UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 25, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Ashok C. Thadani, Director
Office of Nuclear Regulatory Research

SUBJECT: SUMMARY OF THE NRC DELEGATION VISIT TO GERMANY ON SAFETY ASPECTS OF HIGH-TEMPERATURE GAS-COOLED REACTOR DESIGN AND TECHNOLOGY

On July 23-26, 2001, a six-member NRC delegation met with German scientists and engineers. The purpose of the meetings was to open up channels of communication for follow-up exchanges of technical information and to improve the Agency's knowledge and understanding of advanced high-temperature gas-cooled reactor (HTGR) design and technology. The visit was arranged in connection with the NRC staff action plan to expand staff expertise and understanding of world-wide experience in technology specifically applicable to the Pebble Bed Modular Reactor (PBMR) and the Gas Turbine-Modular Helium Reactor (GT-MHR). The delegation consisted of Stuart Rubin and Donald Carlson, Office of Nuclear Regulatory Research (RES), Amy Cubbage and Undine Shoop, Office of Nuclear Reactor Regulation (NRR), Alex Murray, Office of Nuclear Materials Safety and Safeguards (NMSS), and Howard Faulkner, Office of International Programs (OIP). Two days were spent in Cologne and two days were spent at the Jülich Research Center (formerly the Jülich Nuclear Research Center) FZJ. Mr. Edmund Kersting, Head of International Programs, the Company for Reactor Safety (GRS), organized the visit on behalf of the NRC.

Discussions were held on operating and test experience with pebble bed HTGRs. Non-proprietary reports and documents were exchanged, and insights were received on a broad range of technical topics. Discussions focused on: (1) HTGR development in Germany, (2) the German safety assessment of the HTR-Modul and the Thorium High-Temperature Reactor (THTR), (3) safety research and development at Jülich Research Center related to HTGR technology, (4) industrial production and irradiation and post-irradiation testing of pebble fuel in Germany, (5) HTGR nuclear graphite production and testing, (6) pebble bed heat transfer and fluid flow, (7) operating experience and lessons learned from the Arbeitsgemeinschaft Versuchsreaktor (AVR) and the THTR, (8) THTR core physics and pebble flow, (9) the experimental facilities for pebble-bed passive decay heat removal, air ingress, and graphite oxidation at the Jülich Research Center, (10) German HTGR codes and standards, (11) German transfer of HTGR information to ESKOM for development and safety assessment of the PBMR design, (12) the AVR spent fuel intermediate storage facility and the hot cells for irradiated fuel examination, and (13) safety aspects of HTGR spent fuel management. Many follow-up documents were requested and international agreements are being planned to expand NRC's technical understanding of HTGR technology.

A list of the German participants and their affiliations is provided in Attachment 1, and a copy of the full agenda is provided in Attachment 2. Attachment 3 provides a summary of the presentations, discussions and observations during the 4-day visit. Attachment 4 lists the handouts and other documents that were provided in connection with the various presentations, discussions, and tours. Copies are available through the representatives from RES, NRR, NMSS, and IP who participated in the delegation. At the present time, the NRC delegation has not conducted a detailed technical review of the material and, thus, any findings should be considered as preliminary.

Attachments:
1. List of German Participants
2. Agenda for the Visit to Germany
3. Summary of the Visit to Germany
4. List of Handouts and Documents Provided

cc w/atts.:
C. Paperiello, DEDMRS
W. Kane, DEDR
J. Dunn-Lee, OIP
S. Collins, NRR
M. Virgilio, NMSS
A. Szukiewicz, RES
T. King, RES
M. Mayfield, RES
S. Newberry, RES

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DATE	08/22/01*		09/07/01*		08/23/01*		----------		09/25/01*	

Memorandum dated: September 25, 2001

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Distribution w/o atts.:

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JUhle, RES	RCaruso, NRR	SBall, ORNL
CGingrich, RES	TCheng, NRR	
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Visit of the NRC-Delegation to Germany

on the Topic

Safety Aspects of HTR Technology

for Monday, 23 July to Thursday, 26 July 2001

Attachment 2

Monday 23 July 2001, GRS, Schwertnergasse 1, 50667 Köln
Begin: 10:00 a.m., Room 610

Introductory meeting and overview of German activities related to HTR

- Welcome to GRS
- Information about GRS
  (Kersting)
- Mission of the NRC delegation
  (NRC representative)
- Overview on the HTR programme in Germany
  (Schöning)
- Overview on safety assessment of HTR-Module in Germany
  (Nitzki)
- Know-how transfer to ESKOM for a PBMR
  Safety analysis report HTR-Module
  Access to the total HTR-know-how, Consultancy work
  (Schöning, Brinkmann, Kugeler)

Tuesday, 24 July 2001, FZJ, Research Centre Jülich
Begin: 10:00 a.m.

Main Topic:

Research at FZJ related to HTR

- Welcome to the Research Centre Jülich (Eisenbeiß)
- Information on the work of the Research Centre Jülich
  (Eisenbeiß)
- Overview of research and development at the FZJ related to HTR technology
  (Kugeler)
- Fuel element R & D and industrial production in Germany
  (Heidt)
- Fuel element research and development programme, aspects of irradiation and post-irradiation examination: establishment of the retention capability limit temperature of 1600 °C
  (Pott, Nabielek, Schenk)
- Nuclear graphite for the HTR-research, development and industrial production
  (Haag)
- Heat transfer and fluid flow in a pebble bed
  (Barnert, Scherer)

Wednesday, 25 July 2001, Research Centre Jülich
Begin: 10:00 a.m.

Main Topic:

"Operational Experiences of AVR and THTR and Visits of Experimental Facilities"

- AVR operation experiences, test programs, overview highlights, lessons to be learnt
  (Storch, Marnet, Wahlen, Pohl)
- THTR operation experiences, test programs, overview, highlights, lessons to be learnt
  (Dietrich, I. Kalinowski)
- Core Physics and pebble flow
  (H. Kalinowski, Kleine-Tebbe)
- Aspects of waste management
  (Kugeler, Odoj)
- Visit to experimental hall no. IV: experimental work on self-acting removal of decay heat
  (Barnert, Nießen, Kugeler)
- Visit to intermediate storage facility
  (Storch, Marnet)

Optional:

- Visit to AVR
  (Halaszovich)
- Visit to the Hot Cells
  (Duwe, Pott)

Thursday, 26 July 2001, GRS, Schwertnergasse 1, 50667 Köln
Begin: 10:00 a.m., Room 610

Main Topic:
Regulatory Aspects and Safety Assessment

- Safety assessment of HTR module
  (Helmers, Nitzki, Vogel, Brinkmann)
- Safety assessment (Design and operation) of THTR
  (Hofmann)
- Safety issues during licensing of THTR
  (Hohmann)
- Rules and standards
- Final discussion

Summary of July 23–26, 2001, Visit to Germany
On Safety Aspects of High Temperature Gas Cooled Reactor
Design and Technology

INTRODUCTION

Walter Leder, Managing Director of the Company for Reactor Safety (GRS) welcomed the NRC delegation. Mr. Leder gave a short overview of the nuclear power plant situation in Germany. He explained the anti-nuclear stance of the Green political party and their influence in the coalition government. He noted the Consensus Agreement between the federal government and the nuclear utilities to phase out nuclear power over the next 20 years.

Mr. Kersting gave an overview of the GRS [1][1]. He explained that GRS is an organization of technical and scientific experts. They support the federal government in the areas of nuclear safety and waste management. He noted that they have four centers in Germany, each with different areas of specialization. Funding of the company is provided as follows: 77% by the German government, 6% by the European Union and 17% by private contracts. Currently, GRS has 480 staff members and an additional 60 persons are associated with the Institute for Safety Technology, a GRS subsidiary. Funding for 1999 amounted to $45 million.

