirements for the turbine blade material	R. Couturier / CEA
HTR-M1 03/04 D 2 2 13	Draft		HTR-M1 Report on specimen fabrication and procurement for the graphite irradiation tests	A. Vreeling / NRG
HTR-M1 03/09 D 2 5 14	Draft		Graphite with Si C coatings	B. Schröder/ FZJ
HTR-M1 03/06 P 0 0 15	Final	July 2003	30 Month HTR-M plus 18 month HTR-M1 Management Report	D.E.Buckthorpe/N NC
HTR-M1 03/07 M 0 0 16	Draft		Minutes of 2nd Technical Meeting at Petten on HTR-M & M1 Irradiation experiments	D.E.Buckthorpe/N NC
HTR-M1 03/07 M 0 0 17	Final	Nov. 2003	Minutes of the 5th HTR-M plus M1 Progress Meeting	D. Buckthorpe, R. Bestwick, A. A. Braithwaite / NNC

Table 2 (cont)			HTR-M1 Project document list	
HTR No.	**Status**	**Final issue date**	**Title**	**Author / Org**
HTR-M1 03/11 P 0 0 18	Draft		36 Month HTR-M plus 24 month HTR-M1 Management Report	D. E. Buckthorpe/ NNC
HTR-M1 03/11 P 0 0 19	Draft		Third Annual Technical Report for the HTR-M project	D.E. Buckthorpe / NNC, et al.
HTR-M1 03/12 M 0 0 20	Draft		Minutes of HTR-M / M1 Progress Meeting at FZJ, Germany, 18-19 November 2003	D. E. Buckthorpe/ NNC

Figure 1 Vessel welded Joint

Typical cutting plane of weld joint



Microcracks observed in welded joint

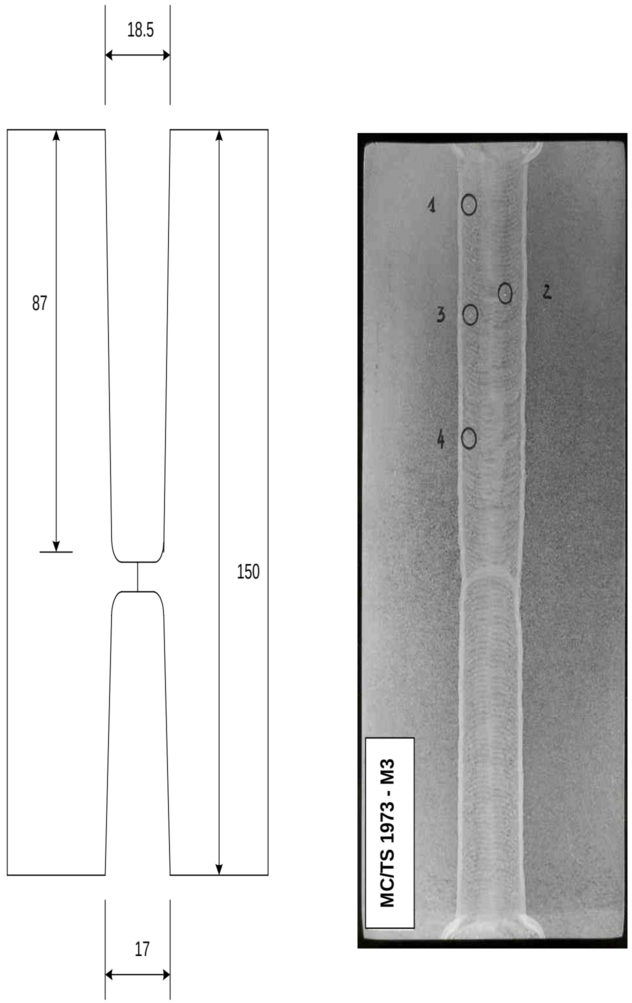


Figure 2 Loading of Vessel weld test specimens

- Loading of LYRA Test Rig
- *Position of each specimen*
- *Flux monitors*
- *Temperature monitoring*

