ABSTRACTS

The Second Information Exchange Meeting on Basic Studies in the Field of High Temperature Engineering

Headquarters of Organisation for Economic Co-operation and Development
Paris, France
10-12, October, 2001

Organised by OECD Nuclear Energy Agency
Co-organised by Japan Atomic Energy Research Institute
Meeting Organisation

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Program of Presentation

10th October

9:00-9:15 Welcome Address by NEA
C. Kessler (OECD/NEA)

9:15-9:30 Opening Address by Chairman
S. Ishino (Tokai University)

Session I "Overviews of High Temperature Engineering Research in Each Country and Organisation"

9:30-10:00 Research Activities on High Temperature Gas-cooled Reactors (HTRs) in the 5th EURATOM RTD Framework Programme
J. Martin-Bermejo, M. Hugon and G. Van Goethem

10:00-10:30 Update on IAEA High Temperature Gas Cooled Reactor Activities
J. M. Kendall and M. A. Methnani

10:30-11:00 Coffee Break

11:00-11:30 Present Status of the Innovative Basic Research on High Temperature Engineering using the HTTR

11:30-12:00 Present Status in the Netherlands of Research Relevant to High Temperature Gas-Cooled Reactor Design
J.C. Kuijper, A.I. van Heek, J.B.M. de Haas, R. Conrad, M. Burghartz and J.L. Kloosterman

12:00-12:30 Current Status of High Temperature Engineering Research in France
D. Hittner, F. Carre and J. Rouault

12:30-14:00 Lunch

14:00-14:30 Overview of High Temperature Reactor Engineering and Research
K. Kugeler and P.-W. Phlippen, M. Kugeler, and H. Hohn

14:30-15:00 The Contribution of UK Organisations in the Development of New High Temperature Reactors
A.J. Wickham, I.M. Coe, P.J. Bramah and T.J. Abram

15:00-15:30 Coffee Break

Session II "Improvement in Material Properties by High-temperature Irradiation"
15:30-16:00 Improvement in Mechanical Properties of Materials by High Temperature Irradiation
H. Kurishita

16:00-16:30 Effects of Superplastic Deformation on Thermal and Mechanical Properties of 3Y-TZP Ceramics
M. Ishihara, T. Shibata, C. Wan, S. Baba, Y. Motohashi and T. Hoshiya

16:30-17:00 Development of an Innovative Carbon-based Ceramic Material - Application in High Temperature, Neutron and Hydrogen Environment -
C. H. Wu

17:00-17:30 A Carbon Dioxide Partial Condensation Cycle for High Temperature Reactors
T. Nitawaki and Y. Kato

11th October

Session III "Development of In-core Material Characterisation Methods and Irradiation Facility"

9:00-9:30 Development of the I-I type Irradiation Equipment for the HTTR
T. Shibata, T. Kikuchi, S. Miyamoto, K. Ogura and Y. Ishigaki

9:30-10:00 Development of Optical Diagnostic System for High Temperature Gas Cooled Reactor
T. Shikama

10:00-10:30 Measurement Methods of In-Core Neutron and Gamma-ray Distributions with Scintillator Optical Fiber Detector and Self-Powered Detector

10:30-11:00 Coffee Break

11:00-11:30 Development of In-Core Test Capabilities for Material Characterisation
C. Vitanza

11:30-12:00 Application of Work Function Measuring Technique to Monitoring/Characterization of Material Surfaces under Irradiation
G-N. Luo, K. Yamaguchi, T. Terai and M. Yamawaki

12:00-12:30 Materials Irradiation Tests at High Temperatures in JMTR
M. Narui, T. Shikama and H. Matsui

12:30-14:00 Lunch
Session IV
"Basic Studies on Behaviour of Irradiated Graphite/Carbon and Ceramic Materials including their Composites under both Operation and Storage Conditions"

14:00-14:30 Investigation of High Temperature Reactor (HTR) Materials
D. Buckthorpe, R. Couturier, B. van der Schaaf, B. Riou, H. Rantala, R. Moormann, F. Alonso and B-C Friedrich

14:30-15:00 Radially Keyed Graphite Moderator Cores: An Investigation into the Stability of Finite Element Models
W.R. Taylor, M.D. Warner, G.B. Neighbour, B. McEnaney and S.E. Clift

15:00-15:30 Understanding of Mechanical Properties of Graphite on the Basis of Mesoscopic Microstructure
M. Ishihara, T. Shibata, T. Takahashi, S. Baba and T. Hoshiya

15:30-16:00 Coffee Break

16:00-16:30 Advanced Graphite Oxidation Models
R. Moormann and H.-K. Hinssen

16:30-17:00 Irradiation Creep in Graphite - A Review
B. J. Marsden and S. D. Preston

17:00-17:30 An Analysis of Irradiation Creep in Nuclear Graphites
G. B. Neighbour and P. J. Hacker

17:30-18:00 Planning for Disposal of Irradiated Graphite: Issues for the New Generation of HTRs
B. McEnaney, A.J. Wickham and M. Dubourg

18:00-18:30 Status of IAEA International Database on Irradiated Nuclear Graphite Properties with respect to HTR Engineering Issues
P. J. Hacker, B. McEnaney, A.J. Wickham, and G. Haag

12th October
Session V
"Basic Studies on HTGR Fuel Fabrication and Performance"

9:00-9:30 Dramatic Fissiion Product Release from Irradiated Nuclear Ceramics
L. Thomè, A. Gentils, J. Jagielski and F. Garrido

9:30-10:00 Technique of the Reactor Tests of HTGR Fuel Elements at Pulse Loading
A. S. Chernikov, V.S. Eremeev, V.Ya. Ivanov and A. A. Kuznetsov

10:00-10:30 Gas Reactor TRISO-Coated Particle Fuel Modeling Activities at the Idaho National Engineering and Environmental Laboratory
D. Petti, J. Maki, G. Miller, D. Varacalle and J. Buongiorno
10:30-11:00 Coffee Break

Session VII "Summary Session"

11:00-13:00
Session I

Overviews of High Temperature Engineering Research in Each Country and Organisation

Chaired by
K. Kugeler(FZJ, Germany) and M. Lecomte(Framatom-ANP, France)
Research activities on High Temperature Gas-cooled Reactors (HTRs) in the 5th EURATOM RTD Framework Programme

Joaquín Martin-Bermejo, Michel Hugon and Georges Van Goethem
European Commission, Belgium

The strategic goal of the 5th EURATOM RTD Framework Programme (FP5) is to help exploit the full potential of nuclear energy in a sustainable manner, by making current technologies even safer and more economical and by exploring promising new concepts. Part of this programme is being implemented through “indirect actions”, i.e. research co-sponsored (up to 50% of total costs) and co-ordinated by DG RESEARCH of the European Commission (EC) but carried out by external public and private organisations as multi-partner projects. The total budget available for these indirect actions during FP5 (1998-2002) is Euro 191 million.