Mr. Kersting also gave an overview of the nuclear power plant regulatory system in Germany [2]. By law, the supervising regulatory authorities are the individual German states (Länder) and not the federal government. However, the state authorities are subject to federal "supervision" by the Ministry of Environment, Nature Conservation, and Nuclear Safety (BMU). The BMU is assisted by the Federal Office for Radiation Protection (BFS) and receives expert advice from the Reactor Safety Commission (RSK).

Howard Faulkner introduced the delegation and gave a brief presentation on the NRC safety mission and organization. Amy Cubbage gave a presentation on current advanced reactor initiatives in the U.S., the NRC's activities in response to these initiatives, including the establishment of the future licensing organization (FLO), in NRR, the interest of Exelon in the Pebble Bed Modular Reactor (PBMR) and General Atomics in the GT-MHR and the resultant pre-application efforts at NRC [3]. Stuart Rubin gave a presentation on the background and purpose of the NRC visit to Germany. He discussed the industry's recent HTGR pre-application initiatives and some of the challenging design, technology, safety and policy review issues that these initiatives raised. He also outlined what we hoped to learn during the visit [4]. Finally, Donald Carlson offered some comments in German reflecting on his past affiliation (1978–83) with the Jülich Nuclear Research Center. To provide background information for future discussions, he also presented Mr. Kersting with two documents from the NRC's past review activities for the Modular High-Temperature Gas-Cooled Reactor (MHTGR) [5] [6]. There were many questions from the German participants about the renewed interest in nuclear power in

[1]Numbers in square brackets refer to the handouts and documents listed in Attachment 4.

the U.S., especially as it related to HTGRs, including the PBMR. The remainder of the discussions and presentations held during the visit followed technical areas and are discussed under the appropriate section headings.

HIGH-TEMPERATURE REACTOR RESEARCH AT THE JÜLICH RESEARCH CENTER

Dr. Gerd Eisenbeiss, Director of Energy Programs, Jülich Research Center, welcomed the NRC delegation and Professor Kurt Kugeler presented an overview of the Center and its past and ongoing research activities related to high-temperature reactors [7].

The Jülich Nuclear Research Center (Kernforschungsanlage, KFA) was established in 1958 near the city of Jülich by the German state of North-Rhine Westphalia with a central mission of research and development of high-temperature reactor technology. Construction on the 15 MWe Arbeitsgemeinschaft-Versuchsreaktor (AVR), the world's first pebble-bed reactor, began in 1961 at a location immediately adjacent to the KFA, and power production commenced in 1967. The AVR was shut down in 1988 after 21 years of operation as a power reactor and large scale test facility. In 1990, the KFA changed its name to Jülich Research Center (Forschungszentrum Jülich, FZJ) to reflect a decline in emphasis on nuclear reactors.

FZJ now employs 4300 workers, including approximately 1000 student researchers and foreign guest scientists, and maintains close ties with several universities in the region. The five main research areas at FZJ are now Energy, Environment, Life, Information, and Matter. The Center's remaining reactor-related R&D is conducted mainly within the Institute for Safety Research and Reactor Technology (ISR), one of twelve research institutes that comprise FZJ. All reactor-related work in ISR is under the direction of Professor Kugeler, who is also Chair of Reactor Safety and Technology at the nearby Technical University of Aachen and serves on the German Reactor Safety Commission (RSK), an advisory body functionally similar to our Advisory Committee on Reactor Safeguards. It was noted that in the early 1970's Jülich had over 600 research staff members working on reactor safety and technology, whereas only about 60 staff members work in these areas today.

The Jülich Research Center's research and development on pebble-bed reactors has included extensive analytical and computational work in addition to tests and experiments involving large test facilities. Research and development has focused on the design and testing of fission-product-retaining fuel elements, high-temperature steel alloys, major reactor plant components (i.e., compressors, turbines, recuperators, hot gas ducts), and specific components associated with the use of helium coolant (i.e., bearings, penetrations, seals, insulations). Steel alloys were found to require special treatments to avoid helium-induced concerns, such as fusion at joints. Methods were also developed to continuously purify the helium coolant, using molecular sieve and other technologies. In addition to the AVR, Germany's large test facilities have included the EVO helium turbine power plant, the HHV helium turbine test loop, the EVA-II helium-heated steam reformer, and the KVK test loop for helium-to-helium intermediate heat exchangers [8]. The SANA test facility for passive decay heat removal phenomena and the NACOK test facility for reactor air ingress phenomena were briefly described by Professor Heiko Barnert in preparation for the facility tours described later in this report.

In recent years, increasing attention has gone to the study of advanced reactor safety features that go beyond current high-temperature reactor (HTR) design and technology [9]. These

ongoing R&D efforts fall under the heading of what Professor Kugeler calls "catastrophe-free" nuclear technology. Included is the developmental testing of silicon carbide coatings to cover and seal (i.e., isolate) the graphite surface of pebble fuel elements and graphite reflector blocks from chemical attack. If successful, such ceramic coatings would prevent self-sustaining graphite oxidation in the case of a potential air-ingress event such as might result, for example, from a postulated large break in the reactor pressure vessel. Towards eliminating the possibility of such large vessel breaks, the "catastrophe-free" developmental work further includes the design and scaled over-pressure testing of burst-protected reactor pressure vessels made of prestressed steel. Also under consideration are mitigation features that utilize sand or other granulates to block the continued ingress of air after a postulated vessel break and thereby prevent or limit graphite oxidation.

HIGH-TEMPERATURE REACTOR DESIGN ACTIVITY IN GERMANY

Dr. Josef Schöning, General Manager, Company for High Temperature Reactors (HTR), made a presentation on the historical development of high temperature reactors in Germany from the vendor's point of view [10]. In the early years both the Brown Boveri Co. (BBC)/ABB and KWU/Seimens designed and developed HTRs in Germany. In 1989, they entered into a joint venture on a 50-50 basis to form HTR GmbH and mutually worked on a number of subsequent designs until 1993. At this point, design work stopped because the vendors did not see a future commercial application of HTRs.

The only two HTRs to operate in Germany were the 15 MWe AVR and the 300 MWe Thorium High Temperature Reactor (THTR). Both were designed by BBC. The AVR was a prototype that operated successfully for more than 20 years commencing in 1967. The AVR was used for significant research and development activities and served as a test bed for developmental pebble fuel elements with initially BISO and later TRISO coated fuel particles. The THTR was a demonstration plant that operated for less than 4 years. The operating utility decided to shutdown the plant in 1989 for primarily non-technical reasons which mainly involved increased estimates of potential financial risks to the owners and operators. Some significant changes in going from AVR to THTR included (a) moving the steam generator from above the reactor core to beside the reactor core, (b) utilizing a prestressed concrete reactor vessel (PCRV) instead of a double steel reactor vessel, (c) shutdown rods inserted into the pebble region of the core instead of into graphite "noses" on the radial reflector, (d) some modifications to the graphite reflector structure, and (d) the addition of a shutdown decay heat removal system because of the higher power level. Both reactors used pebble fuel elements. Dr. Schöning noted that all German HTRs are intended to have a 3-year test program, 1 year each for individual components, commissioning, and initial plant operation. Overall, government research funding for HTRs was about $1.8 billion, which compares to $2.3 to 3.6 billion for LWRs. AVR and THTR operating and testing experiences are further discussed in subsequent sections of the this report.

In addition to the HTRs that operated, a number of additional designs were developed in Germany to varying degrees. These designs ranged in power level from 10 MWt to 3000 MWt. One of the designs featured in the discussions was the HTR-Modul. The HTR-Modul is 85 MWe, with a design similar to the proposed PBMR except that the HTR-Modul incorporates a steam generator in the power conversion system. This HTR-Modul design was characterized as having a low power density, active and passive safety features, a reactor protection system, and a confinement envelope. Safety features of the HTR-Modul design include a strongly

negative temperature coefficient of reactivity that ensures passive initial shutdown upon loss of forced cooling, passive decay heat removal through the reactor vessel wall after a loss of coolant, a primary pressure boundary incorporating the "leak before break" design criterion, and location of the steam generator and helium cross duct at an elevation below the reactor core. For a postulated break of the helium cross duct, the resulting "diving bell" geometry was said to trap hot helium above the break and thereby delay by several days the onset of air circulation and graphite oxidation in the core. For postulated events without active scram, the core is predicted to passively shut down and then return to critical after a day or two but reach no more than 1% of full power in the absence of forced helium circulation.