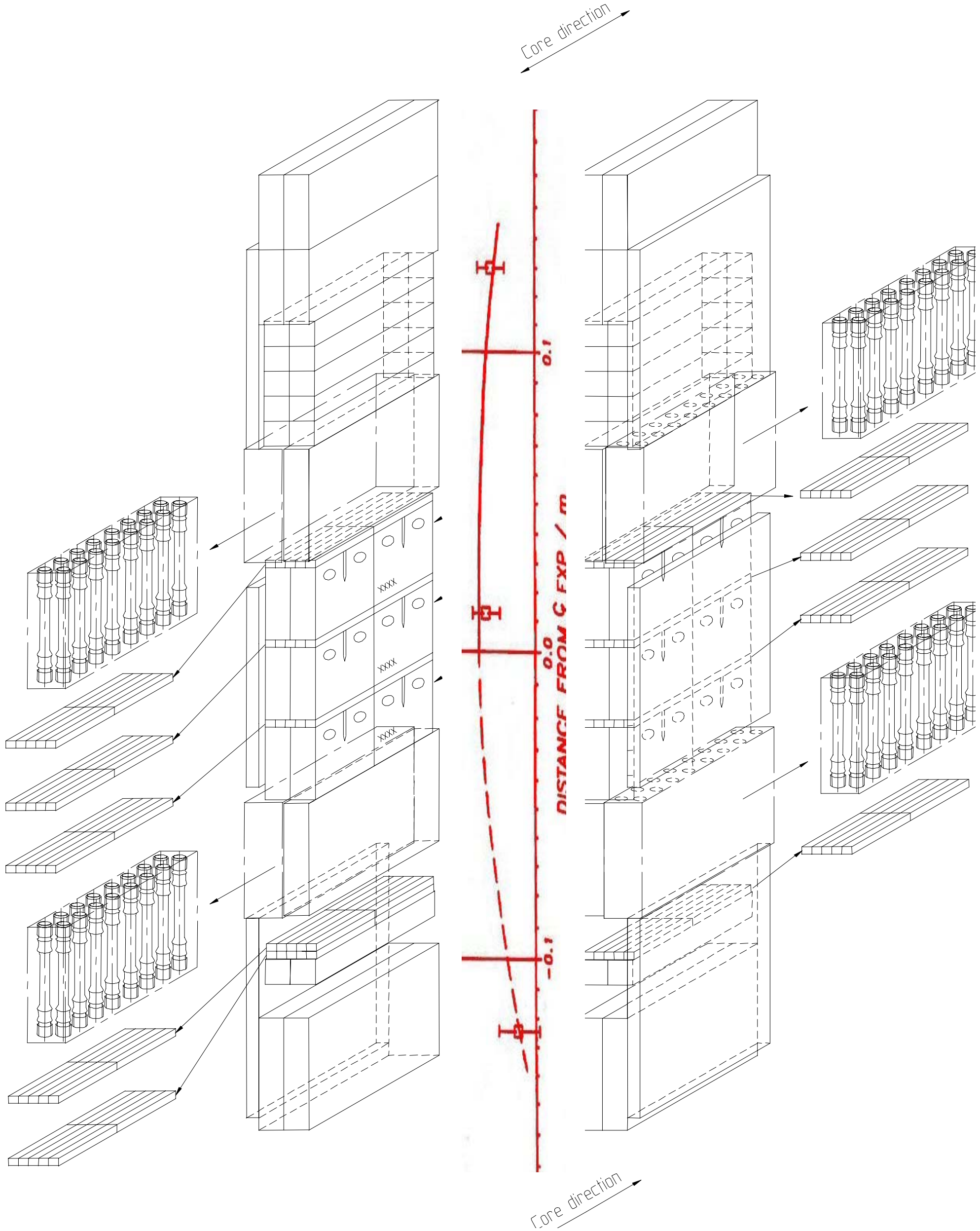


Figure 3 Temperature dependence of thermal conductivity for C-C Composites