Among the priority areas identified in FP5, one contains innovative elements with respect to previous programmes: “safety and efficiency of future systems”. One of the main objectives of this area is to investigate and evaluate new or revisited concepts (both reactors and alternative fuels) for nuclear energy that offer potential longer term benefits in terms of cost, safety, waste management, use of fissile material, less risk of diversion and sustainability.

After two calls for proposals of FP5 (deadlines October 1999 and January 2001), a number of contracts for research projects covering the objectives of the above mentioned area has been successfully negotiated by the EC and different EU organisations. Most of these projects are related to High Temperature Gas-cooled Reactors (HTR). In general, they include both experimental and analytical activities, and address key HTR issues such as fuel technology, fuel cycle, materials, power conversion systems and licensing. Approximately twenty-five different organisations from seven EU member states representing research centres, universities, and, particularly, the industry sector, participate in the projects. The total cost of these projects is about Euro 17 million and the EC financial contribution is roughly Euro 8.5 million (i.e. 50% of total cost).

This paper provides a brief summary of the most relevant aspects of EURATOM FP5, the objectives and work programmes of the HTR-related projects co-sponsored by the EC, and the EC expectations concerning the future RTD prospects in this area.
Update on IAEA High Temperature Gas Cooled Reactor Activities

J. M. Kendall and M. A. Methnani
International Atomic Energy Agency

IAEA activities on high temperature gas cooled reactors are conducted with the review and support of Member States, primarily through the International Working Group on Gas Cooled Reactors (IWGGCR). This paper summarises the results of the IAEA gas cooled reactor project activities in recent years along with ongoing current activities through a review of Co-ordinated Research Projects (CRPs), meetings and other international efforts. In particular, the status of the ongoing CRP on evaluation of high-temperature gas cooled reactor performance is presented, including a summary overview of the experimental work and analytical benchmarks conducted by the various participants. Also presented is an overview of recent Agency meetings and technical document publications.
Present status of the Innovative Basic Research on High Temperature Engineering Using the HTTR

T. Hoshiya¹, M. Ishihara¹, T. Shibata¹, S.Ishino², T.Terai³, H.Itoh¹, T.Oku⁴, Y.Motohashi⁵, S.Tagawa⁶, Y. Katsumura³, M.Yamawaki³, T.Shikama⁷, C.Mori⁸, S.Shiozawa¹, Y.Sudo¹

1: Japan Atomic Energy Research Institute, 2: Tokai University, 3: University of Tokyo, 4: University of the Air, 5: Ibaraki University, 6: Osaka University, 7: Tohoku University, 8: Aichi Institute of Technology

The High Temperature Engineering Test Reactor (HTTR), the first high temperature gas cooled reactor in Japan, achieved its first criticality on November 10th, 1998 at Oarai Research Establishment of the Japan Atomic Energy Research Institute (JAERI). The HTTR is now under the stage for the commissioning test at high power of 20MW and at its full power of 30MW with reactor outlet temperatures of 850°C and 950°C. In officially, the JAERI will obtain the operation permission in 2002.

The innovative basic research on high temperature engineering is one of the key subjects using the HTTR. Three kinds of fields of innovative basic researches, new materials development and advancement field, high-temperature radiation chemistry and fusion technology field and high-temperature in-core instrumentation field have been performed in JAERI since 1994.

In the field of new materials development and advancement, preliminary irradiation tests using research reactors as well as ion accelerators have been performed in order to determine the effective irradiation test conditions in the HTTR. The subjects in the field consist of (1) neutron-irradiation processing of high-temperature oxide superconductors, (2) neutron transmutation doping (NTD) of silicon carbide (SiC) high-temperature semiconductor, and (3) study on the radiation damage mechanism of structural ceramics, ceramics composites, and superplastic ceramics.

In the field of high-temperature radiation chemistry, elaborate works have proceeded to utilize radiation-induced chemical reactions at high temperatures, with production of SiC fibers from a macromolecule compound of polysilane, as well as with decomposition of heavy oils and plastics. In the fusion technology, preparatory studies have been performed to enable in-core measurement of physical property changes of solid tritium breeding materials under irradiation.

In the final field of high-temperature in-core instrumentation, efforts have been made to develop a heat and radiation resistant optical fiber system and other devices in order to monitor the temperature, dose rate of gamma rays and neutron fluence as well as in-core visualization.

In the meeting an overview of the present status and future program of the researches is presented.
Present Status in the Netherlands of Research Relevant to High Temperature Gas-Cooled Reactor Design

J.C. Kuijper, A.I. van Heek, and J.B.M. de Haas
NRG, The Netherlands

R. Conrad, and M. Burghartz
EC-JRC-IAM, The Netherlands

J.L. Kloosterman
IRI, Delft University of Technology, The Netherlands

Research, relevant to the design of high temperature gas-cooled reactors (HTR), is being performed in the Netherlands at NRG Petten/Arnhem, JRC-IAM Petten and IRI–Delft University of Technology, Delft. This paper will describe the ongoing HTR activities in these organisations, also indicating the international –European and world-wide– framework.

An important role in these activities is played by the JRC’s High Flux Reactor (HFR) in Petten. Because of its favourable design and operational characteristics and the availability of dedicated experimental equipment, the HFR has been used extensively as a test bed for HTR fuel and graphite irradiations for more than 30 years. An update will be presented on the HTR-related irradiation program since the First Information Exchange Meeting in 1999.

Besides computational support for the irradiation experiments mentioned above –e.g. design calculations and prediction of sample composition after irradiation– more general HTR-related computational analyses are being carried out by the organisations in the Netherlands, e.g. studies on the ACACIA combined heat and power concept, the South African PBMR, the HTR-10 and the HTTR and also on the implications of Pu-incineration in HTR systems and innovative burnable poison concepts. These analyses comprise, among others, computational HTR core physics, thermal hydraulics and shielding analysis, HTR system safety related transient analysis and also the development, improvement, verification and validation of software for performing these types of analyses. A number of these topics will be highlighted in the paper. Also additional experimental activities will be presented.
Current Status of High Temperature Engineering Research in France

Dominique HITTNER
Framatome –ANP, France

Franck CARRE and Jacques ROUAULT
CEA, Nuclear Energy Division, France

CEA and Framatome ANP researches in the field of high temperature engineering mainly support High Temperature Gas Cooled Reactors (HTGCRs or HTRs) technology development. These researches are for the short term directly connected to the support of the Framatome ANP for the industrial development of small or medium size thermal HTRs with direct cycle and the necessary recovery at national level of expertise on some associated crucial technological issues. For the longer term, CEA main R&D effort is paid to study the technologies in support of HTGCRs with hardened neutron spectra, refractory fuel possibly compliant with on-site reprocessing that appear to be a very promising candidate for future generations of nuclear systems, with regard to the goals of high efficiency, saving of resources, minimization of long-lived waste production and potential for other applications than electricity production (hydrogen, desalination,…). The corresponding programmes concerning high temperature engineering are reviewed in this paper.
Overview of High Temperature Reactor Engineering and Research

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Nuclear Energy has to be used in the future under the conditions of sustainability. Enhanced reactor safety, improved economics, suited waste management and best use of resources are very important topics in this context. The modular high temperature reactor can contribute to fulfil these requirements because it has excellent safety characteristics and can be applied not only to produce electricity very efficient, but also to deliver nuclear process heat for many applications in the energy economy. Broad activities in the field of HTR are underway actually in countries like Japan and China, which took test reactors in operation, or in South Africa, USA, Russia, which plan to build such plants. In countries with a long-term tradition in gas-cooled reactors like Great Britain, France and Germany the interest in this type of reactor is still kept.