SAFETY ASSESSMENT OF THE HTR-MODUL

Dr. Volker Nitzki, Dr. Gerhard Vogel, and Mr. Helmut Helmers, Head of the Division of Energy and Systems Technology, Technical Inspection Society (TÜV)-Hannover, discussed the safety evaluation performed by TÜV for the HTR-Modul design [11] [12]. TÜVs are regional companies that are engaged in safety assessment and inspections of technical equipment. In the nuclear area, they provide technical evaluations to the state regulatory and licensing authorities.

In 1987, HTR GmbH submitted an application for a site-independent license for the HTR-Modul design to the Ministry for the Environment in the German state of Lower Saxony. TÜV-Hannover performed the safety review of the license application as a technical consultant to the state licensing authority. At the time, no technical rules and guidelines were available for the HTR-Modul design and safety assessment. The only available regulations were very specific to the Siemens LWR designs. The existing rules and guidance (laws and ordinances, guidelines, technical rules, and publications) were screened for applicability to the HTR-Modul and concept-specific requirements were added resulting in a comprehensive and consistent set of design and evaluation criteria applicable to the HTR Modul. The TÜV assessment of the HTR-Modul design was based on this set of design and evaluation criteria. Proposed licensing basis events were also reviewed for completeness.and conservatism. This included a screening of LWR events for applicability and expanded to include HTR-Modul specific scenarios.

During the review, additional technical documents (about 250) were requested and received from the applicant. The applicant revised and re-submitted the safety analysis report to address deficiencies identified by TÜV relative to the technical requirements. The revision to the safety analyses resulted in an increase of the fuel design temperature from 1600°C to 1620°C.

In April 1989, as the review was nearing completion, the application was withdrawn for political reasons. The TÜV was requested to continue working on the safety assessment under a contract to the Federal Ministry for Research and Technology. In the final safety evaluation, TÜV concluded that the design of the HTR-Modul could meet the safety requirements imposed on nuclear facilities in Germany at that time. Furthermore, their investigations on risk-reducing measures indicated that the design has inherent safety characteristics that positively affect plant behavior beyond the design basis. The full safety evaluation report is 900 pages. Two documents that summarize the evaluation were provided [12] [13].

In October 1989, the TÜV safety assessment report was provided to the Reactor Safety Commission (RSK) for their review. The RSK stated that the HTR-Modul design has favorable

safety-related characteristics even in the range beyond the design basis, and they concluded that the design of the HTR-Modul met the safety requirements imposed on nuclear facilities in Germany at the time [14].

Additional technical information on the discussions of the safety assessment of the HTR Modul is provided in Appendix A of this attachment.

PEBBLE FUEL ELEMENT RESEARCH, DEVELOPMENT, AND INDUSTRIAL PRODUCTION

Drs. Heit and Froschauer of NUKEM Nuclear GmbH discussed the pebble fuel element research, development, and industrial production in Germany [15]. Topics covered included an overview of the progress of HTR fuel R&D, design of the THTR and the HTR-Modul fuel elements, the process used for manufacturing pebble fuel elements with low-enriched uranium dioxide TRISO particles, the methods used for characterizing the manufactured fuel, production experience, and the special quality assurance system and philosophy for manufacturing German fuel with absolute consistency and the required quality.

Research and development in Germany on HTR coated particle fuel began in 1965 with the development of the BISO coated particle pebble fuel for the THTR. R&D work on TRISO coated particles was initiated shortly thereafter. The German efforts built upon the earlier developments of BISO and TRISO coated particles in the United States, the United Kingdom and Austria, and benefitted from continuing international collaborations. Fuel technology development in Germany was divided among three organizations: BBC which was responsible for fuel element (and reactor) design; NUKEM which was responsible for developing the fuel manufacturing processes, fuel characterization methods, and the manufacture of test fuel elements and production fuel; and the Jülich Nuclear Research Center which was responsible for the fuel irradiation testing and for analyzing and evaluating the test results.

Production of pebble fuel elements for the THTR commenced in the early 1970s and lasted through the late 1980s. Fuel development activities for non-German HTRs utilizing prismatic fuel elements that also incorporated TRISO fuel particles were carried out in the mid-1970s. The goal was to take over the fuel production for the General Atomics HTGRs at a fuel fabrication plant to be built in Germany. However, this effort was discontinued after a few years because of the problems that developed at Fort St. Vrain. Continuing fuel R&D efforts in Germany were then refocused on optimizing the safety and reliability of pebble fuel element design and performance.

German pebble fuel element R&D and production activities included many fuel variations: initially high- and later low-uranium enrichment (for fuel cycle reasons), fuel materials (Th, U, O, C) and coated particle design (BISO particle and later TRISO particle) and manufacturing processes. Many of the fuel element design variations were irradiated in materials test reactors (MTRs) as well as the AVR at Jülich. The AVR served as a large-scale (non-materials test reactor) irradiation facility for the evolving German pebble fuel element designs. Development and irradiation testing on the fuel which was to become the reference low enriched uranium dioxide (LEU) fuel element design for future German HTRs (e.g., HTR-Modul) occurred over a period of about 10 years from the late 1970s to the late 1980s. This fuel was manufactured and irradiation tested and successfully used as the standard AVR reload fuel design beginning in the early 1980s until the reactor was permanently shutdown 1988.

Most research and development activities in Germany on HTR fuel ended in the early 1990s. However, some limited fuel research and development has continued to the present day. The technical knowledge for the HTR pebble fuel that was developed over the years is contained in German records and documents and in the minds of a handful of German fuel experts and specialists. The detailed technical discussions provided by Drs. Heit and Froschauer are further summarized in Appendix B of this attachment.

PEBBLE FUEL ELEMENT IRRADIATION AND ACCIDENT SIMULATION TESTING

Dr. Heinz Nabielek, Jülich Research Center, discussed pebble fuel element irradiation and accident simulation testing conducted in Germany [16] [17] [18]. The release of fission products from TRISO particle pebble fuel elements (or prismatic fuel elements) into the HTR primary circuit come from three sources: free heavy metal contamination located within fuel element graphite matrix introduced by the fuel manufacture process; defective TRISO coated particles from the fuel manufacturing process or from fuel particles that fail due to either the environmental effects of irradiation burn-up (or the effects of a postulated accident heatup) and diffusion through intact TRISO fuel particles. The sequence of release is ^{110m}Ag, ^{137}Cs, ^{134}Cs, ^{85}Kr, ^{90}Sr, ^{106}Ru, ^{95}Zr.

To demonstrate fuel qualification (for in-reactor integrity), the fuel must be (1) manufactured to precise design and manufacturing specifications, (2) irradiation tested over the full range of normal in-core operating conditions and environments, and (3) tested for all postulated off-normal conditions via post-irradiation heatup tests. The quality of the fuel manufacture with respect to defective particles and heavy-metal contamination outside the particle coatings is determined by the destructive burn-leach test of a small sample of manufactured fuel elements from each lot. First the graphite matrix and outer pyrolytic carbon layers are oxidized away at 800°C down to the SiC layer which will not fail at this temperature. The residue particles are then placed in HNO_3 which will leach out all heavy metal not contained within an intact SiC layer. The weight of the uranium in solution is then measured. Since the weight of heavy metal in a single fuel kernel is well defined, it is possible to determine the (effective) number of defective particles in a pebble from the weight of the measured heavy metal in solution. For German reference HTR LEU UO_2 fuel elements, these test results were reported to have met the manufacturing defect rate specification of $6X10^{-5}$.

