Summary Report

Workshop on High-Temperature Gas-Cooled Reactor Safety and Research Issues

October 10-12, 2001
Rockville, MD

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Safety and Research Issues

Table of Contents

EXECUTIVE SUMMARY

From October 10 through 12, 2001, the U.S. Nuclear Regulatory Commission (NRC) hosted a workshop on high temperature gas-cooled reactor (HTGR) safety and research issues, at its headquarters in Rockville, MD. In an information paper titled "Future Licensing and Inspection Readiness Assessment," SECY-01-0188, dated September 17, 2001, the staff made a commitment to the Commission to develop an advanced reactor research plan to support efficient and effective licensing reviews of future reactors. The focus of the FLIRA report was on assessing skills and resources required for NRC to be able to effectively conduct the licensing process for the near-deployment reactor designs. These future reactor designs include two high temperature gas-cooled reactors (HTGRs) — Pebble Bed Modular Reactor and the GT-MHR, and two advanced light water-cooled reactors (ALWRs) — AP-1000 and IRIS. The FLIRA report also discussed the need for developing regulatory infrastructure and for conducting selected anticipatory and confirmatory research to support advanced reactor licensing.

The focus of this workshop was on identifying key HTGR safety issues and the need for future research, including independent tools and data that NRC would need to develop to support licensing reviews of new HTGR designs. Also discussed were various transient and off-normal scenarios that could result in the release of radioactive material. Priorities were assigned to various topics which would be helpful in planning future research programs and assessing and allocating optimum resources.

This report contains the highlights of the workshop. Appendix A contains input received from the European Union on their HTGR research programs. Appendix B of this report includes the workshop agenda. A list of the participants and their affiliations is included in Appendix C. The highlights of HTGR-related experience and current research efforts in various countries as well as issues that need further examination are summarized in the tables contained in Appendix D. A list of acronyms is included in Appendix E.

The workshop was attended by various invited national and international experts from the Federal Republic of Germany, United Kingdom, European Union (represented by the German delegate), Peoples Republic of China, Japan, the Russian Federation, Republic of South Africa, International Atomic Energy Agency (IAEA) (part time), as well as from the U.S. Department of Energy (DOE), various DOE national laboratories, two members of the NRC's Advisory Committee on Reactor Safeguards (ACRS), a representative of the Massachusetts Institute of Technology (MIT), and independent consultants discussed various HTGR safety and research issues. No nuclear reactor designers, developers, vendors, or potential applicants and licensees were invited. The invited experts are knowledgeable of the HTGR design and technology, including ongoing HTGR-related research in the countries and organizations they represented.

The workshop discussion included the following topics: high-temperature material performance; nuclear-grade graphite behavior; TRISO-coated fuel performance; containment performance as well as the issue of containment v. confinement; adequacy of the existing data and analytical tools, including thermo-fluid dynamics codes as well as severe accident analysis codes; and consideration of various accident scenarios including air and water ingress, loss of forced circulation, reactivity insertion, and seismic events, which could lead to the release of radioactive material.

The current status of HTGR-related research in the participating countries and of efforts under the auspices of IAEA were discussed. Several key safety issues that warrant further examination and may be likely candidates for future cooperative research, were also identified.

The following research topics were considered to be of high priority:

(A) High-temperature material performance -- creep-fatigue data; environmental characteristics; and in-service inspection and surveillance plan and techniques;

(B) Nuclear-grade graphite behavior - measurements of changes in physical properties induced by thermal, radiation and chemical exposures; oxidation measurements in the event of an air-ingress accident; and in-service inspection plans and techniques;

(C) Fuel performance - irradiation testing of fuel simulating steady state, reactivity insertion, and slow heat-up during transients, including fission product release data;

(D) Containment performance - evaluation of containment v. confinement option for all accident scenarios, radiological source terms, and emergency planning;

(E) Adequacy of data and analytical tools - developing thermo-fluid dynamics codes as well as severe accident analysis codes; data for code validation and assessment; experimental verification of pebble movement; impact of likely non-uniformity of the central reflector column; and development of probabilistic risk assessment models and approaches; and

(F) Accident scenarios - modeling air and water ingress events and their implications; fission product release in an air environment at prevailing post-accident temperatures; fuel behavior under reactivity insertion accidents; implications of core geometry changes on progression of accident sequence; and seismic margins.

The participants concluded that the information developed on important safety issues and research needs was beneficial in identifying high priority research topics. The priorities assigned to various key issues will be helpful in planning future research as well as facilitating international cooperative efforts. The NRC believes that the insights developed at the workshop will serve as a significant input to its developing an advanced reactor research plan in early 2002, which will guide NRC's future HTGR research programs. The workshop also significantly contributed to the development of the NRC staff's expertise and knowledge related to HTGR design and technology and understanding of the key safety issues which need careful consideration for conducting an effective and efficient licensing process.

I INTRODUCTION

From October 10-12, 2001, the U.S. Nuclear Regulatory Commission (NRC) hosted a workshop at the NRC headquarters in Rockville, MD, USA. The focus of this workshop was on high-temperature gas-cooled reactor (HTGR) safety issues and the need for future research. It was attended by national and international experts on HTGR safety. To facilitate a candid discussion, the workshop participation was by invitation only, and it was intentionally kept free of nuclear reactor designers, developers, vendors, and potential applicants and licensees. Various national and international experts from the Federal Republic of Germany, United Kingdom (UK), European Union (represented by the German delegate -- a letter from the European Commission, dated October 3, 2001, is included in Appendix A), Peoples Republic of China, Japan, the Russian Federation, Republic of South Africa (RSA), International Atomic Energy Agency (IAEA) (part time), as well as from the U.S. Department of Energy (DOE) and various DOE national laboratories, two members of the NRC's Advisory Committee on Reactor Safeguards (ACRS), a representative of the Massachusetts Institute of Technology (MIT), and independent consultants discussed various HTGR safety and research issues. These experts are knowledgeable of HTGR design and technology, including ongoing HTGR-related research in the countries and organizations they represented.

The purpose of the workshop was to discuss HTGR safety issues, identify research needs, and assign priorities as input to the development of an integrated advanced reactor research program to support the review of future HTGR designs. Timely implementation of a comprehensive research program is crucial for developing independent data and tools to support an effective and efficient advanced reactor licensing process.

Appendix A contains input received from the European Union on their HTGR research programs. Appendix B of this report includes the workshop agenda. A list of the participants and their affiliations is included in Appendix C. The highlights of HTGR-related experience and current research efforts in various countries as well as issues that need further examination are summarized in the tables contained in Appendix D. Appendix E includes a list of acronyms.

II BACKGROUND

In a report on Future Licensing and Inspection Readiness Assessment (FLIRA), SECY-01-188, dated September 17, 2001, the staff made a commitment to the Commission to develop an advanced reactor research plan. It was envisaged that for conducting effective and efficient licensing reviews of new reactor designs, the NRC would need to develop independent capabilities to judge the safety of the proposed design and confirm supporting information submitted by applicants. To accomplish this, the NRC would need to plan and conduct in a timely manner selected confirmatory and anticipatory research to develop necessary tools and data to judge the HTGR applicant/licensee's safety claims. Such an approach has been used in the past and has been proven to contribute to the quality, thoroughness and timeliness of staff reviews.

The NRC considers this workshop as an important step in understanding the HTGR experience and status of related research in various countries, identifying and prioritizing topics for future research, assessing prospects for future cooperation, and using these insights for developing an advanced reactor research plan.

III CONDUCT OF WORKSHOP

In sponsoring this workshop, the NRC's objectives were to draw upon international experience and knowledge to identify HTGR-related safety issues and the need for future research. The workshop participation was by invitation only to provide a forum for candid discussion among various national and international experts on HTGR safety and research issues. Although, it was not intended that consensus be reached among the experts on various topics, it was expected that the discussions would provide the NRC with many useful insights in assessing the HTGR design, technology and safety issues that warrant additional considerations.

The 2-1/2-day workshop commenced with welcome and an introductory speech by Thomas King, Director, Division of Systems Analysis and Regulatory Effectiveness (DSARE), Office of Nuclear Regulatory Research (RES). Following introductions, Ashok Thadani, Director, RES, welcomed the guests and stressed the importance of this forum in helping NRC to plan, develop and implement a sound advanced reactor research program to support an effective and efficient HTGR licensing process. Chairman Meserve, in his remarks, affirmed NRC's commitment to continue to ensure public health and safety while conducting HTGR licensing reviews. He emphasized the importance of this workshop in helping NRC identify the key safety and research issues related to the HTGR design, technology and operation, indicating that these insights will serve as key considerations in formulating NRC's future HTGR research program to develop the necessary tools and information base for conducting effective and efficient future licensing reviews. He considered international cooperation vital in NRC's future research endeavors.

The NRC staff presented an overview of the PBMR design and highlights of the current pre-application review process. There was a brief discussion of the GT-MHR design. The representative from South Africa presented a status of the PBMR licensing review in that country. The MIT representative discussed the safety and research issues identified in MIT's pebble bed project. The workshop discussions were organized by topial areas as follows:

(i) high-temperature materials performance;

(ii) nuclear-grade graphite behavior;

(iii) fuel performance and qualification;

(iv) containment performance;

(v) adequacy of data and analytical tools, such as, thermo-fluid dynamics codes and severe accident analysis codes; and

(vi) consideration of various accident scenarios including air and water ingress, loss of forced circulation, reactivity insertion events, and seismic events.

It was agreed that for each major topic, the participants would be requested to discuss the relevant international research experience including the efforts in the countries or organizations they represented. Various facets of each key topic which justify further investigations would be identified. At the end of the workshop, this list would be re-examined to discuss priority for future research.

IV DISCUSSION

The following are the highlights of the workshop discussions on key safety and research topics:

IV. A High-Temperature Materials

IV.A.1 Issues

During operation, various HTGR moderator, reflector and structural elements as well as system components will be exposed to higher temperatures than those in the conventional light water-cooled reactors (LWRs). Therefore, issues that need further consideration would include: (i) applicability of the existing database of currently qualified high-temperature materials, including the impact of various coolant impurity levels, to the specific HTGR applications; (ii) the adequacy of procedures for evaluating material properties for HTGRs; and (iii) in-service inspection examination and surveillance plans and techniques.

Thermal stresses in pipes that are insulated by glass wool encased in a stainless steel casing were discussed. Crevices would naturally exist in the insulation, which raised some questions: (i) What is the effect of gases migrating between the spaces, and consequently causing hot spots and thermal stresses? (ii) What happens to the concentration of chemicals/impurities trapped in the crevices? (iii) How often is insulation replaced? (iv) What is the potential impact on pipes of degradation of the casing and the insulation, and of hot spots and deposition materials in the crevices? (v) Are there other locations, not just within the insulation in the pipes, where crevices may exist and could possibly be a problem?

Since HTGRs will operate at high temperatures and the coolant will never be totally free of contaminants, it is important to identify the detrimental effects of the coolant impurities on the gas turbines. It is believed that helium cycle is less stressful on turbine blades. However, it is important to assess consequences of erosion and corrosion by carbon dust, fission products, and other coolant impurities.