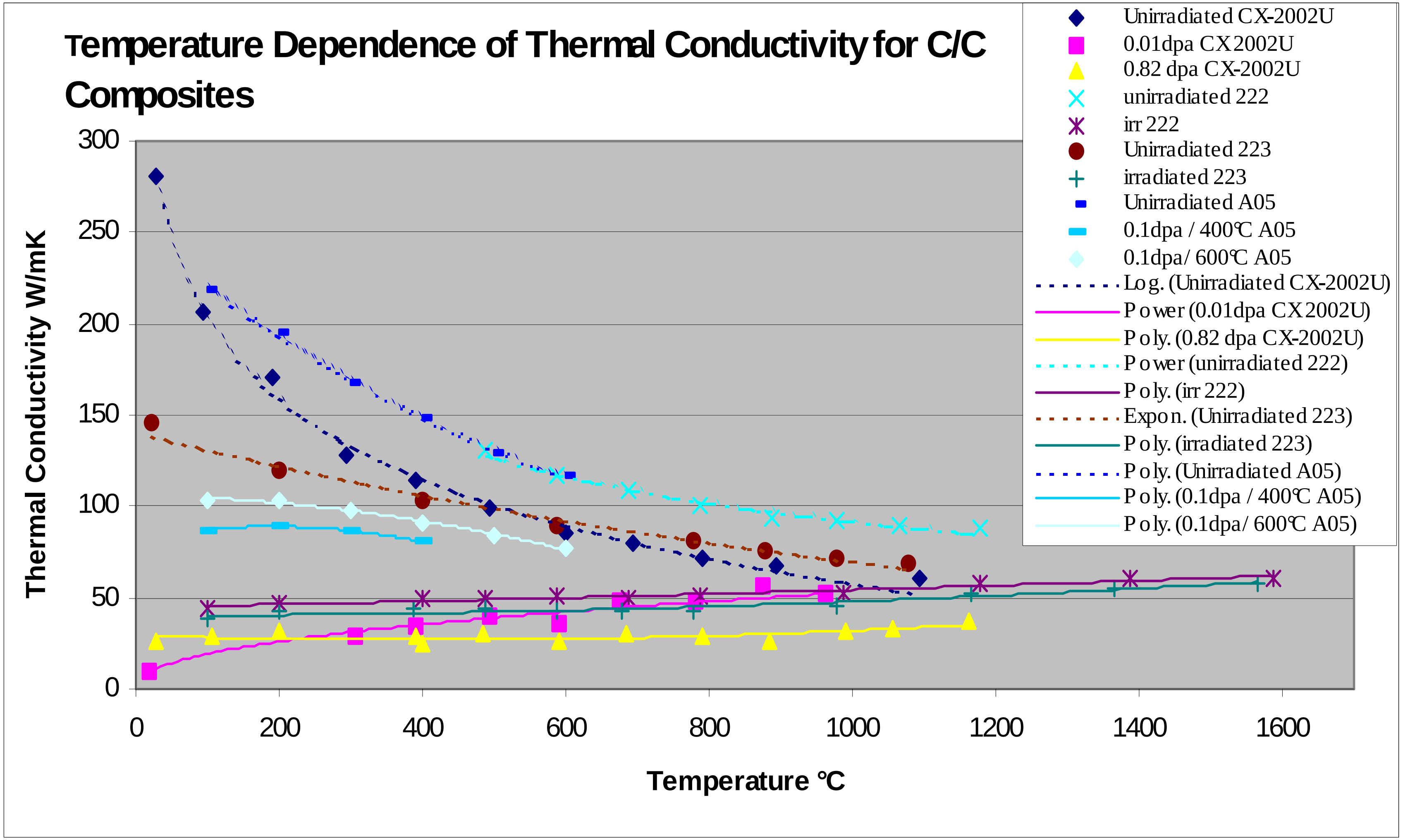


Figure 4 Creep rupture behaviour of Turbine Disc material Udimet 720 in air