The paper will describe and discuss activities in the different programs especially with regard to the future gas turbine application. Important aspects for the technical feasibility of main components of this process combination are considered too.

The safety analysis of these nuclear power plants is dedicated to a system which will avoid the problems which are connected to the today’s existing reactors. These reactors cannot melt even in severe accidents of loss of coolant and fuel cannot be destroyed by other severe accidents. Therefore, the release of radioactivity to the environment is very limited. The state of technology and new developments regarding this topic are explained in more detail in this paper. The total concept of inherent safety including questions of air and water ingress and extreme impacts from outside are discussed and analyzed.

Open questions and future needs for research and development in the field of HTR technology are added to get a impression of the state of HTR technology world wide.
The Contribution of UK Organisations in the Development of New High Temperature Reactors

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Consultant, UK  
(and Bath Nuclear Engineering Group  
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I. M. Coe  
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The UK has a history of the successful design, construction and operation of graphite-moderated gas-cooled reactors. In particular, a central role was played in the original development of HTR, focussed upon the construction of the Dragon prototype reactor at Winfrith. In this regard there was a particular expertise built up around the performance of the fuel particles, and also in the area of graphite oxidation from potential coolant impurities.

The revival of interest in HTR through the development of GT-MHR and PBMR modular designs, through continuing national programmes such as ACACIA in The Netherlands, HTTR in Japan and HTR-10 in China, and the general potential for co-generation and inherently safe operation, has renewed the interest of the principal UK centres of reactor expertise, which are particularly focussed upon the PBMR design. This review covers the present activity of principally NNC Ltd, BNFL and AEA Technology plc: in addition, the activities of some smaller UK companies in support of the programme are also described.
Session II

Improvement in Material Properties by High-temperature Irradiation

Chaired by
T. Shikama (Tohoku Univ., Japan) and B. Marsden (AEA Tech., UK)
Improvement in Mechanical Properties of Materials by High Temperature Irradiation

H. Kurishita
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Embrittlement caused by high energy particle irradiation is the critical concern for use as structural materials exposed to radiation environments. In particular, refractory metals such as molybdenum and tungsten are known to exhibit serious embrittlement even after low level irradiation. It is believed that radiation embrittlement arises from radiation hardening and thus inevitably occurs since irradiation always causes significant hardening. Radiation hardening is mainly due to irradiation-produced defects and therefore it has long been believed that the only way to relieve radiation embrittlement is to suppress radiation hardening by introducing a large number of sinks for irradiation-produced defects.

Recent progress of both the development of irradiation rigs mainly performed in the Japan Materials Testing Reactor (JMTR) and materials fabrication techniques has enabled to change the above understanding of radiation embrittlement. It was found that fast neutron irradiation caused a remarkable increase in low temperature ductility of refractory metal by approximately five times as measured with impact three-point bending tests, regardless of significant radiation hardening. This ductility increase due to irradiation is directly opposite to radiation embrittlement and is called radiation induced ductilization (RIDU). The only possible explanation for the occurrence of RIDU is that since any materials have microstructural inhomogeneity including grain boundaries and interfaces that may act as crack initiation and/or propagation sites, such weak places are strengthened (i.e., become more resistant to fracture) by irradiation and its beneficial effect is much larger than the detrimental effect of radiation hardening. The strengthening of weak places by irradiation was attributed to radiation-enhanced precipitation and segregation which occur preferentially at weak grain boundaries under high energy irradiation.

Effects of precipitation and segregation on mechanical properties of materials depend strongly on the species and contents of precipitates and segregated elements. It is therefore important to control the compositions and contents of constituents so that radiation-enhanced precipitation and segregation may have the beneficial effect of strengthening the weak places. For the control, we can make good use of the data obtained so far for unirradiated samples. Since radiation-enhanced precipitation and segregation may occur easily as the irradiation temperature is increased, it is expected that high temperature irradiation can be a good tool for improvement in mechanical properties of materials.
Effects of Superplastic Deformation on Thermal and Mechanical Properties of 3Y-TZP Ceramics

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1: Japan Atomic Energy Research Institute
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It is well known that when a ceramic sample is pulled in tension it usually breaks at fairly low strain. However, some ceramics are capable of pulling out to very large tensile deformations of the order of some hundred or even thousand of percent. It is said that this phenomena so-called superplasticity on ceramics appears when the size of grains decreases at the magnitude of sub-micron order. The discovery of the superplastic phenomena in ceramic materials expands the application of the ceramics, and many researches have been performed in the field of basic understandings of superplasticity of ceramics as well as application technology of the superplastic deformation, e.g. advanced forming and joining technologies with superplastic deformation etc.

Authors have also investigated the superplastic zirconia-ceramics with the aim of nuclear application at high temperature engineering field, and the neutron irradiation research on the superplastic ceramics at high temperature has been proposed as an innovative basic research using the High Temperature Engineering Test Reactor (HTTR). A preliminary study is started out using other research reactors rather than the HTTR as well as accelerators in order to decide effective irradiation test condition using the HTTR.

From application viewpoint, changes in thermal and mechanical properties after superplastic deformation are one of the key subjects. In the thermal property research, 3Y-TZP, 3 mol% Yttria stabilized Tetragonal Zirconia Polycrystalline specimens were prepared after superplastic deformation process. Nominal amount of deformation in this study was 70%, and different amounts of cavities were produced by changing strain rate in the deformation process. The specific heat was measured by a DSC method, differential scanning calorimetry, at a wide range of temperature from 473K to 1273K. The crystal structure was characterized by measuring X-ray diffraction patterns, and microstructural observation was carried out using a scanning electron microscope, SEM, after thermal etching.

On the other hand, in the mechanical property research, the 3Y-TZP specimens were also prepared after superplastic deformation process. The maximum amount of deformation was about 150% in this study. Changes in the hardness and Young’s modulus were measured by dynamic indentation technique. Pyramid-type indenter was used in the experiment. The microstructural observation was also carried out using the SEM after thermal etching.