IV.A.2 Pertinent International Experience and Research

China

For exterior components, where the temperatures are not as high, HTR-10 uses stainless steel 316. There is limited experience in assessing impact of coolant impurity levels on high temperature materials performance. Currently, China uses the AGR data from UK and AVR/THTR data from Germany.

Germany

During AVR and THTR operation, Germany did not encounter any high temperature material problems. However, the PBMR temperatures are expected to be considerably higher. It is crucial that the high temperature materials issues that need to be addressed should include not only corrosion but also erosion because of particulate contaminants in the circulating gases. For traditional materials, industrial experience should also be considered for applicability.

Some of the specimens removed after decommissioning of AVR have been studied and documented in a report. These investigations have included crystallographic examinations, material property testing, and determining whether materials were used beyond their creep/fatigue-life. AVR did have some instances of air and water ingress; however, over the life of the plant, the reactor pressure vessel suffered no unacceptable damage. At present there may not be sufficient resources to conduct additional tests on the AVR specimens, and the possibility of sending AVR samples out of Germany is not clear; some THTR components have been sent to South Africa.

European Union

Some materials irradiation tests are currently being planned in Europe. The HTGR research programs sponsored by the European Union include testing new materials for possible HTGR applications. These materials are not currently being used in nuclear power plants. The issue of coolant impurities, especially, oxygen, and cobalt in view of erosion[1] and likely plate-out on turbine blades along with fission products, is being addressed.

HTR-M project aims at obtaining material data for key components including the reactor pressure vessel, and other in-vessel high temperature materials as well as turbine applications. Efforts include: review of RPV materials and development of a materials property database, testing to be performed on RPV welded joints and irradiated specimens at Petten HFR to investigate tensile, creep, and/or compact tension fracture; compilation of the existing data of high-temperature materials employed as reactor internals and planning of future R&D efforts; compilation of data related to turbine disk and blade materials and planning of future related R&D efforts; review the state-of-the-art techniques on determining graphite properties to set up a suitable database and perform oxidation tests at high temperatures on a fuel matrix graphite to obtain kinetic data for advanced oxidation (THERA facility at Julich) and advanced carbon-based materials to obtain oxidation resistence in steam and air respectively (INDEX facility at Julich).

Japan

For HTTR, two-chrome-one-moly alloy has been used for the pressure vessel. There is a practice of maintaining low coolant impurity levels to control adverse impact. Japan has studied the impact of coolant impurities on materials performance and has a non-electronic database for various impurity types and levels. Tests were conducted in oxygen environment in the 600-650^{o}C temperature range. Tests have also been conducted on stainless-304 and -316, Alloy 800H, and Hastaloy-XR in oxygen environment. Limited

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[1] One of the US participants elaborated on the matter of erosion, especially on particulate content of the flowing gas, based on some information that was obtained during US delegation's recent visit to China. The participant identified the phenomenon of carbonization from the gases and plateout of other particles on various surfaces. However, it was reported that at present there is insufficient information to conclude whether carbonization can be a problem.

testing has been conducted at 980-1000°C. There are published reports on material developments, and these have been incorporated in the Japanese Code Specifications, which are different than the American Society of Mechanical Engineers (ASME) codes.

Russia

The Russian Federation, along with the GT-MHR designers, is exploring various elements for high-temperature applications. The materials under consideration are both conventional and new materials that are being developed for high-temperature applications.

South Africa

The regulator has similar general concerns as does the NRC. It is likely that PBMR will use stainless steel-304 or -316; however, the PBMR licensee would need to furnish supporting evidence that the material will last through the life of the plant. Although conventional gas turbine data are available, for PBMR it will be necessary to (i) develop bases for selection of various material in high temperature applications; (ii) know limiting conditions for applications; and (iii) establish testing and in-service inspection plans and surveillance techniques.

United Kingdom

Like Japan, UK has its own materials codes and does not use the ASME codes. Therefore, direct extension of UK materials qualification data for US applications may be difficult. Furthermore, because of the steam cycle, the exit gas temperatures in the AGRs are limited to about 600°C. Data at that temperature and 1050 psi are available. However, in the Brayton cycle, one would expect greater high-temperature challenges. Therefore, it was recommended an NRC research program include materials studies under prevailing HTGR conditions.

UK has encountered fatigue, vibration and erosion problems in the AGR pipes. Because of vibrations, the pipe insulation has experienced major integrity problems. The studs that hold the cover plates do show fatigue. Much relevant experimental work has been done in UK. It is believed that consideration of HTGR design details is important and both inside and outside insulation in various pipes need to be evaluated.

United States

Creep and creep-fatigue life of high-temperature materials are important considerations in the HTGR applications. It is believed that non-destructive testing of decommissioned AVR in-service components may yield significant insights in this respect.

Two classes of high-temperature materials are used in gas-cooled reactors -- low carbon steel and various other alloys. Under off-normal conditions, the components could be exposed to temperatures as high as 1000°C which can last for 1000 hours or longer. Code Cases for expansion joints are being developed. The ASME Code Case 499 allows carbon steel applications under limited conditions. Recently, a modified nine-chrome-one-moly alloy has also been accepted into this code case. However, NRC has not yet accepted and endorsed Code Case 499. Therefore, its acceptability is yet to be determined.

Stainless steel -304 and -316, and a quarter-chrome-one-moly steel alloy that could be exposed to up to 1400°C is being tested for GT-MHR. Some of these materials have been tested in helium environment; however, coolant impurities could significantly affect the high temperature materials performance. Carbon-carbon composite materials can withstand as high temperatures as does graphite[2]. Some data are available.

Electric Power Research Institute (EPRI) is currently funding an international database of past gas-cooled reactor experience on the contaminants and fission products in circulating helium. This effort is expected to be completed by the end of the year 2001. The EPRI database will be helpful in deciding on decontamination techniques and choices of possible blading material for future rotating machinery for the Brayton Cycle.

Cracking problems were reported in the Fort St. Vrain steam generators (SG). There were two incidents of SG leaks. However, the root cause could not be determined as the licensee could not get a sample.

IV.A.3 Examples of High Priority Research Needs

Topics to be pursued with additional research include:

– Creep-fatigue data
– Environmental characteristics
– In-service Inspection plans and techniques

IV.B Nuclear-Grade Graphite Behavior

IV.B.1 Issues

There is a need to establish an information base related to the long-term performance and behavior of nuclear-grade graphite under high temperatures and radiation levels expected during normal operating and accident conditions in the HTGRs. The issue of the loss of structural integrity of nuclear-grade graphite also needs careful consideration because it is one of the key issues which would impact the long-term performance of graphite structural elements and the top- and bottom-reflector as well as the end-of-life behavior of all graphite elements, including the moderator balls. It is also important to understand graphite oxidation behavior under accident conditions, such as, air ingress.

Various graphite production variables, including coke source, manufacturing process, impurities, uniformity of batches and samples within a batch; and other parameters such as density, isotropy, strength, fracture toughness, grain size and crystalline size are important considerations. Furthermore, the effect of temperature, radiation (e.g., burn-up, maximum fluence, radiation levels, cumulative life-time dose), chemical attack, and oxidation need to be understood to assess changes in the physical characteristics of nuclear graphite, such

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[2] Oak Ridge National Laboratory and General Atomics have conducted extensive testing of Alloy 800H in helium environment. ORNL and GA have also built and conducted prototype testing on high temperature carbon-carbon composite for control rod clads as an option for the designer to replace Alloy 800H.

as, swelling and shrinkage; creep; cracking; corrosion; distortion; weight loss and porosity changes, which can have implications on its structural integrity.

There are several questions that would have to be addressed: (i) Can “new” graphite be produced to perform at the same level as the “old” graphite? Since “new” graphite will be produced not only with "old" graphite technology but also with new source of feed material, various physical characteristics, such as, grain size, crystallite size, isotropy, fracture toughness, and uniformity, of the “new“ graphite would also need to be assessed for application in the current HTGR designs. Can “old” graphite data be extrapolated to the “new” graphite? (ii) What should be the scope of a robust graphite qualification program, specifically for assessing impact on physical properties because of thermal and irradiation effects, chemical attack, and oxidation? and (iii) What in-service inspection and surveillance plans and techniques are needed for monitoring graphite performance?

iV.B.2 Pertinent International Experience and Research:

The following are the highlights of the country-wise graphite experience and issues:

China

For HTR-10, China imported graphite from US. No new experimental data exist. An appraisal of in-vessel graphite is admittedly very difficult, and the best way to minimize the loss of structural integrity issues of in-vessel graphite components is to limit neutron fluence. British data are available and are considered to be applicable to HTR-10.

European Union

It is proposed that in the HTR-N1 project, structural graphite -- side reflector -- from the decommissioned AVR will be studied. As part of the HTR-M project, which began in November 2000, the planned efforts include review the state-of-the-art techniques for determining graphite properties to set up a suitable database and perform oxidation tests at high temperatures on a fuel matrix graphite to obtain kinetic data for advanced oxidation (THERA facility at Julich) and advanced carbon-based materials to obtain oxidation resistence in steam and air respectively (INDEX facility at Julich). Objectives of the HTR-M1 project include long-tem testing of the materials for the turbine and irradiation tests for graphite components. Since the previous graphites are no longer available because of the depleted coke source and non-existent production techniques and equipment, the project includes verification of models describing the graphite behavior under irradiation and thermal distortions and screening tests for graphite properties. This project was expected to start in November 2001.

Germany

No graphite problems were encountered during either AVR or THTR operation. Germany has AVR off-normal operational data, including air and water ingress events as well as subsequent core flooding. In-service inspection at AVR involved pebble removal and carbon dust removal. Fretting of graphite blocks as a consequence of loss of structural integrity was observed. During AVR decommissioning, a huge cavity in the central reflector column was noticed. Its formation was attributed to thermal distortion and erosion by the circulating gases.

Japan

In HTTR, Japan has used high purity graphite. The HTTR operates at comparatively low radiation levels. No problems have been identified thus far.

South Africa

It is expected that the suppliers for PBMR will use the graphite that is available in the market. The reflector graphite may have to be replaced every 5-6 years; however, no replacement criteria were discussed. South Africa expects to use the UK AGR graphite data. It is also believed that it would be worthwhile to take an independent look at possible graphite degradation in the PBMR. In the event of a seismic event, the core could actually get deformed. The impact on core geometry would need to be assessed because the ensuing configuration may be completely out of the design basis assessments. Other issues that need further examinations include thermal distortions and radiation-induced embrittlement.

Russia

The Russian nuclear-grade graphite comes from a plant in Siberia. It is a new type of graphite. Extensive cooperation is ongoing between the republics of the former Soviet Union regarding assessment of graphite properties. Russia believes that no final HTGR design should be approved without independent experimental qualification of graphite.

United Kingdom

UK has an extensive advanced gas-cooled reactor operating experience. The AGRs employ CO_2 as a coolant and consequently, most of the British data are in a CO_2 environment. Some of this information may not be directly applicable to the currently planned HTGRs that employ helium as a coolant. A comprehensive in-service inspection plan and surveillance program is recommended for monitoring possible graphite degradation.