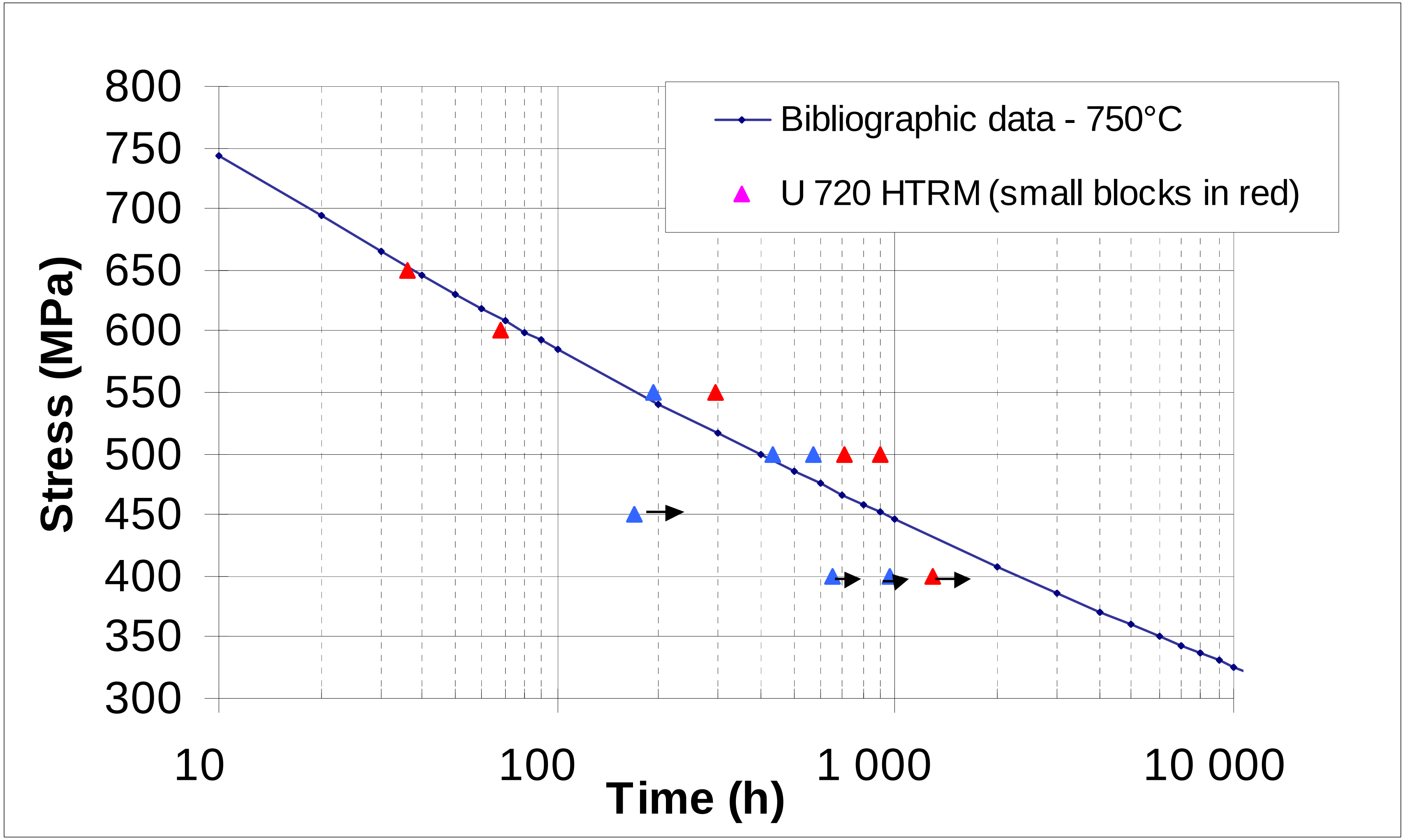


Figure 5 Tensile properties of Turbine Disc material