Irradiation test results presented for German reference fuel irradiated in an MTR to a burnup of about 15% fraction of initial heavy metal atoms (FIMA) shows that the release-to-birth fraction (R/B) of ^{88}Kr (which is an indicator of all gaseous fission products released) in the range of 10^{-8} to 10^{-7}. However, for TRISO particle fuel manufactured and irradiation tested worldwide, ^{88}Kr R/B experience indicates a range of particle defect rate from as low as 10^{-9} to as high as about $5X10^{-3}$.

According to Dr. Nabielek, based on irradiation testing for TRISO particle fuel, the failure rates in an irradiation environment are, in their order of importance: fuel temperature, burn-up, fast fluence, power/temperature gradients and transients, and irradiation time. Models for in-reactor failure of fuel particles have been developed. The models involving a pressure vessel model include the PANAMA and STRESSES codes, while the models based on diffusion coefficients to determine releases from intact defective and broken particles involve the FRESCO code.

Dr. Nabielek suggested that further developmental work on HTR fuel performance might be undertaken. The areas included: (1) reevaluation of the ^{110m}Ag release rates during normal operation for obtaining a better understanding of the source term associated with ^{110m}Ag plate out on the internal surfaces of a direct cycle HTR such as the PBMR. Plate out of ^{110m}Ag is considered a significant potential source of worker exposures in a direct cycle HTR; (2) determination of the effect of fuel burn-ups greater than 10% FIMA on the irradiation performance, including the potential for a reduction in the capability of the TRISO fuel particles to retain fission products up to 1600°C, and (3) development of an improved coated particle failure model for analyzing the performance of fuel particles under accident conditions over 1600°C.

Dr. Nabielek also discussed the results of post-irradiation heat-up tests to simulate postulated fuel heat-up accidents in a helium environment. The heat-up tests involving a temperature ramp up and hold for 500 hours on TRISO coated particle fuel shows that the ^{88}Kr R/B fraction is generally less than 10^{-6} for 1600°C but increase by a couple of orders of magnitude at 1700°C and 3 to 4 orders of magnitude for tests at 1700°C. A slight increase in cesium release (i.e., release fraction increasing from about 10^{-6} to 10^{-5}) is observed after over 200 hours at 1600°C, but these time periods at high temperature are much longer than those predicted for analyzed core heatup events in the HTR-Modul design. At 2100°C, the SiC layer breaks down well within 100 hours. It was noted that the accident simulation heatup tests at 1600 to 1800°C can be used in developing and qualifying the computational models for fuel failure and fission product releases in licensing calculations. Experiments to determine fission product release during depressurization heat-up tests up to 1600°C for German reference TRISO coated particle fuel shows that (a) Cs and Sr are retained in the fuel element, (b) the most important fission product release is iodine – the amount depending on the number of failed particles, (c) the number of defective particles (from manufacture) and the number of additional particles that fail during irradiation and from accidents can only be determined by experimental methods, and (d) the particle failure fraction depends on the quality of the particles.

The following particle failure mechanisms and fission product release effects for German reference HTR TRISO particle fuel elements were presented: (1) in the range from 1800 to 2500°C the number of particles that fail due to "pressure vessel" failure mechanism increases with increasing temperature; above 1800°C corrosion of SiC begins to occur and at 2000°C decomposition of SiC begins to occur; (2) at 1800°C, there is high release fraction for Cs and at 2500°C there is nearly total release; and (3) at 1800°C the release of Kr (or I) from single "pressure vessel" particle failures increases because of additional particle failures; and (3) at 2500°C, the diffusion of Kr (or I) occurs through decomposed/destroyed SiC layer and still intact PyC layers up to 10%. The implications for core heatup simulation experiments up to 1600°C are that, except for ^{110m}Ag, the fission product release is less that $6X10^{-5}$ which is from the heavy metal contamination during manufacturing. For heatup up to 1800°C, single pressure vessel failures and changing of the SiC structure lead to increasing release of Cs, Sr, and Kr/I, in that order.

HIGH-TEMPERATURE REACTOR NUCLEAR GRADE GRAPHITE

Dr. Gerd Haag, Institute for Safety Research and Reactor Technology, Jülich Research Center, discussed the subject of nuclear graphite for the HTR, including graphite research and development and industrial production [19] [20]. The microscopic carbon structure of graphite components may be viewed at the level of the coke particles, the alignment of crystallites within

the coke particles and the arrangement of individual atoms within the crystallites. However, the behavior and material properties of graphite components when exposed to an irradiation environment can be understood only when investigated at the level of the individual atoms within the crystallites (i.e., the lattice structure of the carbon atoms). Neutron irradiation causes individual atoms in graphite to be knocked out of their latices into interstitial positions between the latices. These carbon atom relocations cause the change in dimensions (growth, shrinkage) in graphite components as well as changes in its material properties. A single 1-2 MeV neutron can displace on the order of 20,000 carbon atoms in graphite crystallite. Initially, shrinkage occurs in an irradiation environment but with increased fast fluence expansion will occur. Depending on the isotropy of the graphite the amount of shrinkage in the orthogonal dimensions under fast fluence can be very similar (isotropic) or fairly different (anisotropic).

The feed source of the coke and the component forming techniques have important influences on the properties of the various reactor graphite grades. Cokes can be ordinary pitch cokes or special pitch cokes. Forming of graphite components may be achieved by extrusion or by vibration in molds. Combinations of these factors can affect (a) the graphite density (the higher the density the greater the neutron moderation), (b) the graphite tensile strength, and (c) the degree of anisotropy. Specific grade designations were established and assigned to the reactor-grade graphites that were manufactured for the German reactors. These grades were based on the sources of coke that existed at the time that the graphite R&D for German HTR applications were conducted. Extensive irradiation testing programs were conducted in Germany for these grades to establish their physical properties for use in design analyses. However, the original material sources for these graphites (i.e., grades) may no longer exist.

Dr. Haag provided a number of observations related to nuclear grade graphite: (1) nuclear grade graphite for permanent core components must be nearly isotropic - but not isostatically molded, (2) special coke processing and careful vibrational molding yields the best graphite grades with respect to isotropy, strength, and homogeneity, (3) the expected lifetime of graphite components has to be based on stress analysis using reliable irradiation data for material and physical properties, and (4) today none of the formerly widely-tested graphite grades are still available.

In view of these observations, Dr. Haag provided a number of recommendations related to nuclear grade graphite: (1) graphite for the PBMR reflector components should be produced from material sources on a "best guess" basis using still existing procedures and experience, (2) data for stress analyses (e.g., irradiation induced growth strains and stresses, coefficient of thermal expansion for calculating thermal strains and stresses) should be deduced from the properties developed for similar materials that were previously tested extensively in the German irradiation programs, (3) an international database for graphite should be established and should be composed of data from the US, UK, Japan, Germany and France, and should be supported by possible users, and (4) for future HTR projects, development and irradiation testing of new graphites should resume as soon as possible.

These observations and recommendations are based on the fact that the mechanisms of irradiation and crystallite changes and the relationships between crystallite changes and bulk dimensional changes have not been developed to the point where dimensional and volumetric changes in reactor graphites can be predicted accurately from pre-irradiation properties or structural features.

PEBBLE BED REACTOR CORE HEAT TRANSFER AND FLUID FLOW

Dr. Scherer, Jülich Research Center, made a presentation on the heat transfer, fluid flow, and power feedback modeling techniques used for pebble bed reactors [21]. During normal operation, all three modes of heat transfer (i.e., conduction, convention, and radiation) are important for modeling and predicting the pebble-bed core temperature distribution. For very fast transients, conduction in and between the coated particles is the most important heat transfer mechanism. The conductivity of the pebbles depends on temperature and fast neutron fluence. During normal operation, the temperature difference across the pebble is less than 70°C and the difference between the helium coolant and the pebble surface is less than 30°C. These temperature differences are valid for low-power (modular) pebble-bed reactors operating at 3 MWt/m^3 power density.