United States

Fort St. Vrain used high-purity graphite for the fuel blocks, but not as pure a graphite was used for core support. The latter had a high iron content which was oxidized by moisture resulting in serious loss of strength. However because of extensive design margins, no structural problems were encountered. Two fuel blocks, however, cracked as a result of stress-induced lattice crack between coolant holes and the outside of the blocks. Additionally, because of moisture ingress, the FSV licensee, in agreement with the NRC, instituted a surveillance program and at each refueling, remotely examined the core support graphite blocks to ensure that the cracking problem did not continue. It is recommended that at PBMR, in-service examination of graphite moderator balls, using a statistically valid sample size, should be conducted.

A recent report by Electric Power Research Institute (EPRI), “Graphite for High Temperature Reactors,” dated August 2001, examines nuclear-grade graphite for HTGR applications and compiles pertinent data.

Some of the standards established by the American Society for Testing and Materials (ASTM) may be applicable to nuclear-grade graphite (e.g., C781-9, "Standard Practice for Testing Graphite and Boronated Graphite Components for High-Temperature Gas-Cooled Nuclear Reactors;" and future replacement of E525, "Standard Practice for Reporting Dosimetry Results on Nuclear Graphite," that was discontinued in the year 2001 but no replacement yet has been announced). Applicability of other ASTM standards which have been used for testing graphite properties for non-nuclear applications of graphite and may also be applicable to the HTGR graphite. The existing standards may have to be modified and new standards may need to be developed.

IAEA

Various IAEA Coordinated Research Programs (CRPs) and publications[3], such as, TECDOC-690, TECDOC--901, TECDOC–1198, TECDOC--1154, IWGGCR--11, IWGHTR--3, deal with the subject of world-wide research and experience related to nuclear-grade graphite. Especially noteworthy are the following:

A specialists' meeting was held on the subject of graphite development for gas cooled reactors at the Japan Atomic Energy Research Institute (JAERI) in September 1991. This meeting was attended by representatives from France, Germany, Japan, the Russian Federation, the UK and the US. Papers were presented in the topical areas of graphite design criteria, fracture mechanisms and component tests; graphite materials development and properties; and non-destructive examinations, inspections and surveillance of graphite materials and components. TECDOC--690 contains the details.

In 1995, a "Specialists Meeting on Graphite Moderator Lifecycle Behaviour" was held in Bath, UK. Recognizing that many experts in the field are nearing their retirement with no apparent replacement of qualified professionals in the field, the IAEA's objective in sponsoring this meeting was to establish a central archive facility for the storage on irradiated graphite. Twenty-seven papers were published where the experts representing their countries shared the ongoing graphite research and other pertinent experience. Details of international research activities are included in TECDOC--901.

With support from Japan, South Africa, and the UK, the IAEA has established a database related to irradiated nuclear graphite properties[4]. The objective of this effort is to preserve the existing world-wide knowledge on the physical and thermo-mechanical properties of irradiated graphite, and to provide validated data source to the member countries with interest in graphite-moderated reactors or development of the HTGRs, and to support continued improvement of graphite technology applications. The database is currently being developed and includes a large quantity of data on irradiated graphite properties, with further development of the database software and input of additional data in progress. On-line access will be available to the IAEA member countries. This database is expected to be operational in the year 2003.

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[3] http://www.iaea.org/inis/aws/htgr/abstracts/index.html

[4] http://www-amdis.iaea.org/graphite.html

Under the auspices of IAEA, the objectives of the International Working Group on Gas Cooled Reactors (IWGGCR) are to identify research needs and exchange information on advances in technology for selected topical areas of primary interest to HTR development, and to establish within these topical areas, a centralized coordination function for the conservation, storage, exchange, and dissemination of HTGR-related information. The topical areas identified include irradiation testing of graphite for operation to 1000°C. The duration of this CRP is from 2000 through 2005. This IAEA program is discussed in detail TECDEOC--1198.

NEA

Various NEA conferences held in the past few years have covered the subject of nuclear-grade graphite:

From September 27-29, 1999, NEA/OECD held in Paris the first information exchange meeting on "Survey on Basic Studies in the Field of High Temperature Engineering."[5] The conference was co-sponsored by JAERI. Component behavior, including graphite performance, under normal and accident conditions were discussed. Some of the topics presented include status in the UK and the Netherlands of research relevant to irradiation of fuels and graphite for HTGRS; oxidation of carbon based materials and air ingress accidents in HTR-modules being studied at Julich; graphite selection for the PBMR reflector; study of crack growth in nuclear; the modeling of dimensional change in nuclear graphite; and irradiation effects on carbon-carbon being investigated in Japan.

On October 10-12, 2001, there was an NEA/OECD conference held on "The Second Information Exchange Meeting on Basic Studies in the Field of High Temperature Engineering," in Paris. In the afternoon of the 11th, there was a session dedicated just to "Basic Studies on Behavior of Irradiated Graphite/Carbon and Ceramic Materials including Their Composites under both Operation Storage Conditions" - 8 papers were presented - the last one on the status of the IAEA Graphite Database. Proceedings are not yet available.

International Standards

International cooperation is also crucial in establishing consensus standards, as well as for developing acceptance and performance criteria, for nuclear-grade graphite. It is important to determine which existing national and international standards are applicable to the nuclear-grade graphite, and what, if any, new standards should be developed as acceptance criteria for physical characteristics and operational performance of graphite in HTGR applications. Various ASTM standards would need to be examined for applicability to the nuclear-grade graphite in the new HTGRs.

IV.B.3 Examples of High Priority Research Needs:

Topics to be pursued with additional research include:

– Property measurements as a function of irradiation, temperature, etc.

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[5] **http://www.nea.fr/html/science/htemp/iem1/session1.html**

– Oxidation measurements
– In-service inspection and surveillance plans and techniques

IV.C. Fuel Performance

IV.C.1 Issues

HTGR fuel qualification and performance warrant independent assessments of the licensee submittals. The safety claims of the HTGR design are inherent in the assumption of predicted performance of the TRISO-coated fuel particles under potential accident conditions. The HTGR fuel uses higher enrichment and operates at higher temperatures than the conventional LWRs. The value of 1600°C is typically quoted in the published literature as the maximum permissible fuel temperature beyond which some degradation of the silicon carbide protective coating occurs. Several questions need to be addressed: Is there any information on the effect of temperature gradients across the protective SiC layer? Are the current fuel performance data complete? Are they sufficient? What level of confidence do we have in the existing data? What additional severe accident and transient analyses need to be evaluated? What fuel heat-up profiles need to be used to simulate key accident scenarios? What kind of experiments need to be conducted to simulate fast or slow reactivity insertion scenarios? Can AVR fuel qualification tests be applied to PBMR? What additional data are needed because of the proposed HTGR operating conditions? What other scenarios should be considered that have not been previously examined? How will batch-to-batch fuel qualification be ascertained? What fuel performance models are available and how reliable they are? What data are needed to develop analytical models to support the HTGR licensing process?

Other important items that need consideration are: fission product release, transport, and plate-out. The impact of post-accident temperature, and air and water ingress on fission product release and on chemical forms of fission products also need to be understood. Additionally, the fission product particulate behavior in helium environment (as compared to the steam environment encountered in the LWRs) needs to be examined.

IV.C.2 Pertinent International Experience and Research

China

Using the German equipment and technology, China was able to replicate German quality of fuel; however, their effort was built upon 20 years of experience of producing coated particle fuel. China also improved the fuel by employing a superior gelation process. For HTR-10, testing was done before fuel loading, and at 30,000 - 60,000 MWD/MTU burn-up levels, step-by-step. However, no fission product release measurements were done. For now, China has accepted the 1600°C fuel operational limit; however, more experiments using fuel pebbles are planned for the next 2-3 years. For HTR-10, power density limits have been imposed.

European Union

The objectives of the HTR-F project (which began in October 2000) are to enhance the HTGR fuel fabrication capability in Europe; to qualify the fuel at high burn-up, with a high reliability; and study innovative fuels that are different than employed in the previous HTRs. The following activities are included: (i) collect data from various types of fuels tests in the past in the European reactors and analyze them to get a better understanding of the fuel behavior and performance under irradiation; (ii) define various in-pile and out-of-pile experiments to qualify the fuel particle behavior under irradiation and high temperatures; (iii) model the thermal and mechanical behavior of coated fuel under irradiation and to validate it against the available experimental data; and (iv) review the existing technologies for fuel kernel and coated particle and fabricate first batches of fuel kernels and particles to characterize them and to study alternate coating material, such as, ZrC and TiN. As part of the HTR-F1 project, which complements HTR-F and was expected to begin in November 2001, complete irradiation of the German pebbles in the HFR in Petten is planned to carry out their post irradiation examination (PIE) and to perform heat-up tests under accident conditions. Code developed in HTR-F to model the thermal and mechanical behavior of the coated particles needs to be validated.

Germany

During 20+ years of AVR operation, design of the German fuel kernel remained unchanged. However, current kernel design may be different than the AVR fuel kernel for which extensive experimental data exist at various irradiation levels and duration, as well as range of temperatures and heat-up rates. Nevertheless, if the new coated fuel particles and kernels are manufactured with quality and specifications equivalent to their German counterparts, and it is done so with adequate reproducibility, then there is no reason why the AVR test data could not be extended to the new fuel.

Japan

Beginning some 20 years ago, Japan developed its own process for fabricating the TRISO-coated fuel particles, Failure rates of 1E-03 have been observed with large – 600 micron diameter – particles. The objective is to achieve 1E-04 to 1E-06 failure fractions. The HTTR has a low power density, hence, the operating temperature is limited. Japan has studied fuel behavior under simulated transients and accident conditions, typically at 1350°C but not exceeding 1600°C. The Japanese fission product release data confirm the German results. The burn-up is limited to 1 GW/MTU. Conservative design data for HTTR accident conditions have been published

Russia

Russia began a fuel qualification program using the German test models. There has been extensive Russian-German collaboration on fuel qualification. Models developed for predicting fuel behavior are analytical. Quality control and quality assurance are an integral part of the Russian fuel qualification program. Currently, Russia has an ongoing fuel qualification program for GT-MHR. Irradiation testing of small fuel samples, including TRISO-coated particles and fuel compacts that are specifically intended for GT-MHR, is planned. The test data will be available in few years. Current activities are focused on Pu fuel disposition in the HTGRs.

South Africa

There are concerns regarding the PBMR fuel qualification program, therefore, specifications for each phase are needed to establish confidence in the equivalence of the PBMR and the German Fuel. That is, the PBMR fuel should be of such quality that it will survive under all postulated operating and accident conditions and that the fission product releases will remain within acceptable limits under all foreseeable conditions. Tests need to be conducted to also show that the fuel failure fraction will be acceptable. There is a good chance that certain fuel failure mechanisms may not be obvious when the samples are slowly heated in the laboratory as compared to under the conditions that simulate, say, reactivity surge during a transient. There is also considerable difference in the performance of the fresh v. irradiated fuel; the fresh fuel data are of rather limited importance. Therefore, simulation of actual transient and accident scenarios is crucial to the PBMR fuel qualification program.