Udimet 720 (pancake) in air

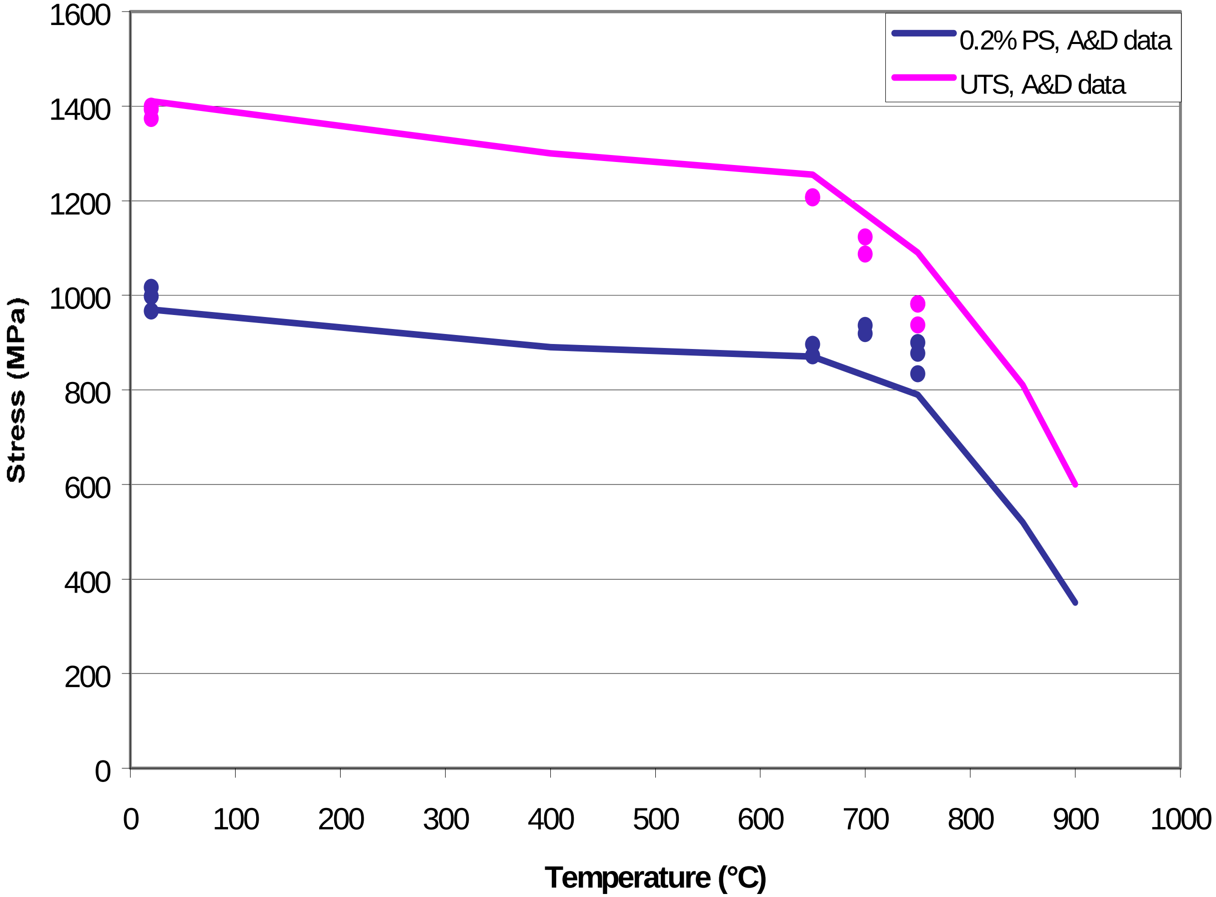
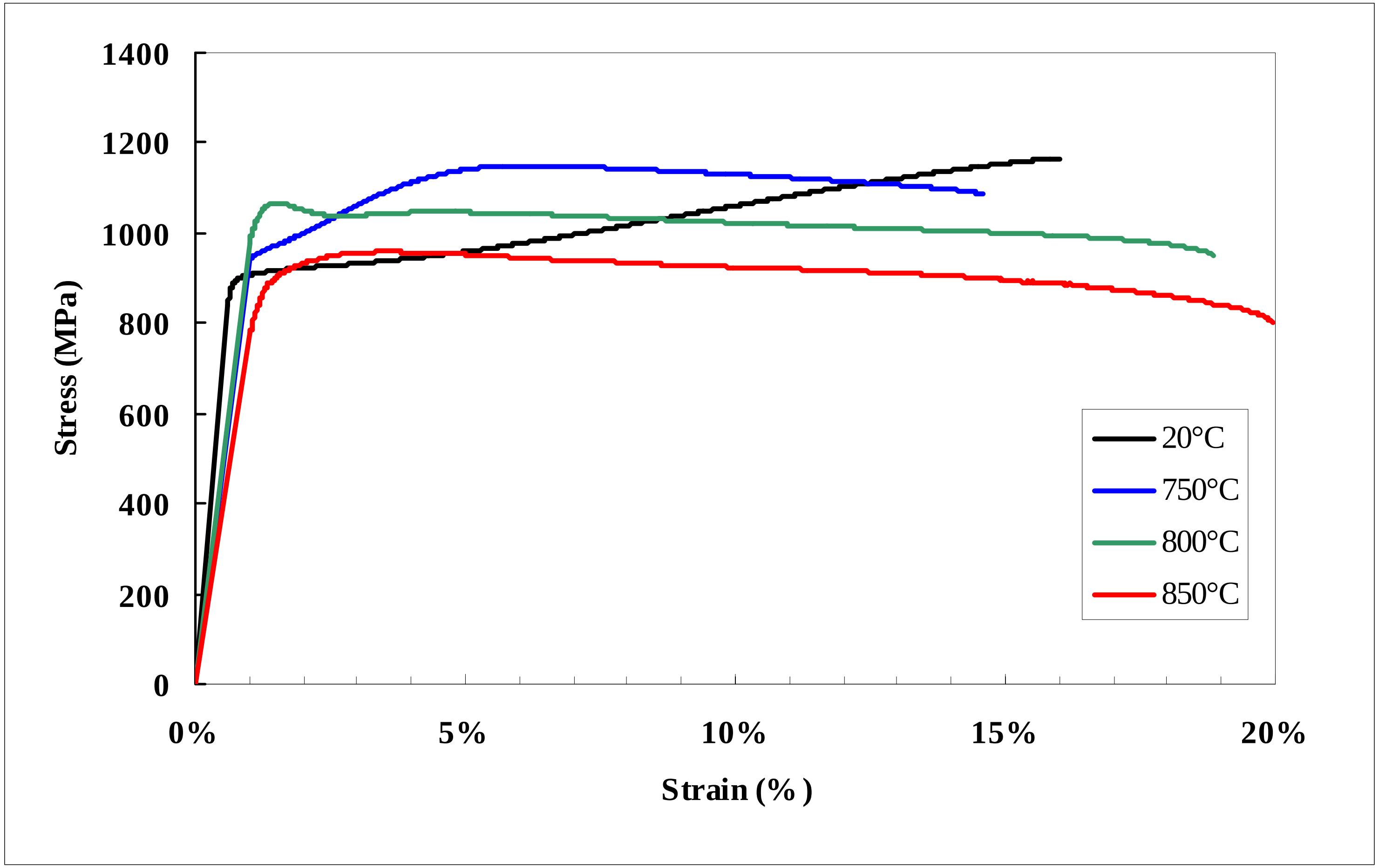