The heat transfer from the fuel pebbles to the coolant is modeled using Nusselt's law with input from experiments. The heat transfer from pebble to pebble by conduction and radiation is modeled using an effective conductivity. The effective conductivity is used in modeling conduction through the pebbles and from pebble to pebble and assumes that the fuel has already been irradiated. The effective conductivity is determined from theoretical principles and the calculated value has been verified to be in close agreement with experimental results. Under conditions of depressurized loss of forced cooling (commonly referred to as a "conduction cool-down" event), the effective conductivity is a dominant factor that limits the maximum fuel temperatures.

Coolant fluid flow in a pebble bed reactor core is difficult to model; therefore, a homogeneous two-dimensional flow model is used. For steady-state conditions, quasi-steady-state flow is assumed.For accident conditions involving low pressure, convective heat transfer is ignored (due to the very low density of helium) and only conduction and radiation heat transfer mechanisms are modeled. A statistical determination of the pebble packing arrangement is used, called a "filling factor." The statistically determined filling factor was verified through experiments.

During normal operation, forced flow in the HTR core is maintained by a blower. For modeling purposes, the pressure drop correlations across the core is obtained from experiments and incorporated in the code. Following a loss of forced flow at high pressure, natural convection will initiate (because helium density is not insignificant). This will cause the core axial temperature distribution to shift upwards so that the upper part of the core is at the highest temperatures. It was mentioned that the analysis of this loss-of-forced-cooling event needs to consider the temperature shift and to determine if the materials in the upper elevations can accommodate the higher temperatures.

The power in a pebble-bed reactor is tightly coupled to helium mass flow rate, mainly because the Doppler effect provides a strong negative feedback via the fuel temperature. Therefore, the helium mass flow rate is used as a means of controlling reactor power. Following a loss of coolant accident depressurization, this same characteristic will shut down the reactor with low-power recriticality occurring only after the decay of xenon. Similarly, a pressurized loss of forced cooling initiates an earlier recriticality due to the initiation of core cooling by natural circulation.

AVR OPERATING EXPERIENCE, TESTING, AND LESSONS LEARNED

Mr. Peter Pohl and Dr. C. Marnet discussed the experiences gained on the AVR pebble-bed reactor [22]. The 21-year operation of the AVR provided a very large source of experiences and test data. The AVR design involved a double reactor pressure vessel made of steel and operated at average helium outlet temperatures up to 950°C. The reactor served as a large-scale test facility for all development stages of pebble fuel elements. The AVR fuel cycling system needed frequent maintenance in the early years but worked well after a series of improvements.

Among the most significant events at AVR was a leak in the steam generator. The AVR's steam generator was located inside the reactor vessel, above the core. In 1978, one of the tubes developed a leak and required isolation. Water had to be removed from the core areas and the pebble refueling piping below the bottom of the reactor vessel. There was, however, no significant damage to the fuel pebbles and none of them had to be removed from the reactor.

During the final several years of operation, tests were conducted at AVR to help demonstrate key safety principles of the HTR-Modul and similar passive modular designs. Experiments simultaneously simulating loss of forced cooling and stuck absorber rods demonstrated passive shutdown without rod insertion. Recriticality occurred after 1 day and stabilized at a very low core power. The response to a complete loss-of-coolant accident without scram was also simulated in an experiment with the AVR running at depressurized conditions and at low power to simulate decay heat. During AVR decommissioning activities, it was found that several fuel pebbles had fallen into and lodged in the helium outlet flow slots in the graphite lower core support structure due to widening of the slots during plant operations. Documents further describing the AVR operating experience and testing program results were identified and will be provided to the NRC staff.

It was reported during the discussions that FZJ is now preparing a report about AVR test HTA-8, which indicated unpredicted local hot spots in the AVR core. In that test, approximately 20% of the 200 unfueled melt-wire pebbles that were passed through the AVR core showed higher-than-expected maximum coolant temperatures (i.e., >1280°C during normal reactor operations with a nominal average outlet temperature of 950°C). The report is expected to provide insights into the implications of these AVR test results with regard to: (a) validating or correcting the code-predicted maximum fuel operating temperatures in a pebble bed reactor design and (b) assessing the need for similar tests and measurements for future pebble bed reactors. It was mentioned that once the report is completed by FZJ and approved by ESKOM, it will be provided to the NRC staff.

THTR OPERATING EXPERIENCE, TESTING, AND LESSONS LEARNED

The 300 MWe Thorium High Temperature Reactor (THTR) was designed during the late 1960s and early 1970s as a demonstration plant toward the planned commercialization of large-scale pebble-bed HTRs in Germany. The long time span between the start of THTR construction in 1972 and initial power operation in 1986 was necessitated largely by design and analysis changes for addressing the evolving regulatory requirements related to external events. Meanwhile, in the early 1980s, development efforts in Germany started a gradual shift away from large-scale HTRs toward more inherently safe modular designs with lower power density, like the HTR-Modul design of the late 1980s. This shift parallels the shift in HTGR development

in the United States, from the Fort St. Vrain reactor (and larger HTGR designs such as the Fulton plant) of the 1970s to the lower power modular HTGR designs of the mid-1980s to early 1990s, leading up to the GT-MHR design development program.

Major technical differences between THTR and today's modular HTR designs include: (a) THTR's prestressed concrete reactor vessel (PCRV) versus the steel reactor vessel needed in the modular designs to accommodate passive heat removal through the vessel wall during accidents, (b) THTR's higher power densities and lower helium temperatures, (c) THTR's use of steam generators instead of the helium turbine power conversion systems used by the latest modular designs, (d) THTR's larger core diameter, (e) a core height-to-diameter ratio of approximately 1:1 for THTR versus approximately 3:1 for modular HTGR designs with reflector-only control and passive heat decay removal through the vessel walls, (f) THTR's use of HEU/Th BISO fuel instead of LEU TRISO fuel, and (g) THTR's use of robust control rods that were mechanically forced into the pebble bed core versus the use of in-reflector control rods and shutdown mechanisms in current modular designs. Despite these differences, THTR operating, testing, and regulatory experiences have yielded relevant technical information and lessons worth considering for modular HTR designs.

From the presentations and discussions by Dr. Josef Schöning of Westinghouse Reaktor GmbH [10], Mr. Guenther Dietrich of Hochtemperatur-Kernkraftwerk GmbH (HKG) [23], and Dr. Helga Kalinowski [24], formerly of HKG and now with the Federal Office for Radiation Protection (BfS), the following THTR "teething" experiences are highlighted:

(a) The frequent breakage of fuel pebbles in THTR resulted in no measurable increases in reactor coolant activity, thus confirming that pebble breakage does not result in significant damage to the embedded coated fuel particles.

(b) The high incidence of broken pebbles in THTR was caused largely by the forceful insertion of control rods into the pebble bed core and was reduced by adding small amounts of ammonia as lubricant. Broken pebbles exacerbated the occurrence of pebble bridging at the core outlet and many became jammed in the fuel handling system. Very little pebble breakage is expected in modular HTR designs due to the absence of in-core control rods

(c) Observed core-bypass helium flows in THTR were nearly three times the predicted design values; the predicted and observed bypass flows were 7% and 18% of the total helium flow, respectively.

(d) Fuel pebbles passed significantly faster through the THTR central core region and significantly slower through the peripheral core region than had been predicted based on pebble flow experiments in air.

(e) Temperature gradients at the core exit were significantly larger than had been predicted, due in large part to the incorrectly predicted pebble flow and the resulting pebble burnup and power profiles. These gradients led to larger than expected thermal stresses in the hot gas ducts and breakage of some insulation attachment bolts due to overstress.

(f) Graphite dust was a greater problem than had been expected and an enhanced filtering arrangement was established for removing the dust. One event involving graphite dust removal resulted in a radiological release off-site within regulatory limits.