The 1600°C limit has not been accepted as the maximum allowable fuel temperature. It is recognized that there are many other influential factors, such as fluence, burn-up levels, pulse or ramp-and-hold heating, rate of heating, etc. that are known to affect the fuel behavior. For the PBMR fuel qualification program, the licensee must substantiate that the PBMR fuel is of the same quality as the German fuel. Then, appropriate tests need to be conducted and test data will have to be examined to determine if the fuel performance is acceptable in view of the conditions simulated, the type of tests conducted, number of kernels tested, and confidence levels in the test data. It must be established that the fuel can withstand the anticipated temperature limits. This must be demonstrated by actual validation of fuel performance. As far as duplicating the German fuel manufacturing or qualification process is concerned, because of the batch-to-batch variations, it will be difficult to ascertain with confidence that the fuel produced is always of the same acceptable quality. It is not just for each fuel pebble but for each kernel that there must be an assurance that it is of the same quality. If AVR fuel data were to be applied to the PBMR fuel, then the German fuel qualification program must be faithfully replicated, with a reasonable confidence. This must be demonstrated each time for each batch of fuel. There are inherent uncertainties, therefore, fuel quality must be proven with appropriate tests. Because the PBMR operating conditions; e.g., power peaking factor, radial flux, temperature profiles, will be very different than those in AVR, it must also be shown with confidence that the German-equivalent fuel will work just as well in the PBMR. Another design difference is the central reflector column in the PBMR which is an altogether different situation than in the AVR core. If the central reflector column does not remain uniform, what will be the consequences.

United States

The published literature on fuel qualifications typically states that as long as the fuel temperature does not exceed 1600°C, there are no significant fission product releases. DOE is presently evaluating the possibility of conducting tests at the ATR facility and possibly other test reactors. To be able to plan and conduct future tests which would deliver the most useful information, it is important to know what tests were done with the German fuel and what additional testing is needed. Once the information gap and the anticipated transients are known, confirmatory tests can be planned. Another point for PBMR (or GT-MHR) fuel qualification is to know what operational conditions existed at the German reactors and what conditions will be expected in the new HTGRs. Since the HTGR operating conditions will be significantly different than in AVR, the HTGR fuel would have to be tested under prevailing operating conditions. The German tests, however, could serve as the base matrix and the new tests could be planned to replicate those experiments albeit simulating actual HTGR operating conditions. This is assuming that the new TRISO-coated particles are fabricated in the same manner and to the same specifications as the German fuel. Naturally, batch-to-batch differences would need to be accounted for by implementing an exhaustive quality assurance and quality control program. Another possibility would be to test the actual fuel fabricated at the Pelindaba plant. It is noted that PBMR aims to fabricate fuel equivalent to the German fuel; however, equivalence of the PBMR and the German fuel must be demonstrated. If fuel equivalence is demonstrated successfully and with reproducibility, then the German fuel tests may adequately encompass a range of parameters - temperature, thermal gradients, fluence, and burn-up levels. Regardless, for optimum benefit, it is crucial that the tests conducted are not just the ramp-and-hold type tests, but that they simulate conditions that are within the realm of PBMR accident /transient scenarios and faithfully represent post-accident temperature profiles.

Most HTGR fuel testing and acceptance criteria have focused on slow heat-up transients and maintaining fuel temperatures below 1600°C. However, reactivity insertion transients could result in different fuel failure. Hence, consideration of reactivity insertion events would require employing different models and criteria. There is little data at present to establish such criteria. Accordingly, additional data and models are needed to understand fuel behavior under sudden and gradual insertion of reactivity. Owing to the fuel design differences between the PBMR and the GT-MHR, and the existence of the central graphite column in the PBMR and the control rod location in the GT-MHR, the reactivity insertion scenarios to be considered will vary by the reactor type. Additional needed research should also be planned by considering various design-specific features of the two reactor types as well as various external initiators, such as seismic events. Changes in control rod geometry and possible rod jamming incidents and subsequent loss of reactor scram should also be examined. Also to be considered are any changes in the core geometry as a result of either loss of structural integrity of the graphite components or damage to the moderator and/or fuel elements, such as, pebble jamming and local changes in the pebble packing fraction in case of PBMR. Various German and Russian ventures as well as experiments at the Sandia National Laboratory may provide a relevant information base.

IV.C.3 Examples of High Priority Research Needs:

Various issues that need be addressed include:

- fuel behavior and limits under reactivity insertion accidents;
- fission product release and transport under accident conditions;
- accelerated vs. real-time irradiation fuel testing; and
- applicability of previous fuel test results to current fuel fabrication and operation issues.

IV.D. Containment Performance

IV.D.1 Issues

For HTGRs, the key issue is whether there should be a containment or a confinement building. Specifically, fission product transport in the event of an off-normal situation, which may have significant impact on the radiological source term, and consequently on emergency preparedness, needs to be examined. Furthermore, the impact of external events that could alter the core geometry, thus rendering it into an unanalyzed configuration, needs careful consideration.

IV.D.2 Pertinent International Experience and Rationale for the Choice

China

For HTR-10, China evaluated the option of containment v. confinement and chose a confinement building on the basis of low fission product release. It is vented for initial filtered release. Thereafter, it reseals and is maintained at a negative pressure. No specifics of accident source term or emergency planning details were discussed.

Germany

Germany had evaluated the two options and chose a confinement for AVR as well as THTR. A 65-mm diameter pipe break was the design basis event. The resulting fission product release, however, did not warrant a containment. The confinement was designed to vent for initial release, after which it would reseal and be maintained at a negative pressure. It was mentioned that in Germany, emergency planning is not the responsibility of the national government but is of the local authorities.

Japan

For HTTR, Japan opted for a containment. It is a steel structure designed to withstand a pressure as high as 4.6 bar. An 80-cm diameter pipe break was used as the design basis accident. No details of source term or emergency planning were discussed.

South Africa

The issue of containment v. confinement is yet to be considered. Risk perspectives will be used to evaluate the two options. The IAEA dose criteria will be used to set the limits for allowable source term. Emergency planning details also remain undetermined.

Russia

Russia expects to opt for the containment option for the Pu burning HTGR, with a steel and re-inforced concrete structure. Details of radiological source term, emergency planning are yet to be considered.

United Kingdom

UK had considered both the containment and the confinement options and chose confinement for the AGRs.

United States

The issue of HTGR containment v. confinement will need serious consideration. Fuel qualification program for TRISO-coated particles, design basis accidents as well as severe accident scenarios, and subsequent fission product release and transport, resulting radiological source term, and risk assessment perspectives all will play a crucial role in preparation of the staff's proposal and recommendation to the Commission for a containment or a confinement. Ultimately, it is a policy decision that the Commission will have to make.

IV.D.3 Example of Research Needs:

Topics to be pursued with additional research include:

– Thorough evaluation of advantages and disadvantages of the containment vs. confinement for all transient and accident scenarios
_ Implications of both options on the ensuing radiological source term
_ Emergency planning considerations

IV.E. Analytical tools

IV.E.1 Issues:

Independent data and tools will be needed to confirm the predicted HTGR performance. Various accident scenarios, such as air and water ingress, and loss of forced cooling - both pressurized and de-pressurized - would need to be appropriately modeled. Unique design features, such as the central reflector column in the PBMR, would require additional design-specific analyses. Validation of analytical tools using plant data, other experimental data or the use of testing via a prototype or demonstration plant need be considered. Furthermore, probabilistic risk assessment tools may have to be developed by considering appropriate models, approach, and data.

IV.E.2 Pertinent International Experience and Research

China

For HTR-10, China has used the German data and tools.

European Union

None reported.

Germany

Germany has extensive experience in modeling and predicting the AVR/THTR performance. Additional work on the HTR Module may be applicable to the HTGRs.

Japan

For HTTR, Japan has developed independent data and tools to predict plant performance under a range of normal operating conditions and various transients and accident scenarios.

Russia

For the GT-MHR-related efforts, extensive development work is ongoing in Russia.

South Africa

For PBMR licensing, South Africa believes that extensive independent assessment of plant performance under various accident scenarios would need to be performed. This would require development of independent analytical tools and data.

United States

Appropriate thermo-fluid dynamics and severe accident analysis codes which can model HTGR design specific features and phenomena will be needed to predict the plant performance under normal operation, and during transients and accidents. Some analytical codes, which have been traditionally used for LWRs, could be modified to address the HTGR features and phenomena, including the capability to model air and water ingress. For accident analysis, it is expected that fission product release and transport could be modeled by using, with some modifications, the existing LWR codes.

IAEA

Complementary to the IWGGCR efforts to identify research needs and exchange information related to the selected topics concerning HTGR technology, the IAEA has continued to sponsor efforts in various topical areas to coordinate conservation, storage, exchange and dissemination of information. As discussed in TECDEOC-1198, an IAEA Coordinated Research Program (CRP), that is expected to last from 2000 through 2005, addresses various research topics, including R&D on high burn-up fuel, R&D on component testing of high efficiency recuperator designs, irradiation testing of graphite for operation to 1000°C; and materials development for turbine blades up to 900°C for long creep life. In addition, the IWGGER includes an international forum for thermo-fluid dynamics code comparison using data from HTTR and HTR-10.

IV.E.3 Examples of Research Needs:

Topics to be pursued with additional research include:

– data for code development/validation/assessment
– experimental validation of pebble movement and He flow predictions
– development of probabilistic risk assessment tools - models/approach/data

IV.F. Accident Scenarios

IV.F.1 Issues:

Various accident scenarios need to be independently examined. The scenarios discussed at the workshop include air ingress, loss of forced circulation, and seismic events, and subsequent fission product release in helium environment. Other issues that need to be addressed include implications of core geometry changes and assessment of seismic margins in the plant design.

IV.F.2 Pertinent International Experience and Research

IV.F.2.a Air Ingress

Possible initiators are thermal- and vibration-induced fatigue, seismic events, radiation- and thermal-induced embrittlement; corrosion; and failure of the turbo-machinery.

China

No data reported.

Germany

NACOK data on air ingress and oxidation are available, including natural convection.

European Union

Additional tests and code modeling efforts in progress.

Japan

Some data are available on air diffusion in the HTTR vessel. Some ongoing efforts to study pipe and joint embrittlement and corrosion are ongoing.

Russia

Ongoing GT-MHR related efforts.

South Africa

None in progress.

United Kingdom

UK has extensive experience in conducting the AGR accident analyses. However, the British data obtained in AGRs, which operated at considerably lower temperatures, employed steam cycle and used CO_2 as a reactor coolant, need to be examined for direct application to the HTGRs which operate at considerably higher temperatures and employ helium as a coolant.

United States

N-reactor data. Some of the findings of the NRC's HTGR research program of the 80's may be relevant. The MHTGR pre-application review effort may also be applicable.