Figure 6 Turbine Blade material CM 247 LC DS Tensile properties



→ Tensile tests at 5.10^{-4} s^{-1}

Figure 7 Effect of irradiation on Dimensional Change for ATR-2E German graphite

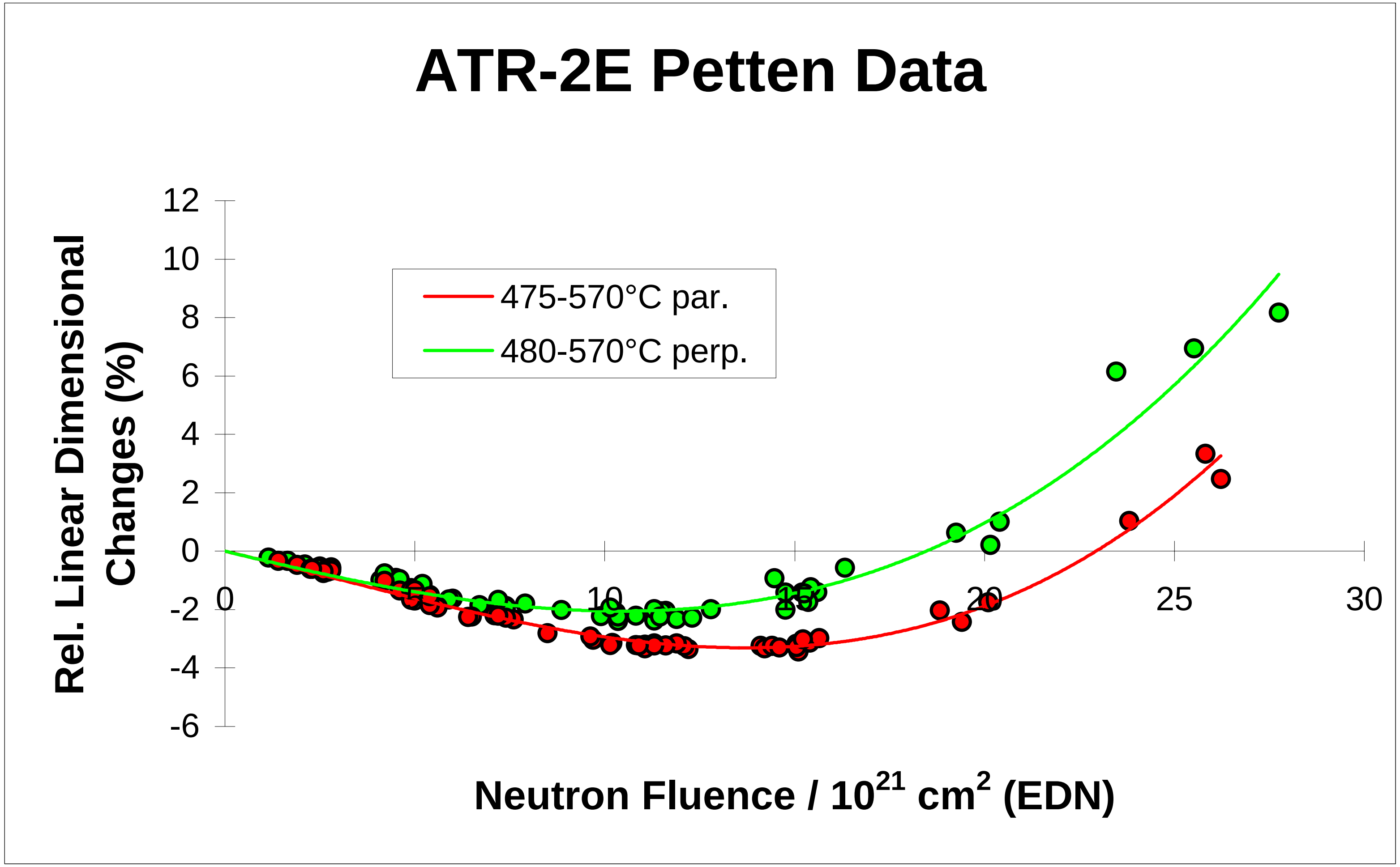
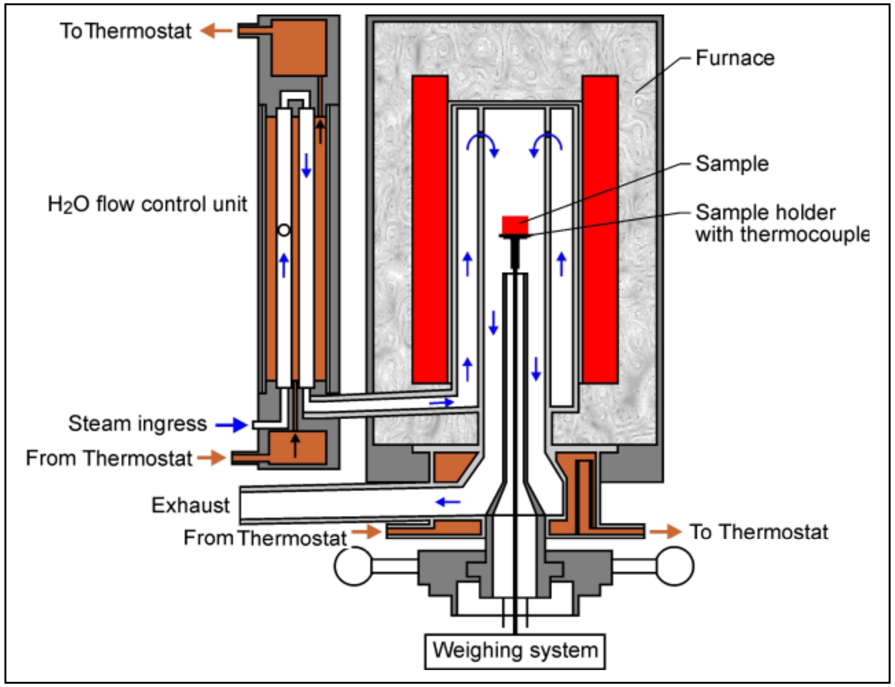
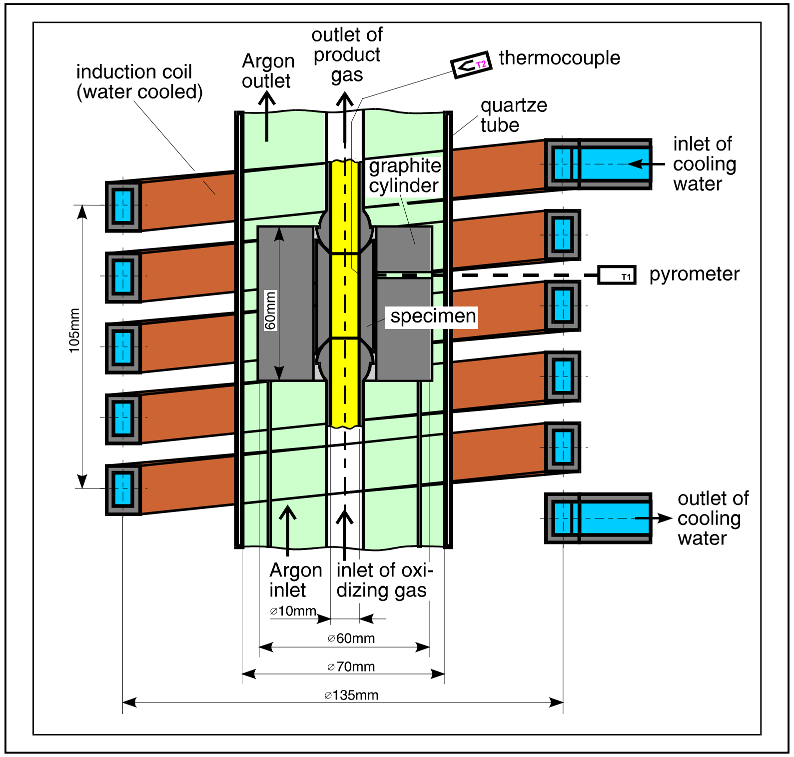