(g) After final shutdown, recriticality became a concern during defueling of the THTR core due to the potential for more reactive fuel from the upper part of the outer core region to fall inward toward the center of the core, much like sand falls in an hourglass. This was resolved by adding absorber pebbles during the defueling process.

Despite the operating problems which occurred over the few years of plant operation, overall operational performance for the THTR demonstration plant was viewed as a success within the German nuclear power community. However, faced with political efforts seeking to shutdown the AVR and the higher estimated operating costs and potential financial risks that had been identified, the parties supporting THTR were not willing to continue to operate the plant. As a result, the reactor was permanently shutdown in 1989 only 4 years after licensing. A decommissioning program has been initiated at the facility.

Documents further describing THTR experiences and lessons learned were identified and will be acquired by the NRC staff.

THTR CORE PHYSICS AND PEBBLE FLOW

Dr. Helga Kalinowski, currently of Federal Office for Radiation Protection (BfS) and formerly of HKG Hamm-Uentrop, made a presentation on the pebble flow and physical properties of the THTR core [24]. The actual core physics and core physics models were not discussed during this presentation.

Pebble flow through the core was difficult to model and the actual behavior of the pebble flow was significantly different than predicted from pebble flow experiments in air. The initial core loading pattern produced a temperature profile with a much higher temperature in the center of the core then at the edge. This temperature difference caused the fuel pebbles in the center of the core to move downward much faster in relation to the fuel pebbles at the outer edges than had been predicted by the experiments. Therefore, the solution was to load more fresh fuel in the peripheral core region than in the center in a ratio of 12 pebbles to the outer core for every 3 in the inner core. The pebble flow is a function of local temperatures. Increased temperature lowers the coefficient of friction between the sliding pebbles allowing the pebbles to flow downward more easily. The resultant pebble flow velocity profile across the core resembles the flow velocity pattern of sand flowing down through an hourglass. The pebbles at the outer edge of the core move more slowly and achieve greater burnup by the time they reach the core bottom. This results in the coolant temperatures at the outer edges being lower due to the lower power production. This in turn results in relatively higher friction between pebbles, further slowing the pebble movement. This temperature effect was not seen in the scale model tests which were conducted in air at uniform temperature.

Achieving the optimal pebble flow and loading pattern for the reactor took considerable effort and needed to be continuously monitored. The core diameter-to-height ratio of 1:1 of the THTR was found to promote the increased velocity in the central core region. The ratio was changed to 1:3 (long slender core) for the later designs to achieve passive decay heat removal characteristics and to allow control and shutdown using reflector control elements only. This

change is also expected to improve pebble flow so that the flow across the core is closer to the model predictions. An additional reason why the THTR core did not follow the predicted behavior is because all the experiments used to develop the predicted behavior used air, which results in a pebble flow friction coefficient significantly lower than that in helium. These differences had a significant impact and rendered the tests unreliable for predicting the actual core pebble flow and the resulting neutronic behavior.

The optimal (i.e., desired) temperature profile for the THTR was a flat temperature distribution across the core exit. A flat temperature is optimal for the gas entering the hot gas duct to the steam generators because it reduces thermal stresses in the ceramic and metallic materials that might otherwise be caused by large temperature gradients.

For pebble refueling management to achieve a flat temperature profile, several principles were used to calculate the optimum pebble reload pattern. First, the THTR fuel management process employed pebble conservation. For every pebble that was discharged to spent fuel storage, a fresh pebble was added. Full-power days were used as a measure of the burnup of the core and it was discovered that a correlation existed between the number of full-power days and the number of pebbles that needed to be replaced in the core. To maintain the reactivity of the core, additional fresh fuel needed to be added on a daily basis.

Six refueling parameters were used to determine the optimum pebble reload pattern: pebble conservation, fuel ratio (inner core to outer core), absorber pebble ratio (inner core to outer core), configuring the temperature of the core with the previously burned fuel, allocating more previously burned fuel to the inner core, and allocating the previously used absorber pebbles to the inner core. Depending on the state of the core, not all six of the refueling parameters were strictly maintained, but they proved to be useful starting points when evaluating the core refueling requirements.

Because the behavior of the pebble bed core did not follow predictions, the physical properties of the core had to be periodically confirmed. The physical properties that must be reviewed include: temperatures at the core bottom, control rod worth (differential and total), reactivity (in rod worth equivalence), control rod insertion time, and discharged pebble distribution.

THTR SAFETY ASSESSMENT

Dr. Knud Hofmann, Head of the Energy and Environmental Division, TÜV-Essen, discussed the safety assessment of the THTR [25]. When construction began on the THTR in 1971, technical rules and guidelines for the THTR-specific reactor concept were not in place. The German Federal Ministry of the Interior (BMI) established safety criteria in 1977, but these criteria did not consider the specific characteristics of HTRs. In 1978, a reactor-specific interpretation of these criteria was established with the agreement of the Ministry for the Economy, Trade and Technology of the State of North-Rhine Westphalia (MWMT). In 1980, safety criteria for HTRs were developed by TÜV-Essen under contract to the BMI. These criteria went into effect during the construction of THTR and provided new and more detailed requirements relating to external impact, internal impact and radiation protection requirements. This resulted in significant modifications to the plant design which led to lengthy construction delays.

During operation of the THTR, several operational and design problems were observed, but these issues were not considered to be of high safety significance by operations, design, or

regulatory organizations. These included breakage of fuel elements caused by the insertion of the in-core control rods, failure of bolts in the thermal insulation of the hot-gas ducts due to an elevated temperature gradient at the core exit, difficulties with the fuel handling system that initially limiting refueling activities to less than 40% power, and larger than anticipated quantities of graphite dust in the primary system. Despite these operational and design problems, the THTR demonstration plant was considered a technical success and was viewed as generally providing confirmation of the safety and the feasibility of an HTR based on the pebble bed reactor core concept.

THTR LICENSING SAFETY ISSUES

Mr. Wilfried Hohmann, Ministry for the Economy, Trade and Technology of the State of North Rhine Westphalia (MWMT), discussed safety issues during the licensing of the THTR from the perspective of the state regulatory and licensing authority [26]. The MWMT was the authority responsible for licensing the THTR, and Mr. Hohmann oversaw the licensing process for THTR.

An overview of the THTR design and a chronology of the licensing process and operating life of THTR were provided. The circumstances surrounding the premature shutdown of THTR were discussed. Following the Chernobyl accident, there was political pressure to shut down the THTR because of negative public perception of graphite reactors. This reduced government funding in support of the facility. As a result, the reactor was permanently shutdown in 1989 after only 4 years of operation. A program to decommission the THTR has commenced.

From Mr. Hohmann's perspective, the following are the lessons to be learned from the THTR experience: (1) In-core control rods are "forbidden" in future reactors; (2) There is a need for a strong confinement structure to protect against external impacts; and (3) The behavior of HTRs is dynamically slow and this should be considered in technical regulations. In response to a question as to HTR safety compared to LWR safety, Mr. Hohmann stated that the HTR has potential safety advantages as compared to existing LWR design and technology.

INTERMEDIATE STORAGE FACILITY TOUR

The delegation was taken on a tour of an intermediate fuel storage facility located on the FZJ site. The intermediate storage facility accepts spent fuel and low level waste (LLW) from the decommissioning activities (i.e., to a SAFESTOR level) at AVR. The LLW waste is packaged in drums. Under a SAFESTOR approach, most large components remain at the reactor facility. However, larger pieces removed during decommissioning are sectioned as necessary to fit into drums. The majority of the time involved visiting the hot cell area for spent nuclear fuel handling. Spent fuel pebbles are received in various containers from AVR and its storage areas. The pebbles are repackaged into thin-walled, stainless steel canisters, by gravity or pneumatic methods. Each canister holds about 950 pebbles, and has a small free space. The canisters are closed by a plug inserted into the recessed top. Elastomeric o-rings provide the sealing. The void space consists of air at atmospheric pressure – no helium backfilling or pressurization is performed. A filled canister has a radiation field around 100 R/hr. Several canisters were visible through the hot cell windows.