Examples of Research Needs:

Topics to be pursued with additional research include:

- Air ingress modeling and implications
- Fission product release and transport in an air environment

– Implications of core geometry on accident response – Hot spots? Seismic?
– Seismic margins

IV.F.2.b Loss of Forced Circulation

It is essential to fully understand consequences of both the pressurized and the depressurized loss of forced circulation. Various issues that need code validation include: heat rejection mechanisms for various accident scenarios and equipment failures; core hot spots, core thermal conductivity changes; concrete exposures to prevailing high temperatures; and changes in thermal conductivity of the nuclear-grade graphite as a function of temperature. Data are available for pressurized LOFC; however, for depressurized LOFC codes need to be benchmarked.

China

For HTR-10, China has studied various break sizes from 10-mm to 65-mm diameter pipes. Future tests for pressurized LOFC are planned.

Germany

SANA experiment data available. Data for uniform pebble packing at a small facility are available; however, scaling issues in order to extrapolate this information to a full-scale facility need to be examined.

European Union

None reported.

Japan

A comprehensive test program is a part of the Japanese licensing process. Currently, no depressurized LOFC tests are planned. Vessel cooling for HTTR is being studied by a joint venture of nine countries, and code-to-code data comparisons are planned. This test program is jointly sponsored by the IAEA and JAERI.

Russia

For GT-MHR, pressurized LOFC scenario are being investigated. Depressurized LOFC scenario is still evolving, Associated neutronics tests are also being planned.

South Africa

No ongoing efforts.

United Kingdom

UK has extensive AGR operating experience. No specific ongoing efforts.

United States

Past experience at Fort St. Vrain, especially, four LOFC transients may provide the much needed data for future code validation. ORNL is currently conducting sensitivity studies for prismatic fuel.

IAEA

Experiments conducted under Coordinated Research Project (CRP)-3 sponsored by the IAEA are documented in TECDOC 1163.

Examples of Research Needs:

Topics to be pursued with additional research include:

- Data for the depressurized loss of forced circulation;
- Code validation and code-to-code comparisons;
- Modeling heat rejection mechanisms for various accident scenarios and equipment failures, and assessment of consequences;
- Impact on core - hot spots, conductivity changes, and core reactivity changes induced by changes in the pebble packing fraction;
- Concrete exposures to high temperatures; and
- Changes in graphite thermal conductivity with temperature.

IV.F.2.c Seismic Events

Seismic events, as a class of initiators of an air ingress event or a loss of forced circulation event or sudden reactivity insertion events, need due consideration. Potential impact on plant safety and changes in core geometry and properties need to be evaluated. Control rod jamming is possible and subsequent loss of ability of the reactor to scram need to be considered. Other issues that need to be examined include operator response from a common control room to a multi-module facility in the event of a seismic event, especially in the light of different scenarios developing at different modules.

China

No data available.

Germany

Germany had calculated earthquake-induced reactivity effects which were determined to be insignificant. Also conducted was a 6-foot fuel drop test. Details are unknown.

European Union

No data reported.

Japan

No data reported.

Russia

No data reported.

South Africa

No data reported.

United Kingdom

No data reported.

United Statess

No data reported.

IAEA

None discussed.

Examples of Research Needs:

The following areas need to be investigated:

- Structural response of graphite elements
- Core geometry implications including reactivity insertion
- Graphite property changes with time and service
- Determination of seismic margins; e.g, flow blockage; distortions affecting control rod insertion and the resulting failure to scram; operator response to multiple failure in a multi-module facility; response of shutdown rods; and shutdown system diversity.

IV.F.2.d. Reactivity Insertion Events

Because of time limitations, specific details of reactivity insertion events were not discussed in detail. However, during individual discussion of various topics, such as seismic events and HTGR fuel qualification and performance, the need for consideration of data simulating reactivity insertion accidents was duly recognized. There are some data; however, additional research including developing models to understand fuel behavior under reactivity insertion at different rates, and impact of air and moisture ingress should be evaluated. The fuel design differences between the PBMR and the GT-MHR, and the existence of the central graphite column in the PBMR and the control rod location in the GT-MHR should also be considered. The reactivity insertion scenarios will vary by the reactor type. Control rod jamming and possible loss of reactor "scrammability" would need to examined. Also to be considered are operator response issues in the event of a seismic event at a multi-module facility, where different scenarios could likely develop at different modules. Research should also consider various design-specific features of the two reactor types as well as various external initiators, including seismic events, and potential changes in the core geometry.

V SUMMARY

There is extensive gas-cooled reactor operational experience in Germany and UK, including fuel qualification data from the German AVR and graphite behavior data from the British AGRs. Documented data from Coordinated Research Programs (CRPs) sponsored by the IAEA also provide a significant information base. Both the past operational experience and research data will provide significant insights in planning future international HTGR research programs. HTR-10 and HTTR can play a crucial role in providing the necessary experimental data for code validation. Other ongoing efforts in various countries, such as, air ingress and loss of forced circulation studies in Germany; materials, fuel performance, neutronics and equipment qualification related efforts sponsored by the European Union; zero power neutronics experiments, fuel performance under reactivity insertion accidents, and other programs in support of GT-MHR and HTGR development for Pu disposition in Russia; and CRPs on code validation using data from HTR-10 and HTTR, as well the graphite database being developed under the sponsorship of IAEA are all vital to developing a through understanding and establishing sufficient confidence in the HTGR design, safety and technology issues. Additionally, EPRI is sponsoring some studies on HTGR technology whcih can be of value.

VI Future Plans

The participants concluded that the discussions at the workshop and information developed on important HTGR safety issues, research needs, and priorities were useful in identifying safety issues. These insights will serve as an important input to development of NRC's advanced reactor research plan in early 2002 that will guide its future advanced reactor research program. The workshop discussions also contributed to development of NRC staff expertise and knowledge. They also identified several opportunities for international cooperative research which will be followed upon and the NRC will continue to draw upon the existing domestic and international experience. There will be follow-up efforts with the international partners in conducting future HTGR-related research for optimum mutual benefits and to leverage costs.

APPENDIX A

EUROPEAN COMMISSION
RESEARCH DIRECTORATE-GENERAL
Directorate J - Preserving the Ecosystem – Energy efficiency
Unit 4 :Nuclear fission and radiation protection

Brussels, 3 October 2001
DG Research/Dir.J /4/GVG/ma D(01)

Dr. Thomas L. King, Director

UNITED STATES

NRC - Division of Systems Analysis and Regulatory Effectiveness

WASHINGTON, DC 20555-0001

USA

Subject: Workshop on HTGR Safety and Research Issues

Dear Dr King,

Further to our telephone conversation held yesterday I would like to thank you again for your invitation to attend the subject workshop and to apologise for not being able to participate.

As I told you during our conversation, our resources are very limited at the present moment and the dates chosen for the workshop are in conflict with a number of other relevant events previously committed (e.g. NEA meeting in Paris, GIF meeting in Miami, several kick-off meetings of EC-sponsored projects). These are the main reasons that prevent us to send a qualified representative to the workshop, which we find of high interest.

On the other hand, we have noticed that two EU member states (Germany and the UK) will send representatives to this meeting. One of them, Dr. Gerd Brinkmann (Framatome ANP GmbH) is a contractor in several EC co-sponsored projects (i.e. HTR-L, HTR-E) as well as member of the European network HTR-TN. He is therefore very knowledgeable of the HTGR related research activities under the 5th Euratom Framework Programme (FP5) as well as of the prospects for FP6 (2002-2006). We believe that he should

Rue de la Loi 200, B-1049 Bruxelles/Wetstraat 200, B-1049 Brussel - Belgium - Office: MO75 5/27.
Telephone: direct line (+32-2)295.14.24, switchboard 299.11.11. Fax: 295.49.91.
Telex: COMEU B 21877. Telegraphic address: COMEUR Brussels-Internet: georges.van-goethem@cec.eu.int

be able to brief on the FP5 on-going projects and future FP6 developments as well as on the HTR-TN activities should it be deemed necessary at this workshop. We have already contacted him (who has kindly accepted this request) and provided him with all the necessary materials. In turn, he will report to us about the main discussions and conclusions.

As an advanced information you will find herewith attached a short description of the EC co-sponsored projects related to HTGRs in FP5 and as well as of the European Network on “High Temperature Reactor Technology” (***HTR-TN***). This might help you to better understand the research being undertaken in the EU and to identify potential areas of future co-operation in FP6. Please feel free to distribute it among the participants.

We would be very grateful if you could keep us informed of the outcome of this workshop and of any further developments on this subject.

Wishing you a very fruitful and successful workshop,

Very truly yours,

Georges VAN GOETHEM
Co-ordinator of RTD Activities in Reactor Safety

Cc : Messrs G. Brinkmann (Framatome ANP), H. Forsström, M. Hugon, J. Martín Bermejo

Attachment 1

RESEARCH ON HTRs IN EURATOM FP5

Current EC-sponsored Projects

The nine HTR-related projects selected by the European Commission (EC) form a consistent and structured cluster covering both fundamental research and technological aspects (see table 1). They were selected after two calls for proposals with deadlines 4 October 1999 and 22 January 2001. The latter targeted on complementary R&D activities on HTRs with emphasis on issues which were not possible to address in the former due to budget and scheduling constraints.

Following is a brief description of the objectives as well as the main experimental and analytical activities foreseen within the above-mentioned projects. Around 25 different organisations, representing research centres, universities, regulators, utilities and vendors from 9 EU member states and Switzerland are involved.

Table 1. On going HTR-related research projects in Euratom FP5

Acronym	**Subject of Research**	**Co-ordinator (country)**	**Number of partners**	**Duration (months)**	**EC funding (Million EURO)**
HTR-F	HTR Fuel Technology	CEA (F)	7	48[*]	1.7
HTR-F1					0.8
HTR-N	HTR Reactor Physics and cycle	FZJ (D)	14	54[∇]	1.0
HTR-N1					0.55
HTR-M	HTR Materials	NNC Ltd. (UK)	8	54[□]	1.1
HTR-M1					0.7
HTR-E	Innovative components and systems in direct cycles of	Framatome (F)	14	48	1.9
HTR-L	HTRs licensing safety approach and main	Tractebel (B)	8	36	0.5
HTR-C	HTR Programme co-ordination	Framatome (F)	6	48	0.2

(*) Duration of combined projects HTR-F and HTR-F1

(∇) Duration of combined projects HTR-N and HTR-N1

(□) Duration of combined projects HTR-M and HTR-M1

Projects HTR-F and HTR-F1

These projects are "shared-cost" actions to be carried out by a consortium of 7 organisations (CEA, FZJ, JRC-IAM, JRC-ITU, BNFL, Framatome and NRG) under the co-ordination of CEA. The duration foreseen for the combined projects is 48 months.