Figure 8 Schematics of THERA & INDEX Test Facilities used in the graphite oxidation work

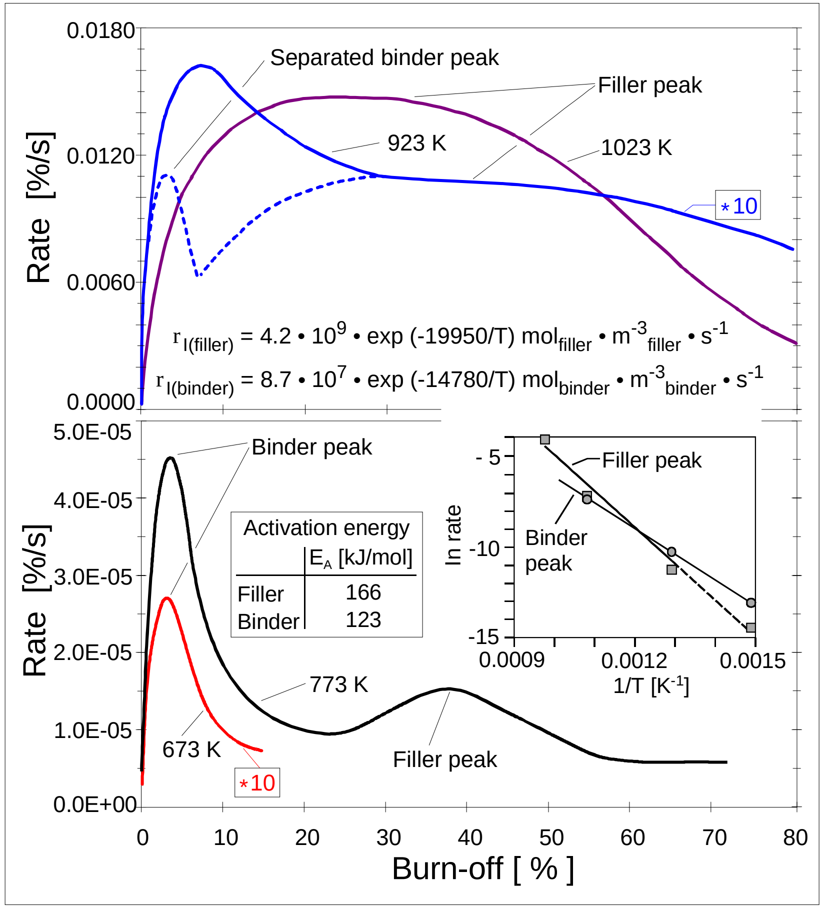


Schematic of THERA test facility

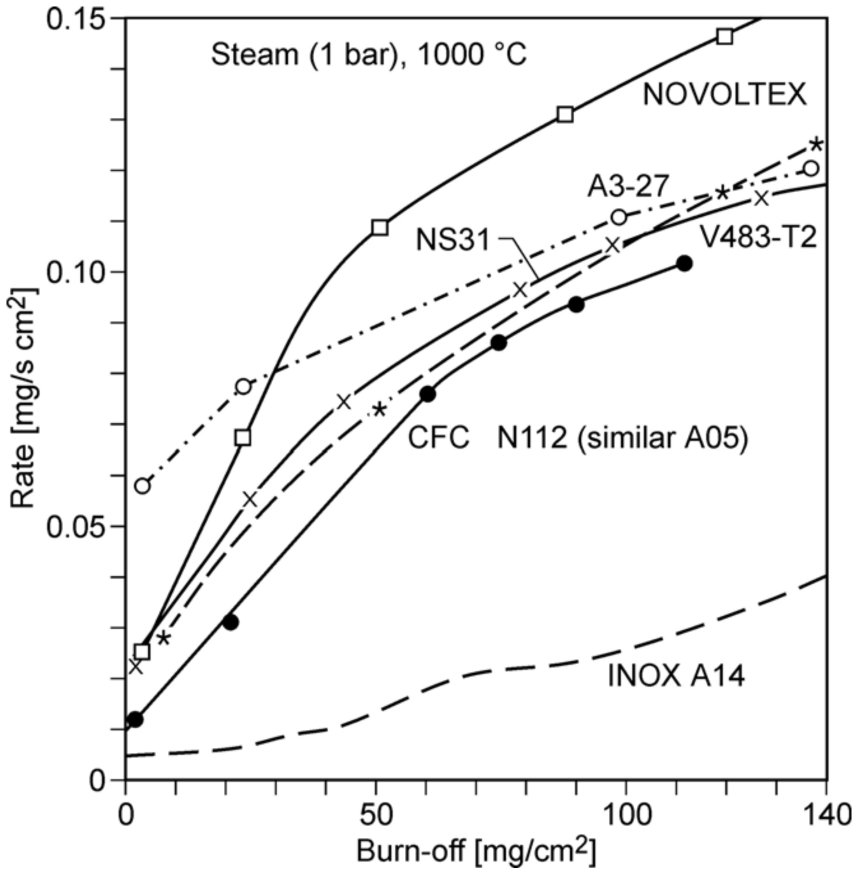


Schematic of INDEX test facility

Figure 9 Oxidation test results in air & steam



Oxidation in oxygen/air for Matrix Graphite A3-27

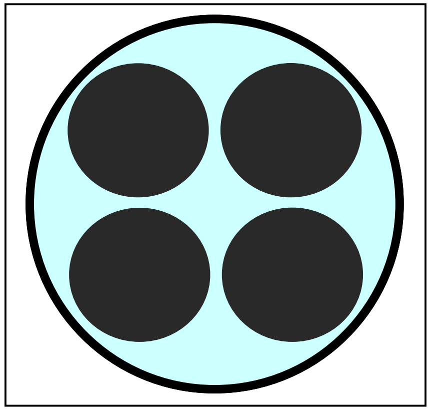


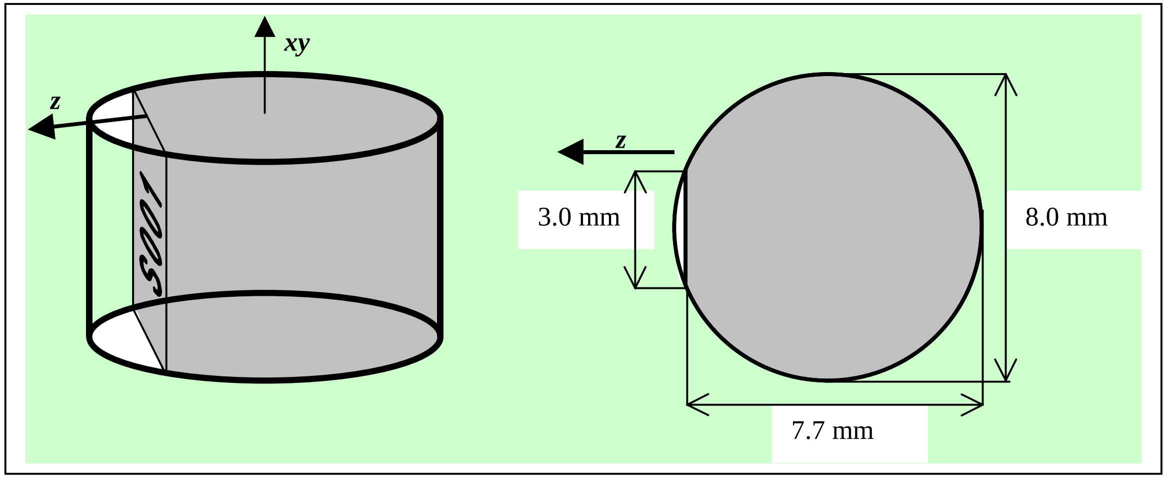
Oxidation in steam for C-based materials

Figure 10 Details of graphite specimens

SGL	PITCH COKE	PET COKE
EXTRUDED	**NBG-10**	*NBG-20*
ISO-MOULDED		**NBG-25**
GRAFTECH	PITCH COKE	PET COKE
EXTRUDED	*PPEA*	**PCEA**
ISO-MOULDED		**PCIB-SFG**
TOYO TANSO	PITCH COKE	PET COKE
EXTRUDED		
ISO-MOULDED	*IG-430*	**IG-110**

Layout in the Irradiation Rig Drum





Specimen Design

Cylinders Dia 8 mm and H = 6 or 12 mm

Flattened side for specimen ID

Figure 11 Details of irradiation in HFR

HFR

- 45 MW Pool type reactor