CASTOR-type storage casks are used. Several casks were being delivered during the visit. Remote operations place two canisters – one on top of the other – inside each cask. An end closure with two metallic O-ring seals is then inserted. After bolting, the operators pressurize

the space between the seals with helium, typically to 5–7 bar of pressure (1.01325 bar = 14.72 psia). Sensors continuously monitor the helium pressure between the seals and alarm on low pressure (i.e., as indicative of a leak; typically at a pressure of 3 bars). Filled casks are vertically oriented in an array that provides adequate spacing for air cooling. The NRC delegation viewed the cask storage area. This consisted of a vault-like building with reinforced concrete walls (nominally 1.3 m thick) surrounding the cask array. Approximately 120 casks were visible containing spent pebbles from the AVR. The IAEA maintains cameras at various locations for safeguards purposes.

AVR SITE TOUR

The delegation was driven to the AVR site. The AVR reactor building is a relatively tall structure for its power level. In an adjacent office area, the delegation viewed mockups of the AVR and graphite blocks and discussed some specific aspects of AVR operations and decommissioning activities. One presenter demonstrated the toughness of graphite pebbles by bouncing one on a hard concrete floor without causing any damage to the pebble. AVR rooms and cells have relatively limited access and are small, but there are many penetrations through the vessel and containment shells. This requires a considerable amount of effort for sealing penetrations as part of the SAFESTOR operations. In particular, the steam generator consists of multiple, independent tube passes and is located within the pressure vessel, above the core. One of the tubes developed a leak in 1978 and required isolation, and water had to be removed from the core areas. Inspection of the AVR internals necessitated boring through the steel shells and inserting a camera. Significantly, the spacings in the bottom gas distributor had widened slightly during operations and this had allowed a small number of pebble fuel elements to fall into the lower gas inlet areas. These fuel pebbles were found during decommissioning and cannot be retrieved until major dismantling commences (i.e., in the future, after the SAFESTOR period). Graphite dust was noted as a concern for both the operational and decommissioning phases, and contributed significantly to operator doses during maintenance activities. The AVR personnel recommended the use of HEPA filtration and appropriate respiratory protection wherever maintenance activities might be performed. Online coolant filtration appeared to be limited to that needed to protect the molecular sieves in the gas purification circuit.

EXPERIMENTAL FACILITY TOUR

The delegation visited FZJ's Experimental Hall No. IV where a number of tests and experiments have been performed on various HTR safety-related structures, systems, and components. The experimental facilities in Hall No. IV at the time of the visit were for HTR passive core cooling phenomena and graphite-air corrosion reactions under simulated accident conditions in modular pebble-bed reactors.

The SANA test facility uses electrical heaters and an ordered packing of pebbles to investigate passive core cooling effects (i.e., conductive, radiative, and convective heat transfer) under pressurized and depressurized accident conditions. Maximum test temperatures up to 1200°C have been achieved. Both graphite pebbles and stone pebbles in air are used to dimensionally model a range of heat transfer relationships of helium/graphite. The SANA test results have been used to validate the analytical models and methods that are used to calculate fuel temperatures in modular HTRs during pressurized and depressurized loss-of-forced-cooling accidents.

A large test apparatus called NACOK (Natural Convection in Core with Corrosion) models a 7-meter high vertical cross-section of an HTR-Modul core with graphite pebbles, electric heaters, and piping arrangements to simulate the reactor vessel and bottom cross gas ducts. Both natural circulation and air ingress (corrosion) tests have been conducted. Maximum temperatures of 1200°C are achievable [27]. From the NACOK experiments, it was found that after a depressurization accident caused by a postulated break in the helium cross duct near the bottom of an HTR-Modul reactor vessel, the "diving bell" geometry will initially limit the rate of diffusion mixing of outside air and hot helium in the core. Specifically, the scaled NACOK test results were reported to indicate an 80-hour "grace period" (i.e., time delay) before the onset of natural convection flow of air through the HTR-Modul core. Convection occurs when the (very low density) helium gas in the vertical "core" region is eventually displaced (via air diffusion) by the relatively high density air from the outside. Air entry in the core initiates sufficient driving force to establish natural convective flows through the system. In the worst case, the integrated analysis of an HTR-Modul, a helium primary circuit and an isolated 50,000 cubic meter confinement (i.e., containing air) would result on about 1600 kg of carbon corroded out of the total of 500,000 kg of carbon in the HTR-Modul design. Note that this implies all of the oxygen in the air reacts with carbon, without any equilibrium limitation. The delegation requested the technical reports on the NACOK experiments conducted to date.

At the time of the visit, developmental testing had been ongoing on various coatings of graphite pebbles [28]. The principal coating investigated was silicon carbide. Tests showed uncoated graphite pebbles would corrode rapidly in air at elevated temperatures, and kinetic expressions were developed. Several silicon carbide coatings and methods were being investigated with the goal of having essentially no corrosion in air up to the maximum allowable accident core temperature (i.e., 1600°C). Extensive measurements have been performed in Germany on heated beds of uncoated graphite pebbles in flowing air. For example, results with pebbles at 900°C indicate graphite corrosion rates of approximately 200 milligrams of reacted O_2 per cm^2 per hour with air flowing at 0.046 meters per second. The reported corrosion rates cover a range of air flow velocities and graphite temperatures from 600°C to 1200°C.

HIGH-TEMPERATURE REACTOR CODES AND STANDARDS

Dr. Ivar Kalinowski, Managing Director of the Secretariat of Nuclear Safety Standards Commission (KTA), provided an overview of activities in Germany related to KTA safety standards for gas reactor technology [29]. The KTA is comprised of 50 members including authorities, experts, utilities, and manufacturers.

Dr. Kalinowski explained the hierarchy of German nuclear safety regulations:

Laws and ordinances – obligatory
BMU guidelines – partially obligatory
Technical rules such as the KTA standards – obligatory and concept-specific.

Dr. Kalinowski provided the delegation with a complete list of the KTA standards including the HTR safety standards which were established by the KTA subcommittee for HTR standards. The HTR safety standards include standards for metallic HTR components, standards for monitoring radioactivity in HTRs and standards for reactor core design for HTRs including calculation of the material properties of helium, heat transfer in spherical fuel elements, loss of

pressure through friction in pebble bed cores, thermal-hydraulic analytical models for stationary and quasi-stationary conditions in pebble bed cores, and systematic and statistical errors in the thermal-hydraulic core design of the pebble bed reactor.

Also, the delegation was presented with the most up-to-date set of the KTA standards for HTRs [30]. These standards were utilized for the regulatory safety review of the HTR-Modul as a source for identifying potential additional HTR concept-specific safety requirements to supplement the existing LWR safety requirements. It is similarly expected that the KTA standards will provide a useful resource to the staff in establishing regulatory design criteria for modular HTGRs such as the PBMR and GT-MHR designs. However, it should be noted that the KTA subcommittee for HTR standards is not active and the KTA standards for metallic HTR components were never issued in final form. The other HTR safety standards were issued in final form but have not been updated or re-affirmed in the last 10 years. Dr. Kalinowski expressed the hope that work on HTR standards development could be resumed with the support and participation of international user organizations.

SPENT NUCLEAR FUEL AND ASPECTS OF WASTE MANAGEMENT

Dr. Kurt Kugeler provided a short presentation on HTR radioactive waste management aspects that complemented the visit to the storage facility the day before [31].

The irradiation time for fuel pebbles in the reactor averages approximately 3 years. Germany's plans for spent HTR pebbles (from AVR and THTR, and recommended for any future HTRs) consists of two phases:

Intermediate storage: this would be for 50–100 years after discharge from the reactor. During intermediate storage, the storage approach would be designed and operated to maintain pebble temperatures below 100°C.