In this paper, obtained thermal and mechanical properties after superplastic deformation are presented.
Development of an Innovative Carbon-based Ceramic Material
- Application in High Temperature, Neutron and Hydrogen Environment -

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The next step fusion devices (fusion reactor) will be long pulse or steady state operation, the wall loading will be $\geq 10^{-20}$ MW m$^{-2}$, the expected plasma density is $\geq 10^{19}$ cm$^{-2}$ s$^{-1}$. In addition intense neutron will be produced, the assessed neutron flux will be around $3.5-9.0 \times 10^{14}$ cm$^{-2}$s$^{-1}$ for the first wall, whilst the neutron flux for the divertor is around $1-3 \times 10^{14}$ cm$^{-2}$s$^{-1}$, which will lead to change of material properties. From economical and safety point of view, an adequate long lifetime of Plasma Facing Components and acceptable low tritium retention are required. However, all low-Z materials possess high erosion yields via D$^+$/T$^+$ sputtering and relatively high tritium retention. Consequently, frequent removal of tritium and replacements of components are indispensable.

To improve the properties of carbon materials, one strives for increasing thermal conductivity ($300$ W m$^{-1}$K$^{-1}$, at 20 °C, 145 W m$^{-1}$K$^{-1}$, at 800 °C), reduced tritium inventory, reduced chemical erosion by interactions with deuterium/tritium, increased the resistance to water/oxygen at elevated temperatures and possible to increase the neutron stability.

In the framework of the European Fusion Research Program, a great effort has been made to develop an innovative carbon-based ceramic material to meet all of these requirements. After a decade of research and development, It is succeeded to develop an advanced material: namely, a 3D CFC, contains about 8-10 at% at. of Silicon with porosity is about 3-5%.

This advanced ceramic material possess very high thermal conductivity, dimensional stability under the neutron irradiation, lower chemical erosion (longer life time), lower tritium retention and lower reactivity with water and oxygen (safety concern).

This innovative ceramic materials seems very promising for application in the high temperature, neutron and hydrogen environment. A detailed discussion on development, properties and application of material is presented.
A Carbon Dioxide Partial Condensation Cycle for High Temperature Reactors

Takeshi NITAWAKI and Yasuyoshi KATO
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Helium is currently used as coolant in high temperature gas cooled reactors. However, carbon dioxide is used for AGRs, since there are some advantages in use of carbon dioxide as coolant: higher heat transport capacity and heat transfer coefficient (Nusselt number) between fuel cladding surface and coolant; and lower pumping power, depressurization rate in case of accidents and price, reference to helium. Inheriting these advantages and using condensability of carbon dioxide, we have been proposing a new concept of a carbon dioxide cooled partial condensation direct cycle for high temperature reactors with a passive decay heat removal system.

Fig.1 shows a schematic diagram of the partial condensation cycle system. Fig.2 shows variation of the cycle efficiency with the reactor outlet temperature and pressure. The cycle efficiency is higher in the system with partial condensation of about 40% than those of a simple Brayton cycle with no condensation and a Rankine cycle with full or 100% condensation.

Recently, an ultra high-purity Cr-Fe alloy has been developed in Tohoku University, whose tensile strength is four times higher at 1200 degree C than that of a Ni-based alloy. If this ultra high-purity Cr-Fe alloy is used as core material, the cycle efficiency exceeds 50% when the reactor outlet pressure is 12.5MPa and the reactor outlet temperature is 900 degree C (cladding hot spot temperature = 1200 degree C). The efficiency is 1.5 and 1.25 times higher than those of LWRs and LMFRs, respectively.

A liquid carbon dioxide storage tank is allocated between the circulation pump and the recuperator II as shown in Fig. 1. In case of coolant depressurization accidents, the core decay heat is passively removed by carbon dioxide gasified and supplied from the tank.
Session III

Development of In-core Material Characterisation Methods and Irradiation Facility

Chaired by
C. Vitanza(OECD) and M. Yamawaki(Univ. of Tokyo, Japan)
Development of the I-I type Irradiation Equipment for the HTTR

Taiju Shibata\textsuperscript{a}, Takayuki Kikuchi\textsuperscript{a}, Satoshi Miyamoto\textsuperscript{b}, Kazutomo Ogura\textsuperscript{b} and Yoshinobu Ishigaki\textsuperscript{c}

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\textsuperscript{b} The Japan Atomic Power Company, Japan
\textsuperscript{c} Fuji Electric Co., Ltd., Japan

The High Temperature Engineering Test Reactor (HTTR) is a graphite moderated, helium gas-cooled test reactor with a maximum power of 30MW. It is the first high temperature gas cooled reactor (HTGR) in Japan, and has a unique and superior capability served for irradiation tests by using its large and high temperature irradiation space.

The I-I type irradiation equipment, which is the first irradiation one for the HTTR, is now under construction after several experiments such as in-core monitoring method. It is served for an in-pile creep test for a stainless steel developed as a structural material of the fast reactor. The creep rate and rupture time of the in-pile creep specimens will be measured at elevated temperatures in the reactor. The equipment can give a big load to tensile specimens taking advantage of the large and high temperature irradiation space of the HTTR. Therefore, the specimens with a nominal gauge length of 30mm and diameter of 6mm can be adapted for the in-pile test. This specimen size is not a small one generally served for in-pile tests but the standard one served for usual non-irradiation creep tests. Specimens for out-pile creep tests after the irradiation are also installed in the equipment.

This equipment was designed so as to carry out in-pile creep tests in the HTGR. A part of the equipment is installed in the reactor pressure vessel through a standpipe of the HTTR. The installed part has a length of about 8900mm. The upper end of the part is fixed to the standpipe. The lower end of it is constituted of three tubular parts, which are two irradiation units and a guide tube, and is installed in the reactor core. The units are held in elevated ambient temperatures. In addition to the ambient temperatures and neutron and gamma heatings, electric heaters surrounding the specimens control the irradiation temperatures of the units to 550 and 600 °C, respectively. The weight loading system located outside the reactor gives a stable and precise tensile load on the specimens by using levers and weights. The maximum load of this system is about 10kN. The creep behavior of the specimen is detected by a differential transformer developed for high temperature condition.

The fabrication of this equipment was started in 1999. Functional tests for each component, such as the weight loading system, heater and differential transformer, were carried out outside the reactor. Then, the equipment was fabricated from components. Its performance is also demonstrated outside the reactor after the fabrication. The construction is in the final stage. This paper describes the progressing of the development of the I-I type equipment.
Development of Optical Diagnostic System for High Temperature Gas Cooled Reactor

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Optical diagnostic systems have several and definite advantages over other diagnostic systems such as conventional electronic systems for application in high temperature gas cooled reactors (HTGR). One of engineering datafacts, which clearly supports this statement, will be that there is no good electrical insulator which could function properly above 900K. Optical diagnostic systems utilizing silica core optical fibers have been developed in Japan for applications in high temperature test reactor (HTTR) developed by Japan Atomic Energy Research Institute.