- Core lattice is 9 x 9 array containing 33 fuel assemblies, 6 control rods, 19 experimental positions and 23 Be reflector elements • Active height 600 mm
- More than 11 cycles (25 FPD) per year: typical total 280 Full Power Days

<u>In-core positions</u> C7/E7 or equivalent

–Fluence rate $\O_{tot}$ = 6-9 E18 $m^{-2}s^{-1}$–EDN fluence ~ 6-7 E25 m-2, or ~ 8 dpa

–Required 11 to 15 cycles take 1-1.5 years

–Ratio min/max ~ 65%

<u>Irradiation temperature</u> 750°C

<u>Helium environment</u>

–6.0 quality helium, slow purge

–moisture monitoring

–CO addition possible

<u>Rig design</u> based on D85 series (German HTR-Modul progamme up to 1994), and actual SiCSiC irradiation (2002)

Overview HFR loading geometry

		A	B	C	D	E	F	G	H		I
	1	Be	Be	Be	Be	Be	Be	Be	Be	1	Be
	2	F	F	F	D2	F	F2	F	H2	2	Be
	3	F	F	C3	F	E3	F	G3	F	3	Be
PSF	4	F	CR	F	CR	F	CR	F	H4	4	Be
	5	F	F	C5	F	E5	F	G5	F	5	Be
WEST	6	F	CR	F	CR	F	CR	F	H6	6	Be
	7	F	F	C7	F	E7	F	G7	F	7	Be
	8	F	F	F	D8	F	F8	F	H8	8	Be
	9	Be	Be	Be	Be	Be	Be	Be	Be	9	Be

F	**Fuel element**
CR	**Control rod**
Be	**Reflectorelement**
E5	**Experiment**