The objectives of **HTR-F** are: (i) to restore (and improve) the fuel fabrication capability in Europe, (ii) to qualify the fuel at high burn up with a high reliability and (iii) to study innovative fuels that can be used for applications different from former HTR designs. The project started in October 2000 and its Work Programme includes the following activities:

- to collect data from the various types of fuels tested in the past in European reactors (e.g. HFR, THTR, DRAGON, OSIRIS, SILOE, etc.) and to analyse them in order to better understand the fuel behaviour and performance under irradiation
- to define experimental programmes (in-pile and out-of-pile) in order to qualify the fuel particle behaviour under irradiation and high temperatures. A first irradiation test is planned in the HFR reactor on pebbles from the last German high quality fuel production with the objective to reach a burn-up of 200 000 MWd/t. Concerning the heat-up tests, the Cold Finger Furnace (KÜFA) facility, in which temperatures can reach up to 1800 °C, was transferred from Jülich (FZJ) to Karlsruhe (JRC/ITU) where it will be commissioned after having tested one irradiated pebble.
- to model the thermal and mechanical behaviour of coated fuel under irradiation and to validate it against the experimental results available. The models in existing codes (e.g. PANAMA, FRESCO, COCONUT, etc) will be used to develop a common European code.
- to review the existing technologies for fabrication of kernels and coated particles, to fabricate first batches of U-bearing kernels and coated particles, to characterise them and to study alternative coating materials (e.g. ZrC and TiN). Kernels and particles will be fabricated in different laboratories (two at CEA and one at JRC/ITU) and the first coatings tests will be performed on simulated and depleted uranium kernels.

The programme of **HTR-F1**, which should start in November 2001, is fully complementary of HTR-F. It will enable to complete the irradiation of the German pebbles in the HFR in Petten, to carry out their post irradiation examination (PIE) and to perform heat-up tests under accident conditions in the modified KÜFA facility at JRC/ITU. Also, the code developed in HTR-F to modelling the thermal and mechanical behaviour of the coated fuel particles should be validated. Finally, the production of coated particles and kernels should start at CEA and JRC/ITU.

Projects HTR-N and HTR-N1

These projects are "shared-cost" actions to be carried out by a consortium of 14 organisations (FZJ, Ansaldo, BNFL, CEA, COGEMA, Framatome ANP SAS and GmbH, NNC Ltd., NRG, JRC-ITU, Subatech, and the Universities of Delft, Pisa and Stuttgart) under the co-ordination of FZJ. The duration foreseen for the combined projects is 54 months.

The main objectives of **HTR-N** are: to provide numerical nuclear physics tools (and check the availability of nuclear data) for the analysis and design of innovative HTR cores, to investigate different fuel cycles that can minimise the generation of long-lived actinides and optimise the Pu-burning capabilities, and to analyse the HTR-specific waste and the disposal behaviour of spent fuel. The project started in September 2000 and its Work Programme includes the following activities:

- to validate present core physics code packages for innovative HTR concepts (of both prismatic block and pebble bed types) against tests of Japan's High Temperature Test Reactor (HTTR) and to use these codes to predict the first criticality of China's HTR-10 experimental reactor

– to evaluate the impact of nuclear data uncertainties on the calculation of reactor reactivity and mass balances (particularly for high burn-up). Sensitivity analyses will be performed by different methods on the basis of today's available data sets (ENDF/B-VI, JEFF-3, JENDL 3.2/3).
– to study selected variations of the two main reactor concepts (i.e. hexagonal block type and pebble-bed) and their associated loading schemes and fuel cycles (i.e. the static batch-loaded cores and continuously loaded cores) in order to assess burn-up increase, waste minimisation capabilities, economics and safety.
– to analyse the HTR operational and decommissioning waste streams for both prismatic block and pebble bed types and to compare them with the waste stream of LWR.
– to perform different tests (e.g. corrosion, leaching, dissolution) with fuel kernels such as UO_2 and $(Th,U)O_2$ and coating materials of different compositions (e.g. SiC, PyC) in order to evaluate and generate the data needed to model the geo-chemical behaviour of the spent fuel under different final disposal conditions, i.e. salt brines, clay water and granite.

The **HTR-N1** project proposes to: extend the nuclear physics analysis of HTR-N to the hot conditions of Low-enriched Uranium (LEU) cores with data from HTTR and HTR-10; to investigate the potential to treat or purify specific HTR decommissioning waste (e.g. structural graphite) on the basis of samples taken from the AVR side reflector and to continue the leaching experiments for disposed spent fuel with irradiated fuel (instead of dummies) for initial commissioning of the test rigs. The project is due to start in October 2001.

Projects HTR-M and HTR-M1

These projects are "shared-cost" action to be carried out by a consortium of 8 organisations (NNC Ltd., Framatome, CEA, NRG, FZJ, Siemens, Empresarios Agrupados and JRC-IAM) under the co-ordination of NNC Ltd. The duration foreseen for the combined projects is 54 months.

The objectives of **HTR-M** are to provide materials data for key components of the development of HTR technology in Europe including: reactor pressure vessel (RPV), high temperature areas (internal structures and turbine) and graphite structures. The project started in November 2000 and its Work Programme consists of the following basic activities:

– review of RPV materials, focusing on previous HTRs in order to set up a materials property database on design properties. Specific mechanical tests will be performed on RPV welded joints (Framatome facilities) and irradiated specimens (Petten HFR) covering tensile, creep and/or compact tension fracture.
– compilation of existing data about materials for reactor internals having a high potential interest, selection of the most promising grades for further R&D efforts, and development and testing of available alloys. Mechanical and creep tests will be performed at CEA on candidate materials at temperatures up to 1100° C with focus on the control rod cladding.
– compilation of existing data about turbine disk and blade materials, selection of the most promising grades for further R&D efforts, and development and testing of available alloys. Tensile and creep tests (in air and vacuum) from 850° C up to 1300° C and fatigue testing at 1000° C will be performed at facilities at CEA while creep and creep/fatigue tests in Helium will be performed at JRC.
– review the state of the art on graphite properties in order to set up a suitable database and perform oxidation tests at high temperatures on: (i) a fuel matrix graphite to obtain kinetic data for advanced oxidation (THERA facility at FZJ) and (ii) advanced carbon-based materials to obtain oxidation resistance in steam and in air respectively (INDEX facility at FZJ).

The **HTR-M1** project complements HTR-M, as it concentrates on the long-term testing of the materials for the turbine and irradiation tests for the HTR graphite components. Special attention is put on the fact that previous graphites are no longer available because the coke used as the raw material has either run out and the manufacturer's experience lost, or production techniques and equipment do no longer exist. The work programme includes verification of models describing the graphite behaviour under irradiation and screening tests of recent graphite qualities. The project should start in November 2001.

Project HTR-E

This project is a “shared-cost” action to be carried out by a consortium of 14 organisations (Framatome ANP SAS, Ansaldo, Balcke Dürr, CEA, Empresarios Agrupados, Framatome ANP GmbH, FZJ, Heatric, Jeumont Industrie, NRG, NNC Ltd., S2M, University of Zittau and Von Karman Institute) under the co-ordination of Framatome ANP SAS. The duration foreseen for this project is 48 months and the expected commencement date is December 2001.

This project addresses the innovative key components, systems and equipment related to the direct cycle of modern HTRs. These include turbine, recuperator heat exchanger, active and permanent magnetic bearings, rotating seals, sliding parts (tribology) and the helium purification system. The programme contains both design studies (e.g. Computer Fluid Dynamics and Finite Element analyses) and also experiments (e.g. magnetic bearing tests at Zittau facility, validation tests of the recuperator at CEA’s CLAIRE loop or tribological investigations at Framatome’s Technical Centre).

Project HTR-L

This project is a “shared-cost” action to be carried out by a consortium of 8 organisations (Tractebel, Ansaldo, Empresarios Agrupados, Framatome ANP SAS, Framatome ANP GmbH, FZJ, NRG, and NNC Ltd.) under the co-ordination of Tractebel. The duration foreseen for this project is 36 months and the commencement date is October 2001.

The project proposes a safety approach for a licensing framework specific to Modular High Temperature Reactors and a classification for the design basis operating conditions and associated acceptance criteria. Special attention will be put on the confinement requirements and the rules for system, structure and component classification as well as a component qualification level being compatible with economical targets.

Project HTR-C

This is a “concerted action” to be carried out by a consortium of 6 organisations (Framatome, FZJ, CEA, NNC Ltd., NRG, and JRC) under the co-ordination of Framatome. The duration foreseen is 48 months.

This project, which started in October 2000, is devoted to the co-ordination and the integration of the work to be performed in all the above-mentioned projects. Moreover, HTR-C should organise a world-wide “technological watch” and develop international co-operation, with first priority to China and Japan, which have now the only research HTRs in the world. In order to promote and disseminate the achievements of the EC-sponsored projects, HTR-C will organise presentations in international conferences.

Attachment 2

THE "HIGH TEMPERATURE REACTOR TECHNOLOGY NETWORK" (HTR-TN)

In the beginning of 2000, fifteen EU organisations signed a multi-partner collaboration agreement to set up a European Network on "High Temperature Reactor Technology" hereinafter referred to as the "**HTR-TN**". The agreement does not involve cash flow between the members and all contributions are made in kind. The operating agent and the manager of this network is the JRC-IAM (Petten) and the rest of the partners are: Ansaldo (I), Belgatom (B), BNFL (UK), CEA (F), Empresarios Agrupados (E), Framatome (F), FZJ (D), FZR (D), IKE (D), University of Zittau (D), Delft University (NL), NNC (UK), NRG (NL) and Siemens (D). Many of these organisations had already been working together in the "INNOHTR" Concerted Action of the Euratom FP4 (contract FI4I-CT97-0015).

The general objective of this network is to co-ordinate and manage the expertise and resources of the participant organisations in developing advanced technologies for modern HTRs, in order to support the design of these reactors. The primary focus will be to recover and make available to the European nuclear industry the data and the know-how accumulated in the past in Europe and possibly in other parts of the world. The Network should also work on the consolidation of the unique safety approach and of the specific spent fuel disposal characteristics of HTR, providing data, tools and methodologies which could be available for the safety assessment of European Safety Authorities. The EC-sponsored projects under Euratom FP5 are the initial "kernel" from which the HTRTN has departed.

The activities of this network started officially in April 2000 at the kick-off meeting held in Petten (The Netherlands). During this meeting the Steering Committee of the network was constituted and different task groups were set up in order to implement the agreement. Six technical task groups were created to address the following areas: components technology, system and applications studies, material performance evaluation, safety and licensing, fuel testing, physics and fuel cycle including waste. In addition to these technical task groups some "horizontal" task groups were also formed to cover aspects such as strategies for future common projects, internal and external communications, and international relationships.

At the second Steering Committee meeting of the HTR-TN held in Brussels on November 2000 three new organisations, Balcke-Dürr (D), COGEMA (F) and VTT (FI) joined HTR-TN. The network remains open for further partners or associates from Europe and elsewhere. An HTR-TN web page has been set up by the network members using the «CIRCA» server of the JRC (http:/www.jrc.nl/htr-tn).

APPENDIX B

Appendix B

High-Temperature Gas-Cooled Reactor
Safety and Research Issues Workshop
October 10–12, 2001
Two White Flint North – Room T-2 B3
U.S. Nuclear Regulatory Commission
Rockville, MD 20852

Meeting Objectives

- Discuss and reach agreement on the dominant accident scenarios for HTGRs.
- Discuss and reach agreement on the primary evaluation criterion of criteria to be used in ranking issue importance for each scenario.
- Consider each scenario description, identify the primary phenomena, processes and safety issues for the scenario, and rank each relative to the primary evaluation criterion.
- Discuss research needs (including ongoing research programs) for high-priority safety issues.