First, radiation resistant optical fibers were developed which will endure a neutron fluence up to \(1 \times 10^{24} \text{n/m}^2\), which will correspond to a fluence expected in HTTR for more than one year operation in a central core region, though it should be pointed out that optical loss induced by radiation effects will depend strongly on radiation temperatures and fundamental understandings in mechanisms of introduction of optical transmission loss by neutron associated heavy irradiation. Using these radiation-resistant optical fibers, feasibility of optical diagnostics were demonstrated in a HTTR-relevant conditions in a Japan Materials Testing Reactor (JMTR). Examples are temperature measurements, and gamma-ray dose rate measurements, which will be proportional to a reactor nuclear power.

It is expected that utilization of radiation-induced luminescence (radioluminescence) will broaden applicability of optical diagnostic systems in HTGRs, however, scintillators are scarce, which could give intense radioluminescence at elevated temperatures. The most popular ceramic scintillator, a ruby, would not give intense optical signals above 600K. In case of silica itself, a few radioluminescence peaks were found such as peaks at 450nm and 1270nm. Origins of these peaks are not clearly identified yet but their intensity had very weak temperature dependence. Studies in search for new ceramic-scintillators usable at working temperatures of HTGRs are under way.

Also, a project, trying to demonstrate applicability of optical fiber systems in HTTR is under way. Technical difficulties associated with installation of optical fibers in HTTR have been already overcome by the HTTR technological group. And, above-mentioned developed radiation-resistant optical fibers will be inserted in a HTTR core region with scintillators.
Measurement Methods of In-Core Neutron and Gamma-Ray Distributions with Scintillator Optical Fiber Detector and Self-Powered Detector

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4) University of Tokyo, Japan
5) Nagoya University, Japan
6) Institute for Nuclear Safety Systems, Japan
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Two methods were developed for the measurement of thermal neutron flux and gamma-ray intensity distributions. One is the method composed of a scintillator-optical fiber with a scanning driver and the other is self-powered detector with the driver.

On the optical fiber method, we successfully developed, about more than 10 years ago, for the thermal neutron flux less than $10^8$ n/cm²s in a critical assembly, in which ZnS(Ag) scintillation powder on the tip of a fiber was attached and the other end of the fiber was connected to a photomultiplier. The number of pulses from the photomultiplier was counted. When the scintillator is mixed with $^6$LiF powder, neutron flux was measured, and without $^6$LiF, gamma-ray intensity was measured. The position distribution was obtained by scanning the fiber through the driver.

However, in a reactor core with neutron flux around $10^{12-13}$ n/cm²s, it was not able to use such detectors, because uncountable huge number of pulses were appeared and also ZnS(Ag) scintillator deteriorated due to radiation damage. We therefore used aluminum oxide scintillator “Desmarquest” which is highly resistive to radiation damage, and measured the electric current intensity from the photomultiplier. To distinguish the scintillation light component with longer wave length of 690 nm from the Cherenkov light component with shorter wave length, a band pass interference optical filter was used. With this new method, neutron flux distribution and gamma-ray intensity distribution in a reactor core were obtained over 2 meters in about 10 minutes with position resolution of about 1mm. This method is very useful for the characterization of ordinary research reactors. However, it is applicable up to only about 100 °C, because the light emission property deteriorates at high temperature.

On the self-powered detector, we are now trying to find a better insulator between the central needle electrode such as platinum and the outer cylindrical cathode of inconel, which can be used at high temperature. Pure aluminum oxide insulator was able to be used up to about 500 °C. Quartz insulator was able to be used up to about 650 °C. Other materials are under investigation.
Development of In-Core Test Capabilities for Material Characterisation

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This paper presents a concise review of some of the test capabilities developed at the OECD Halden Project for the characterisation of fuel and materials properties under irradiation. Insights on some of the instrumentation utilised in these tests are also presented. The paper addresses some of the irradiation techniques used for gas reactor investigations, notably in the field of AGR fuel and materials.

(1) Currently at the OECD NEA
Application of Work Function Measuring Technique to Monitoring/Characterization of Material Surfaces under Irradiation

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Experimental devices have been developed for examining the work function (WF) change of metallic and ceramic materials due to ion irradiation in low energy (500eV) or high energy (MeV) ranges. The charging effect on the performance of Kelvin probe (KP) was reduced efficiently using appropriate shielding. An error deduction method has been suggested to effectively eliminate the influence due to charging by introducing a reference sample. Polycrystalline Ni and W samples with 99.95% purity each were used in the research. The experiments were performed using He⁺ and H⁺ ion beams of 1MeV, 2x10¹⁶ ions/m²/s under a pressure of 1x10⁻⁴ Pa, and a He⁺ beam of 500eV, 2x10¹⁶ ions/m²/s under a pressure of about 1x10⁻²Pa. The results indicated that the irradiation of 500eV He⁺ resulted in a WF decrease, then an increase till saturation, while 1MeV ions only induced a WF decrease, then saturation. A surface model of loosely bound adsorbed layer plus native oxide layer on metals is presented to explain the observed phenomena. The nuclear stopping is responsible for the results in the case of 500eV He⁺ irradiation that is powerful enough to sputter away the whole overlayer from the bulk surface. In MeV case, the electronic stopping plays a decisive role, which allows merely the topmost adsorbed layer to be removed by He⁺ and H⁺ ions of 1MeV. The application of this technique to monitoring/characterization of material surfaces in a nuclear reactor is to be discussed.
Materials irradiation tests at high temperatures in JMTR

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There are increasing demands for high temperature irradiations in a fission reactor from researchers of universities in Japan. There, a precise temperature control is important especially for fundamental studies of irradiation effects in materials for high temperature applications.

Below 500°C, a technique of temperature control by electrical heating systems has been established and irradiation temperatures are controlled, being independent of a reactor nuclear power in Japan Materials Testing Reactor (JMTR). An irradiation rig composed of several subcapsules, which could be retrieved from a reactor core during irradiation, makes it possible to study irradiation effects in materials as functions of a temperature and a neutron fluence.

Several irradiations have been carried out at temperatures above 800°C for study of irradiation effects in materials for HTGR (high temperature gas cooled reactor)-applications, such as silicon-carbide (SiC) composites. Also, preliminary studies have been carried out prior to their planned executions at High Temperature Test Reactor (HTTR). There, temperature control is carried out by a so-called helium-gas pressure control method (GP-method). Thermal conductance between irradiated materials and a capsule wall could be controlled thermal conductance of helium gas filling a capsule.