Wednesday, October 10, 2001

8:15 a.m.	Check-in at front desk
8:30	**Research Director's Welcome (A. Thadani)**
8:40	**NRC Chairman's opening Remarks (R. Meserve)**
9:00	**Overview of NRC Advanced Reactor Research (A. Thadani)**
9:15	**Scope, Goals and Expected Outcome for Workshop (T. King)**
9:35	**General Description of the Pebble Bed Modular Reactor (PBMR) and NRC's PBMR Pre-Application Activities (S. Rubin and D. Carlson)**
10:20	**GT-MHR General Description (D. Carlson)**
10:40	Break
11:00	**Status of PBMR Licensing Review in South Africa (G. Clapisson)**
11:45	**Safety and Research Issues Identified in MIT Pebble Bed Reactor Project (A. Kadak)**
12:15 p.m.	Lunch
1:15	**Overview of Workshop Structure and Approach (R. Meyer)**
1:45	**Identification of HTGR Event Scenarios – All**
3:15	Break
3:30	**Discussion of Steady State Operational Issues – All**
5:00	**Adjourn**

Thursday, October 11, 2001

8:15 a.m.	Check-in at front desk
8:30	**Discussion of Loss of Forced Cooling Scenarios – All** – *Scenario description* – *Phenomena and issue identification and priority* – *Research needs*
10:30	Break
10:45	Loss of Forced Cooling (Continued)
12:15 p.m.	Lunch
1:15	**Discussion of Air Ingress and Water Ingress Scenarios – All** – *Scenario description* – *Phenomena and issue identification and priority (begin with Previous List/modify)* – *Research needs*
3:15	Break
3:30	**Discussion of Seismic Scenarios – All** – *Scenario description* – *Phenomena and issue identification and priority (begin with Previous Lists/modify* – *Research needs*
5:30	**Adjourn**

Friday, October 12, 2001

8:15 a.m.	Check-in at front desk
8:30	**Reactivity Event Scenarios – All** – *Scenario description* – *Phenomena and issue identification and priority (begin with Previous Lists/modify* – *Research needs*
10:15	Break
10:30	**Summary of Workshop Outcomes – NRC/All**
12:15 p.m.	**Adjourn**

APPENDIX C

Participants in October 10–12, 2001,
High-Temperature Gas-Cooled Reactor Safety
and Reserach Issues Workshop

Name	Organization	Telephone	e-mail Address
Ader, Charles	NRC/RES	(301) 415-0135	cea@nrc.gov
Arndt, Steven	NRC/RES	(301) 415-6502	see@nrc.gov
Bagchi, Goutam	NRC/NRR	(301) 415-3298	gxb1@nrc.gov
Ball, Syd	Oakridge National Laboratory (ORNL)	(865) 574-0415	sjb@ornl.gov
Bari, Bob	Brookhaven National Laboratory (BNL)	(631) 344-2629	bari@bnl.gov
Boyak, Brent	Los Alamos National Laboratory (LANL)		bboyack@lanl.gov
bBrey, H.L.	Self	(970) 476-1537	larrybrey@aol.com
Brinkmann, Gerd	Framatome ANP GmbH	49 913 1189 6630	gerd.brinkmann@framatome-anp.de
Burchell, Tim	ORNL	(865) 576-8595	burchelltd@ornl.gov
Carlson, Donald	NRC/RES	(301) 415-0109	dec1@nrc.gov
Chokshi, Nilesh	NRC/RES	(301) 415-=0190	ncc1@nrc.gov
Clapisson, Guy	National Nuclear Regulator, South Africa	1-21-12-674-7199	gclapiss@nrr.co.sa
Corum, J.M.	ORNL	(865) 574-0718	corumjm@ornl.gov
Cubbage, Amy	NRC/NRR	(301) 415-2875	aec@nrc.gov
Elzeftawy, Med	NRC/ACRS	(301) 415-6889	mme@nrc.gov
Feltus, Madeline	Office of Nuclear Energy, Sicence and Technology, U.S. Department of Energy	(30s) 902-2308	madeline.feltus@hq.doe.gov

Name	Organization	Telephone	e-mail Address
Fischer, David	NRC/NRR	(301) 415-2728	dcf@nrc.gov
Flack, John	NRC/RES	(301) 415-5739	jhf@nrc.gov
Fomichenko, Peter	RRC Kurchatov Institute	7 095 1967479	pf@dhtp.kiae.ru
Gannett, Randy	Sandia National Laboratory (SNL)	(505) 284-3989	rogaunt@sandia.gov
Gasparini, Marco	Division of Nuclear Installation Safety, IAEA	6312600	m.gasparini@iaea.org
Hannon, John	NRC/NRR	(301) 415-1992	jnh@nrc.gov
Horak, W.C.	BNL	(631) 344-2667	horak@bnl.gov
Ibarra, Jose	NRC/RES	(301) 415-6345	jgi@nrc.gov
Jackson, Diane	NRC/NRR	(301) 415-8548	dtj@nrc.gov
Kadak, Andrew C.	MIT		kadak@mit.edu
Kadambi, N.P.	NRC/RES	(301) 415-5896	npk@nrc.gov
King, Thomas	NRC/RES	(301) 415-7499	tlk@nrc.gov
Kress, Thomas	NRC/ACRS		
LaVie, S.	NRC/NRR	(301) 415-1081	sfl@nrc.gov
Lee, Richard	NRC/RES	(301) 415-6795	jel@nrc.gov
Levin, Alan	NRC/OCM	(301) 415-1750	ael1@nrc.gov
Lois, Erasmia	NRC/RES	(301) 415-6560	exl!@nrc.gov
Meyer, Ralph	NRC/RES	(301) 415-6789	rom@nrc.gov
Murley, Thomas	Self	(301) 469-7573	temurley@erols.com
Muscara, Joseph	NRC/RES	(301) 415-5844	jxm8@nrc.gov
Odar, Frank	NRC/RES	(301) 415-6500	fxo@nrc.gov

Name	Organization	Telephone	e-mail Address
Orechwa, Yuri	NRC/NRR	(301) 415-1057	yxo@nrc.gov
Pickard, Paul	SNL	(505) 845-3046	pspickard@sandia.gov
Powers, Dana	NRC/ACRS		
Rae, Alan	NRC/NRLPO	(301) 415-1102	alancrae@hotmail.com
Rao, Dasari V.	LANL	(505) 667-5098	dvrao@lanl.gov
Rubin, Stuart	NRC/RES	(301) 415-7480	sdr1@nrc.gov
Scarbrough, Thomas	NRC/NRR/DE	(301) 415-2794	tgs@nrc.gov
Shoop, Undine	NRC/NRR	(301) 415-2063	uss@nrc.gov
Southworth, Finis	Idaho National Engineering and Environmental Laboratory (INEEL)	(208) 526-8150	fin@inel.gov
Spore, Jay	LANL	(505) 667-7573	jay-spore@lanl.gov
Summers, Lyn	Nuclear Installation Health and Safety, United Kingdom	44 1 519 514109	lyn.summers@hse.gsi.gov.uk
Tanaka, Toshiyuki	Japan Atomic Energy Research Institute	81 29270 7474	ttanaka@hems.jaeri.go.jp
Terao, David	NRC/NRR	(301) 415-3317	dxt@nrc.gov
Tripathi, Raji	NRC/RES	(301) 415-7472	rrt1@nrc.gov
Wright, Steven	SNL	(505) 845-3014	sawright@sandia.gov
Yuanhui, Xu	Institute of Nuclear Energy Technology, China	86 10 627 84808	xuyuanhui@tsinghua.edu.cn

APPENDIX D

Table D-1 High-Temperature Materials

Country	Issue							
	Materials	**Codes**	**Acceptability of Existing Data**	**Independent Evaluation for HTGR Applications**	**Problems and Practices**	**In-service Inspection Plans and Techniques**	**Future Testing Planned**	**Topics for Future Research**
China	SS	–	Use UK and German data	Limited	–	–	–	• Creep fatigue data • Impact of impurities on sweeping gas • In-service examination and inspection plans and techniques • New materials for high temperature applications • Carbon-Carbon composites for control rod clad
Germany	SS	–	–	External AVR data	–	–	Post decomm. testing of AVR specimens	
European Union	New and conventional materials	–	Being investigated	Ongoing	–	–	New materials for HTGR applications	
Japan	304-SS 316-SS Chrome-Moly Alloy Alloy 800 H Hastaloy- XR	Non-ASME	–	–	Low level of contaminants High coolant purity	–	–	
South Africa	TBD	TBD	TBD	–	–	–	Possible testing of post-decomm. THTR components	
Russia	Conventional and new materials for GT-MHR	GT-MHR info per US codes	–	–	–	–	–	
United Kingdom	SS	Non-ASME	–	High pressure High temp. fatigue testing	Fatigue Vibration Erosion	–	–	
United States	Carbon–Carbon Composites Low-C Steel Chrome-Moly Alloy	ASME* (Code Case 499 not endorsed)	TBD	EPRI database by end of 2001 DOE NERI programs	–	–	TBD	
IAEA	http://www.iaea.org/inis/aws/htgr/abstracts/index.html (e.g., IWGGCR-18, IWGGCR-4, IWGHTR–3, IWGGCR-2)							

Table D-2 Nuclear- Grade Graphite Behavior

Country	Issue								
	Experience	Data	Current Nuclear-grade Graphite Qualification Program					In-service Inspection plans and techniques	Topics for Future Research
			Radiation	Thermal	Oxidation	Chemical Attack	Water Ingress		• Applicability of the “old” graphite data to the “new” graphite •Qualification of “new” graphite for HTGR applications • Physical property changes (e.g., growth; stress; corrosion/weight loss; failures; graphite dust generation, deposition and oxidation) • Distortion of structural elements and changes in core geometry • Distortion of control elements and possible failure to scram
China	–	–	–	–	–	–	–	–	
Germany	√	AVR/THTR	–	–	–	–	–	–	
European Union✷	–	HTR-M and HTR-M1	√	√	√	√	√	–	
Japan	–	–	–	–	–	–	–	–	
Republic of South Africa	–	–	–	–	–	–	–	–	
Russia✷✷	New Graphite	–	√	√	√	√	√	–	
United Kingdom	Extensive	AGRs	–	–	√	–	√	–	
United States	Fort St.Vrain N-Reactor data EPRI report	ORNL/TM-13661 GRSAC ✷✷✷	–	–	√	–	√	–	
IAEA	Technical Documents: **http://www.iaea.org/inis/aws/htgr/abstracts/index.html** For example, TECDOC-690, TECDOC--901, TECDOC–1198, TECDOC-1154, IWGGCR--11, IWGHTR-3 IAEA Graphite Database under development: **http://www-amdis.iaea.org/graphite.html**								

✷ See Appendix A of the report for letter from the European Union
✷✷ For GT-MHR
✷✷✷ See Table D-6-a

Table D-3 – Fuel Performance

Country	Issue									
	Acceptability of the existing data (e.g., 1600ºC fuel op. limit)	**Sufficiency of the existing data**	**Independent HTGR fuel classification program**	**Addl. data needed**	**Analytical Models**	**Independent testing for F.P. release**	**Reactivity Tests**	**Transient off-normal behavior**	**Problems**	**Topics for Future Research**
China	√	√	√	–	–	Limited	–	√	–	• Challenge of replicating the German fuel manufacturing process • Kernel-to-environment release mechanisms • Reactivity-initiated accidents and fuel damage mechanisms • Transient testing • Fuel testing under normal operating, design basis, and beyond design basis conditions
Germany	√	√	√	–	–	√	–	–	–	
European Union	TBD	TBD	Ongoing	√	√	√	√	–	–	
Japan	√	√	√	–	–	√	–	√	–	
Republic of South Africa	TBD	TBD	Planned	TBD	TBD	TBD	TBD	TBD	✷	
Russia	√	TBD	√	TBD	TBD	√	√	√	–	
United Kingdom	√	√	–	–	–	–	–	–	–	
United States	TBD	TBD	√	TBD	TBD	TBD	TBD	TBD	✷	
IAEA	**Technical Documents:** http://www.iaea.org/inis/aws/htgr/abstracts/index.html For Example, TECDOC–757, TECDOC–784, TECDOC–978, TECDOC–1163, IWGGCR–13, IWGGCR-25									

✷ Equivalence of the German and the PBMR fuel need to be established.