It is demonstrated that the GP-method could control irradiation temperatures independent of the JMTR reactor power above 10MW, being 20% of its full power. Thus, effects of low-temperature irradiation effects could be minimized by the GP-method. The GP-method has a large time constant to regulate temperature and it needs a sophisticated program for a computer-assisted automatic system. Recently, the reactor operating group in the JMTR in JAERI-Oarai Establishment developed a computer program for the GP-method successfully, which made the GP-method applicable at wider temperature ranges.

The paper will describe a recent status of irradiation tests in JMTR for Japanese university research activities in HTTR related fields. Also, the paper will describe technical details of temperature control in a fission reactor at elevated temperatures.
Session IV

Basic Studies on Behaviour of Irradiated Graphite/Carbon and Ceramic Materials including their Composites under both Operation and Storage Conditions

Chaired by
B. McEnaney(Uinv. of Bath, UK) and A. Wickham(Consultant, UK)
Investigation of High Temperature Reactor (HTR) materials

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The high Temperature (HTR) and gas-cooled reactors (GCR) have been developed within Europe over many decades and considerable expertise exists within the European Community countries in this technology. A common European approach to the renewal of HTR technology through the direction of a European HTR Technology Network (HTR-TN) has been established to enable and encourage work-shared structures within this nuclear R&D field.

European Community Framework Programmes have been launched to consolidate and advance the modular HTR technology in Europe. These involve partnerships of principal industrial and research organisations from countries of the European Union, with JRC, to develop and consolidate European expertise and experience. This paper gives a description and first results for the Fifth Framework project HTR-M on materials, which focuses on the materials requirements for reactor pressure vessel, high temperature resistant alloys for the internal structures and turbine, and graphite for the reactor core.

Design information on the HTR Reactor Pressure Vessel (RPV) under HTR relevant conditions is needed in the areas of design analysis, structural integrity analysis and materials properties data. Also information is needed on manufacturing of products and parts, design property
data, influence of environment (neutron-irradiation fluence, operational temperature, and helium environment) and welding. The latter in particular is seen as an important feasibility issue. The objective is to confirm the choice of candidate material for different HTR concepts, in relation to both normal and accident conditions, and compile their design properties. The work will involve testing of selected steels and welded features under irradiated and non-irradiated conditions.

The components of the primary circuit operate at temperature of the order of 850°C in order to reach high energy levels of efficiency. For such components a number of metallic and ceramic materials are required for the core and reactor internals materials (metallic, ceramic and composites) and for the core support and other structures. For these materials and components, some data exists but the information and experience are not well established for the HTR environment. Also highly loaded metallic materials are required for the turbine components for and a significant effort of material development is required in this area. The work involves compilation of a database and testing of important material options within simulated HTR environments for specific components of the reactor internals (control rod) and the turbine.

The need for reliable graphite data is a crucial issue for existing and new HTR projects and an important requirement for future decommissioning activities. Also oxidation resistance of graphites at high temperature requires special attention due to its relevance for safety analyses of air and water ingress accidents, licensing procedure and normal operation. Limited test work associated with graphite oxidation on the consequences of severe air ingress with core burning and advanced C-based options is planned. Also investigations will be conducted to establish a graphite data base (taking benefit from existing data bases) as a preliminary to more detailed work involving testing and development of new graphites.
Radially keyed graphite moderator cores: An investigation into the stability of finite element models

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The majority of the United Kingdom's gas-cooled thermal nuclear reactors have a graphite moderator core of the radially keyed design type. This core design is composed of large graphite bricks interspaced with interlocking moderator keys, configured in a manner which produces a negative Poisson's ratio overall under thermal and mechanical influences. It is critically important that the key safety functions of control rod insertion and fuel cooling can be maintained throughout the reactor life. Based on existing evidence, it is presently believed that there is a large degree of redundancy in this functionality but this has yet to be quantified.

The possibilities for quantifying graphite moderator core behaviour in service are obviously extremely limited. One rather more approachable way to proceed with an exploration of the mechanical behaviour of the core is to develop computer models, Finite-Element (FE) analysis providing an obvious choice of technique. In this study, FE models of a single key and keyway have been constructed and used to build larger multi-brick arrangements in order to explore the conditions generated. Our study has used the ABAQUS finite element package and has explored the use of two methods of modelling key-keyway contact; springs and frictional contact surfaces. While the work has been prompted by current UK requirements, the FE methodologies deployed are relevant to graphite-moderated, high temperature nuclear reactors.

Results demonstrated that the graphite bricks themselves deformed very little. Closing up the gaps between the layers of bricks accommodated the deformation applied to the model. However, FE analyses of key-keyway interactions that use springs to represent the interactions between the contacting surfaces displayed unstable behaviour associated with keys rattling inside the keyways. This led to rapidly changing contact conditions and failure of the model to produce a convergent solution. Grounded springs reduced this rattle, but this reduction was not sufficiently large enough to produce stability in multi-key models.

The removal of springs and the incorporation of inter-layer friction between the model and a rigid substrate produced a stable solution in the single key arrangement. However, in order to do this it was necessary to create a three-dimensional FE model whereas previously a two dimensional model was being used. The inclusion of inter-layer friction also stabilised a larger multi-brick model. However, the large number of contacting surfaces, and the necessity to use three-dimensional elements reduced the efficiency of the model and produced impractical analysis times.

Recommendation for further work: As the bricks themselves have been shown to undergo minimal deformation, the FE models could be made more efficient by reducing the numbers of elements used to represent them. This could convey considerable advantage in reducing the computer
run times. This would allow the behaviour of larger models, with more realistic boundary conditions, to be explored.

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In general, it is important to study the macroscopic properties such as mechanical and thermal properties from a viewpoint of mesoscopic microstructure in order to develop a microstructurally controlled new materials with necessary material properties as well as to develop design and/or maintenance methods of structural materials. The authors have investigated the mesoscopic microstructure related mechanical properties as well as thermal properties of ceramics for the purpose of understanding of relationship between properties and microstructure. In the paper, our recent activities of mechanical properties, strength and Young’s modulus, are described on the basis of mesoscopic microstructure.

In the field of strength research, brittle fracture model taking account of the mesoscopic microstructure was proposed by Burchell. Several researchers have expanded the model so as to treat the strength under stress gradient condition, e.g. bending strength etc., since the model was applicable only to uniform stress condition; namely, the model can predict only tensile strength. Furthermore, we have modified the model so as to treat interaction effect between crack and pore in order to predict a wide range of the strength such as strength under oxidation condition etc. Moreover, the performance of the mesoscopic microstructure based fracture model was investigated from a comparison with the Weibull strength theory, which is generally applied in the ceramics strength research.

In the field of the Young’s modulus research, a prediction model has been developed. In the model, the interaction is considered between ultrasonic wave and mesoscopic pores. Three kinds of wave/pore interaction modes are treated in the model; 1) direct wave mode, which means that the wave has no interaction with pores, (2) creeping wave mode, the wave impinges on pores and then it propagates around the pore as a creeping wave and (3) direct wave mode, the wave impinges on pores and scatters away. The model can predict the ultrasonic wave propagation velocity as well as the height of echo signal due to pores, and the Young’s modulus is obtained on the basis of wave propagation theory within a solid. The MonteCarlo simulation technique was also used so as to predict the nonuniformly distributed pore arrangements.

Generally, presenting these models are basically applicable to the other ceramic materials, and their applicability to ceramics materials, such as superplastic material of 3Y-TZP zirconia (3 mol% Yttria stabilized Tetragonal Zirconia Polycrystal), silicon carbide, etc., is discussed.
Advanced graphite oxidation models

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Graphite burning in severe air ingress accidents of HTRs or fusion reactors and other graphite oxidation processes in these systems are calculated by computer models, which are based up to now on isothermally measured kinetic equations for chemical reactions of oxidation regime II; external mass transfer is considered in these codes by common mass transfer rules, but internal mass transfer (in-pore diffusion) and elementary chemical reaction on the internal surfaces, which together are the relevant phenomena in the oxidation regime II, are not separately treated. This paper lines out the limitations of this procedure, which are mainly:

- Deviations from real behaviour in case of nonisothermal oxidation.
- No information about depth of penetration of the oxidation process within the graphite pore system.

A computer model for advanced graphite oxidation (PROFIL), which considers explicitly in-pore diffusion and chemical reaction (both dependent on the graphite burn-off, which itself changes during oxidation within the depth of the graphite pore system) is explained. Input data for this code, which are mainly burn-off dependent chemical (regime I) reactivities of graphite against oxygen and effective (in-pore) diffusion coefficients, are presented. By comparison of PROFIL-calculations with experiments in regime II it is shown, that a sufficient agreement is obtained only, if erosion as graphite removal mechanism is considered in addition to graphite gasification.

Besides that, possible graphite oxidation model improvements for regime III (external mass transfer control), which consider interactions between mass transfer and chemical reaction, leading to a mass transfer and graphite oxidation rate increase by up to an order of magnitude, are discussed.
Irradiation Creep In Graphite - A Review

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In the presence of fast neutron irradiation graphite will creep at a much faster rate than would be the case without irradiation.

It was first realised that graphite was creeping in reactors in the early 1950s. Before this it had been thought that shrinkage and thermal stresses in graphite blocks would severely limit the life time of graphite moderated reactors. Work on irradiation creep carried out since the early 1950s has given rise to the following conclusions:

1. Graphite exhibits primary, secondary and tertiary creep.
2. Tertiary creep appears to be dependent on the type and magnitude of loading i.e. compression or tension.
3. If the creep rate is normalised by the unirradiated Young’s modulus, a common irradiation creep law can be applied to all nuclear graphites.
4. Primary creep is recoverable with time.
5. Poisson’s ratio in creep has been measured and appears to be the same as, or similar to, the elastic Poisson’s ratio.
6. The increase in modulus due to irradiation pinning (or hardening) does not affect creep rate.
7. Higher dose irradiation structural changes modify the creep rate. (This partially accounts for the tertiary creep.)
8. Irradiation creep modifies the Coefficient of Thermal Expansion (CTE) of graphite. (A similar effect has been observed on measuring the CTE of unirradiated specimens under static loading.)
9. Data on the change in CTE transverse to the loading direction measured on irradiated graphite is limited and unclear. However the tests on unirradiated samples described above indicate that the lateral effect is the negative of the direct effect and its magnitude is given by the unirradiated Poisson’s ratio times the direct effect.
10. The load dependence of tertiary creep appears to be explained by a change in the dimensional change rate induced by the change in CTE.

The basis and theory behind these statements is described in this paper.
An Analysis of Irradiation Creep in Nuclear Graphites

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Nuclear graphite under load shows remarkably high creep ductility with neutron irradiation, well in excess of any strain experienced in unirradiated graphite. As this behaviour compensates, to some extent, some other irradiation effects such as dimensional change or thermal shutdown stresses, it is an important property. This paper briefly reviews the approach to irradiation creep in the UK, as described the UK Creep Law. It then offers an alternative analysis of irradiation creep, using the AGR moderator graphite as an example, to high values of neutron fluence, applied stress and radiolytic weight loss.

The UK Creep Law

The total creep strain, $\varepsilon_c$, in irradiated graphite is simply the sum of primary and secondary creep components. Primary creep is characterised by a rapid (reversible) deformation that soon decreases in rate. Secondary creep strain is characterised by a linear dependence on stress, $\sigma$ and neutron fluence, $\gamma$, and an inverse proportionality to the Young's modulus, $E$. In the UK, the creep strain is described by the UK Creep Law such that

$$\varepsilon_c = \frac{\sigma}{E_c}(1 - \exp(-4\gamma)) + 0.23\frac{\sigma}{E_c}\gamma$$

and

$$E_c = E_{0.5}SW$$

where $E_c$ and $E_{0.5}$ are the irradiated effective and unirradiated static Young's modulus, respectively, $S$ is the structure term which represents the structural disruption by irradiation damage, $W$ is given by $\exp(-bx)$ where $b = 3.4$ (AGR graphite) and $x$ is the fractional weight loss. Generally, the UK Creep Law represents creep data well for neutron fluence up to $\sim 60 \times 10^{20}$ n/cm$^2$ (EDN). At higher fluence, the UK Creep Law progressively underestimates the measured creep strain, $\varepsilon'_c$. Originally, the structure term was added to account for this discrepancy between the measured and predicted creep strain, however this term has limited success, no satisfactory physical meaning and cannot be measured directly.

A New Approach

The original definition of creep strain is the difference in length between a stressed and a control specimen. Originally, it had been assumed that the properties measured on a control sample exposed unstressed apply also to the stressed sample. However, as dimensional change is directly proportional to the Coefficient of Thermal Expansion (CTE) and since the CTE is modified by the creep strain, it has been recently realised that the dimensional change component in the stressed specimen cannot be the same as that in the control specimen. That is, the true creep strain, $\varepsilon_c$, will be higher in tension and lower in compression than previously assumed. In the evaluation of creep strain, the following correction to the creep strain is required.
\[ \varepsilon_c = \varepsilon_c' - \int_0^\gamma \left( \frac{\alpha_x' - \alpha_x}{\alpha_x - \alpha_a} \right) \frac{dX_T}{d\gamma'} d\gamma' \]  

where \( \alpha_x, \alpha_x', \alpha_a \) and \( \alpha_c \) are the CTEs of the bulk unstressed and crept graphite in direction \( x, a \) and \( c \) axes of the graphite crystallites, respectively, and \( X_T \) is a crystal shape change parameter. Using the above correction, and without reference to a structure term, this paper presents a new analysis to generate a relationship between the apparent creep strain, \( \varepsilon_c' \), and neutron fluence, \( \gamma \), for high levels of neutron fluence, applied stress and simultaneous radiolytic weight loss. The analysis provides a more physically satisfying explanation for the decrease in the apparent creep strain observed at high fluence, and, using AGR moderator graphite data, can be shown to improve the prediction of creep strain compared with the UK Creep Law.