Table D-4 – Analytical Tools and Data

<table>
<tr><th rowspan="2">Country</th><th colspan="9">Issues</th></tr>
<tr><th colspan="3">Data/Tools available for predicting plant performance</th><th colspan="2">Code Development Modifications Planned</th><th rowspan="2">Experimental Data</th><th rowspan="2">PRA - Models Approach Data</th><th rowspan="2">Prototype Testing Future Efforts</th><th rowspan="2">Topics for Future Research</th></tr>
<tr><th></th><th>Normal Op</th><th>Transients</th><th>Accidents</th><th>Thermal-fluid Dynamics</th><th>Severe Accident</th></tr>
<tr><td>China</td><td>German</td><td>German</td><td>German</td><td>–</td><td>–</td><td>HTR-10</td><td>–</td><td>–</td><td rowspan="9">Experimental data for code validation

Experimental validation of pebble movement and helium flow

Impact of pebble packing fraction

PRA tools -models approach data</td></tr>
<tr><td>Germany</td><td>√</td><td>√</td><td>√</td><td>–</td><td>–</td><td>HTR Modul</td><td>–</td><td>–</td></tr>
<tr><td>European Union</td><td>√</td><td>√</td><td>√</td><td>√</td><td>√</td><td>√</td><td>√</td><td>–</td></tr>
<tr><td>Japan</td><td>√</td><td>√</td><td>√</td><td>–</td><td>–</td><td>HTTR</td><td>–</td><td>–</td></tr>
<tr><td>South Africa</td><td>TBD</td><td>TBD</td><td>TBD</td><td>√</td><td>√</td><td>–</td><td>√</td><td>TBD</td></tr>
<tr><td>Russian Federation</td><td>√</td><td>√</td><td>√</td><td>–</td><td>–</td><td>√</td><td>√</td><td>–</td></tr>
<tr><td>United Kingdom</td><td>√</td><td>√</td><td>√</td><td>–</td><td>–</td><td>–</td><td>–</td><td>–</td></tr>
<tr><td>United States</td><td>TBD</td><td>TBD</td><td>TBD</td><td>√</td><td>√</td><td>√</td><td>√</td><td>TBD</td></tr>
<tr><td>IAEA</td><td colspan="8">Technical Documents: http://www.iaea.org/inis/aws/htgr/abstracts/index.html
TECDOC–757, TECDOC--978, TECDOC--1163,TECDOC-1249, IWGGCR-25</td></tr>
</table>

Table D-5 – Containment Performance

Country	Issue										
	Containment v. Confinement Option Considered	**Containment**	**Confinement**	**Basis**	**Design specs, if known**	**Confinement**					**Source Term Emergency Planning Considerations**
						Vent	**Filter**	**Release**	**Reseal**	**Negative Pressure**	
China	√	X	√	Low FP release	–	√	–	√		√	–
Germany	√	X	√	65-mm pipe break	–	√	–	√	√	–	Local Responsibility
European Union	HTR-L	–	–	TBD	–	–	–	–	–	–	–
Japan	√	√	X	80-cm pipe break	Steel 4.6 bar	–	–	–	–	–	–
Republic of South Africa	TBD	–	–	Risk Perspectives	–	–	–	–	–	–	IAEA Dose Criteria
Russian Federation	√	√	X	Risk Perspectives	Steel with re-enforced concrete	–	–	–	–	–	–
United Kingdom	√	X	√	Low FP release	–	–	–	√	–	√	–
United States	TBD	–	–	Risk Perspectives Policy decision by the Commission	–	–	–	√	–	√	NRC Regulations Safety Goals
IAEA											

Table D-6-A
Accident Scenarios – Air Ingress

<table>
<tr><th rowspan="2">Country</th><th colspan="3">Issue</th></tr>
<tr><th>Available Data</th><th>Challenges</th><th>Initiatives</th></tr>
<tr><td>China</td><td>–</td><td rowspan="8">High temperature graphite oxidation

Code validation

Fuel behavior and FP release in helium environment and after air ingress

Applicability of “old” data to “new” graphite Forms</td><td rowspan="8">Thermally-induced fatigue

Vibration–induced fatigue

External events (e.g., seismic)

Embrittlement

Corrosion</td></tr>
<tr><td>Germany</td><td>NACOK–Natural convection

Graphite oxidation and fuel failure</td></tr>
<tr><td>European Union</td><td>HTR-M and HTR-M1</td></tr>
<tr><td>Japan</td><td>HTTR – air diffusion in vessel</td></tr>
<tr><td>Republic of South Africa</td><td>–</td></tr>
<tr><td>Russian Federation</td><td>–</td></tr>
<tr><td>United Kingdom</td><td>√</td></tr>
<tr><td>United States</td><td>HTGR research program of the 1980's

ORNL/TM-13661 Potential Damage to Gas Cooled Reactor Graphite due to Severe Accidents, April 1999

GRSAC - Graphite Reactor Severe Accident Code</td></tr>
<tr><td>IAEA</td><td colspan="3">Technical Documents: http://www.iaea.org/inis/aws/htgr/abstracts/index.html
For example, TECDOC-784, TECDOC–1163, TECDOC-1198, IWGGCR-25</td></tr>
</table>

Table D-6-B
Accident Scenarios - Loss of Forced Circulation

Country	Issues		
	Available Data	**Challenges to be Addressed**	**Tests Planned**
China	HTR-10	–	–
Germany	10-65 mm diameter double ended pipe breaks SANA experiments	Scaling issue – Applying the small facility data to a full-scale facility	–
European Union	–	–	–
Japan	–	–	Vessel cooling for HTTR for code validation (joint venture with 9 countries)
Republic of South Africa	–	–	Investigating pressurized LOFC Still evolving depressurized LOFC scenario Neutronics tests
Russian Federation	–	–	–
United Kingdom	–	–	–
United States	Ft. St. Vrain – Four LOFC events may serve as data for future code validation		
IAEA	**Technical Documents: http://www.iaea.org/inis/aws/htgr/abstracts/index.html** CRP-3 - Experiments for RCCS/ultimate heat sink -- TECDOC-1163, TECDOC–757, IWGGCR-25		

Table D-6-C
Accident Scenarios - Seismic Events

<table>
<tr><th rowspan="2">Country</th><th colspan="2">Issue</th></tr>
<tr><th>Available Data</th><th>Need to Conduct Research</th></tr>
<tr><td>China</td><td>–</td><td rowspan="9">• Structural response of graphite elements

• Core geometry implications

• Graphite property changes with time and service

• Determination of seismic margins (e.g., flow blockage; distortions affecting control rod insertion and resulting failure to scram; operator response to multiple failures in a multi-module facility.

• Response of shutdown rods.

• Shutdown system diversity</td></tr>
<tr><td>Germany</td><td>Calculated earthquake reactivity effect – not significant

Conducted a fuel drop test</td></tr>
<tr><td>European Union</td><td>–</td></tr>
<tr><td>Japan</td><td>–</td></tr>
<tr><td>Republic of South Africa</td><td>–</td></tr>
<tr><td>Russian Federation</td><td>–</td></tr>
<tr><td>United Kingdom</td><td>–</td></tr>
<tr><td>United States</td><td>–</td></tr>
<tr><td>IAEA</td><td>Technical Documents: http://www.iaea.org/inis/aws/htgr/abstracts/index.html
For example, TECDOC--690, TEDOC--901, TECDOC--1154, IWGGCR--6, IWGGCR--22</td></tr>
</table>

APPENDIX E

ACRS	[NRC] Advisory Committee on Reactor Safeguards
AGR	Advanced Gas Cooled Reactor
ALWR	Advanced Light Water Reactor
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
AVR	**A**rbeitsgemeinschaft **V**ersuchs**r**eaktor
CO_2	Carbon Dioxide
CRP	Coordinated Research Project
DOE	U.S. Department of Energy
DSARE	Division of Safety Analysis and Regulatory effectiveness
EPRI	Electric Power Research Institute
EU	European Union
FLIRA	Future Licensing and Inspection Readiness Assessment
FRG	Federal Republic of Germany
FSV	Fort St. Vrain
GA	General Atomics
GT-MHR	Gas Turbine-Modular Helium Reactor
GW	Gigawatt
HTGR	High Temperature Gas-Cooled Reactor
HTR	High Temperature Reactor
HTTR	High Temperature Test Reactor
IAEA	International Atomic Energy Agency
IWGGCR	International Working Group on Gas Cooled Reactors
JAERI	Japan Atomic Energy Research Institute
KW	Kilowatt
LOFC	Loss of Forced Circulation
MHTGR	Modular High Temperature Gas-Cooled Reactor
LWRs	Light Water-cooled Reactors
MD	Maryland
MIT	Massachusetts Institute of Technology
mm	Millimeter
MW	Megawatt-Days
MTU	Metric Ton Unit
NACOK	Natural Convection in Core with Corrosion
NRC	U.S. Nuclear Regulatory Commission

ORNL	[DOE] Oak Ridge National Laboratory
PBMR	Pebble Bed Modular Reactor
Pu	Plutonium
R&D	Research & Development
RES	[NRC] Office of Nuclear Regulatory Research
RF	Russian Federation/Russia
RSA	Republic of South Africa
SANA	**S**elbsttätige **A**bfuhr der **Na**chwärme bei einem HTR-Modul-Reaktor
SiC	Silicon Carbide
SNL	[DOE] Sandia National Laboratory
SG	Steam Generator
THTR	**T**horium-**H**och**t**emperatur**r**eaktor
UK	United Kingdom
US	United States