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Planning for Disposal of Irradiated Graphite: Issues for the New Generation of HTRs

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The disposal of radioactive wastes is an issue which is exercising the authorities in numerous countries. Most plan to dispose of wastes in underground repositories, but the execution of such plans have been fraught with difficulty and delay, and no adequate analysis of comparative risks seems to have been applied to the issue.

In the specific cases of graphite and carbon-composite components, the argument for consideration of other disposal options is strong, especially when all relevant factors are assessed on a technical basis. One may conclude that it is entirely the political agenda, based upon inadequately analysed environmental perceptions, which has led to the intent to dispose of carbonaceous wastes by land burial.

In this paper, the numerous alternatives are discussed in the context of existing and potential HTR carbonaceous materials as intended for use in reflectors and fuel compacts, highlighting the issues which the designer needs to take into account in preparation for the eventual decommissioning of his plant and disposal of the irradiated material. Particular reference is made to the valuable experience gained from the successful dismantling of the Fort St. Vrain plant, and the value of applying this experience at the design stage is highlighted.
The International Database on Irradiated Nuclear Graphite Properties contains data on the physical, chemical, mechanical and other relevant properties of the graphites. Its purpose is to provide a platform that makes these properties accessible to approved users in the fields of nuclear power, nuclear safety and other nuclear science and technology applications. The Database is constructed using Microsoft Access 97 software and has a controlled distribution by CD ROM to approved users. This paper describes the organisation and management of the Database through administrative arrangements approved by the IAEA. It also outlines the operation of the Database. The paper concludes with some remarks upon and illustrations of the usefulness of the Database for the design and operation of HTR.
Session V

Basic Studies on HTGR Fuel Fabrication and Performance

Chaired by
J. Kuijper (NRG, the Netherlands) and R. Moormann (FZJ, Germany)
Dramatic Fission Product Release from Irradiated Nuclear Ceramics

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Crystalline oxide ceramics, more particularly zirconia and spinel, are promising matrices for plutonium and minor actinide transmutation in specifically-devoted nuclear reactors, due to their high melting point, their reasonable thermal conductivity and their strong resistance against irradiation. An important issue of research concerning these materials is the evaluation of their ability to confine radiotoxic elements resulting from the fission of actinides. The present paper reports the study of the diffusion and release upon annealing or irradiation at high temperature of a very toxic fission product (Cs) in zirconia and spinel. The foreign species are introduced by ion implantation and the diffusion is studied by nuclear microanalysis (Rutherford backscattering experiments). The results emphasize the decisive influence of the fission product concentration on the diffusion properties. In both matrices the Cs mobility is strongly increased when the impurity concentration exceeds a threshold of the order of a few atomic per cent. Irradiation with medium-energy heavy ions is shown to enhance Cs diffusion with respect to annealing at the same temperature. The effects described in this letter have to be taken into account when developing innovative nuclear fuels, in order to decrease the risk of accidental release of radiotoxic elements into the geosphere.
Method of Reactor tests of HTGR Fuel Elements at an Impulse Loading

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The method of HTGR fuel element tests in conditions of impulse loading (during several tens seconds) in the impulse graphite reactor (IGR) is described. Experiments with spherical fuel elements have shown reliability of the control system providing maintenance and measurement of the specified level of temperature and energy release.

The program for computation of temperature fields in fuel elements and coated fuel particles and temperature stresses distribution along section of fuel elements was developed. As illustration, the results of fuel elements tests are presented at an energy release of 22 kW.
Gas Reactor TRISO-Coated Particle Fuel Modeling Activities at the Idaho National Engineering and Environmental Laboratory

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The INEEL has begun the development of an integrated mechanistic fuel performance model for TRISO-coated gas-reactor particle fuel termed PARFUME (PARticle FUel ModEl). The objective of PARFUME is to physically describe the behavior of the fuel particle under irradiation. Both the mechanical and physico-chemical behavior of the particle under irradiation is being considered. PARFUME is based on multi-dimensional finite element modeling of TRISO-coated gas reactor fuel. Important phenomena include:

- Gas production in the kernel,
- Thermal expansion, shrinkage, swelling and creep of the different layers under irradiation
- Cracking of layers as a result of irradiation induced stresses and the anisotropic response of the pyrolytic carbon layers to irradiation (shrinkage, swelling and creep that are functions of temperature, fluence and orientation/direction in the carbon)
- Failure behavior of the PyC and SiC layers of the coating system using the classic Weibull formulation for a brittle material

Data on the critical structural properties (e.g., fracture strength, elastic modulus, Poisson's ratio, creep constants, bond strength between layers) of the different layers (i.e., pyrocarbon, SiC) have been assembled from the existing database and correlated as functions of irradiation temperature, fast neutron fluence and morphology of the layer where possible. Extensions to other coating systems (e.g., ZrC) will also be considered.

The physico-chemical model is based on tracking the chemical changes of the fuel kernel during irradiation (changes in carbon/oxygen, carbon/metal and/or oxygen/metal ratio depending on the kernel fuel type) and its influence on fission product and/or kernel attack on the particle coatings. Specific phenomena include

- Fission product inventories as a function of burnup and enrichment of the particle
- Fission product chemistry (chemical equilibrium of the fission products and fuel based on the chemical potential in the particle),
- Gas production (CO and CO2) from free oxygen produced by fission
- Fission gas release and swelling,
- Kernel migration,
- Fission product diffusion, migration and segregation, and

The model will focus on carbide, oxide and oxycarbide uranium fuel kernels. Extensions to other fissile and fertile materials may be considered in the future. Beyond the mechanical and physico-chemical aspects of the fuel, the goal is to develop a performance model for particle fuel that has the
proper dimensionality and still captures the statistical nature of the fuel. The statistical variation of key properties of the particle associated with the production process requires Monte Carlo analysis of a very large number of particles to understand the aggregate behavior. Thus, state-of-the-art statistical techniques are being used to incorporate the results of the detailed multi-dimension stress calculations and the fission product chemical interactions into PARFUME.

This paper reviews the current status of the model, discusses calculations of TRISO-coated fuel performance for cracking observed in recent US gas reactor irradiations, and presents predictions of the behavior of TRISO-coated fuel at high burnup that is currently under consideration in Europe.

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