Conditioning for final storage/disposal: This would be designed to keep the pebble temperature below 50°C in final storage/disposal.

Curves were presented showing the decay heat versus time curves for HTR-Modul and other HTR fuels. For the HTR-Modul design, the approximate values are: years after discharge (watts/pebble): 1 (0.4), 2 (0.2), 5 (0.08), 10 (0.05). The intermediate storage approach uses a can in cask method, with remote operations in cells.

The canister/cask system accommodates heat loads of up to 800 watts. For 1900 fuel pebbles at 1 year after discharge, the heat load was stated as about 760 watts. Most of the loaded casks contain fuel over 10 years old, and, thus, typical decay heats are around 60 watts per cask. Pebble fuel temperatures were stated to be under 200°C at the beginning of storage and would be below the 100°C target temperature sometime during intermediate storage; actual temperature decay curves were not presented. The accident analysis did not identify any events resulting in "non-allowable" releases of fission products. A paper on the cask approach was provided.

The presentation also discussed final storage (disposal) options. FZJ has investigated using interstitial steel balls within the pebbles and silicon carbide filling as methods for increasing the conductivity and performance of waste disposal packages. Samples were passed around. Box,

drum, and pressure-resistant disposal packages have been investigated and have been analytically shown to meet dose criteria. Analytical curves also compared the doses from disposal of the graphite fuel pebble with the same quantities of radionuclides in glass; the fuel pebble doses were lower. Some test data indicated a cesium leach rate of 100 Bq/day from a fuel pebble immersed in simulated groundwater. Curves were shown comparing fuel pebble toxicity to the uranium ore. These implied a time period of around 100,000 years before the HTR fuel toxicity equaled that of the natural ore. No specifics were given. Additional toxicity/time curves were presented for partition and transmutation. These displayed a reduction of the time period to around 1000 years for comparable toxicity to the uranium ore. FZJ acknowledged that additional water immersion, leaching testing, and disposal analyses need to be performed.

From the information presented, the decommissioning program is placing approximately 1900 spent AVR fuel pebbles into two cans, with a total (unshielded) volume of about 0.51 cubic meters [31]. For the THTR, approximately 2100 spent fuel pebbles are placed into one can with an unshielded volume of about 0.61 cubic meters [31]. From this experience, it is estimated that the unshielded packaged volume of spent nuclear fuel elements from reactors similar to the German HTR designs could potentially correspond to roughly an order of magnitude increase over that from light water reactors for the same electrical output.

FZJ has initiated decommissioning of the AVR. Based upon one of the papers, the following are the non-fuel inventories of radionuclides in the AVR system, as of 1992:

Cobalt-60	3.2E15 Bq (8.6E4 Ci)
Strontium-90	4.9E13 (1.3E3)
Cesium-137	2.6E13 (7.0E2)
Carbon-14	1.2E13 (3.2E2)
Tritium	1.5E15 (4.1E4)

Note that carbon-14 is the principal long-lived isotope. The AVR non-fuel graphite amounts to approximately 500 tonnes. No estimates for the quantities of graphite involved or anticipated in other HTR designs, such as the HTR-Modul or the PBMR, were presented. However, due to their higher power levels and larger cores, the quantity of graphite is likely to be more than 500 tonnes. It was also indicated that the German program will most likely dispose of the graphite material in a subsurface disposal unit. AVR decommissioning operations will have to address the small number of pebble fuel elements that fell into and lodged flow slots in the graphite lower core support structure. Decommissioning will also have to address potential contamination from the graphite dust via adequate confinement during dismantling.

TRANSFER OF KNOW-HOW FROM GERMANY TO ESKOM

Josef Schöning, General Manager, HTR-GmbH, Heiko Barnert, Jülich Research Center, and Helmut Helmers, TÜV-Nord Hannover, gave presentations on commercial agreements between their respective organizations and ESKOM in the Republic of South Africa (RSA). These agreements involve the transfer of HTR design and technology "know-how" from Germany to ESKOM.

In 1996, a German working group and HTR GmbH signed a memorandum of understanding (MOU) with ESKOM documenting the intent of the German organizations to support ESKOM's

development of the Pebble Bed Modular Reactor and to provide ESKOM access to German HTR know-how. Later in 1996, HTR GmbH entered into a license agreement with ESKOM to provide ESKOM with the complete safety analysis report that has been prepared for the HTR-Modul and to provide ESKOM technical support for the PBMR feasibility study. In 1999, HTR GmbH entered into another license agreement with ESKOM to provide ESKOM with access to HTR technology documents including fuel technology documents filed in the HTR GmbH archives. The agreement also provided for technical assistance and specific consulting work to ESKOM.

In 2000, the Jülich Research Center entered into a license agreement with ESKOM [32]. This agreement gave ESKOM access to all HTR technical documents at the Jülich Research Center involving experimental work that supported the design and development of the HTR (e.g., plant concept, fuel development and behavior, AVR operational experience and test results, reactor ceramic materials high temperature materials technology, HTR component tests, pebble fuel proof tests, nuclear waste management).

In early 2001, TÜV-Nord Hannover entered into a contract to provide ESKOM to conduct an independent review of the safety evaluation prepared by ESKOM for the PBMR in support of PBMR licensing in the RSA. Most recently, in June 2001, HTR GmbH entered into a license agreement with PBMR, Pty, the consortium of companies with an ownership stake in the PBMR, to provide HTR-Module equipment layout, design and construction drawings, and design calculations for HTR-Modul components and systems. The agreement also provided for HTR GmbH to provide technical assistance on specific issues such as graphite dust, solid fission product plate-out, and helium technology issues (e.g., bearings, seals, coatings). NUKEM also has a contract with ESKOM to support the design and construction of the planned Pelindaba fuel fabrication facility in the RSA.

It was mentioned that the agreements with ESKOM are such that ESKOM is not allowed to give the information that is provided to third parties such as PBMR Pty. (or any members) or the NRC. During the visit, the NRC delegation occasionally requested copies of reference information that was included in these agreements. Generally, this information was not provided to the delegation. It was noted that the Technical University of Aachen, which has significant R&D experience with HTR technology, is free of any such agreements.

KEY FINDINGS

The NRC delegation considers the technical information obtained during the visit to be an important step in the development of NRC staff expertise and capabilities with the goal of conducting effective and efficient safety reviews of HTGRs such as the PBMR and GT-MHR. The delegation therefore strongly encourages the technical staff to read in full the technical documents that were obtained in their respective areas of technical or professional interest to maximize HTGR technology transfer effectiveness. At the present time, the NRC delegation has not performed a detailed technical review of the material, and, thus, all findings should be considered as preliminary. The following technical information is viewed by the delegation as important to the safety or operational assessment of modular HTGRs:

1. The manufacture of high quality fuel that consistently achieves fuel performance within expectations during irradiation and accident testing requires meticulous adherence to proven manufacturing equipment, processes and procedures and precise adherence to

established quality measures for all aspects of fuel manufacture. Exact compliance in these areas is essential. Based on Germany's 25 years of experience with fuel development and manufacture, a maximum initial defect fraction of 6×10^{-5} has been specified.

2. German fuel manufacturing for the HTR-Modul design would use 7–9% enriched uranium. More chemicals and flammable materials and more operating processes are used in producing coated fuel particles and fuel pebbles than in producing pellet-type fuels for light-water reactors. New sources of graphitic materials may need to be identified and qualified. The higher enrichment level and the small size of the fuel components (particles and pebbles) may require additional considerations for material control and accounting (MC&A) and safeguards. Fuel pebbles in the German HTR program did not have unique identifiers or labels for quality control and tracking purposes.

3. The Natural Convection in Core with Corrosion (NACOK) experiments were conducted at Jülich to assess air ingress into an HTR-Modul reactor for a postulated break in the lower hot gas duct. From the experiments it was found that after a depressurization accident caused by a postulated break in the helium cross duct near the bottom of an HTR-Modul pressure vessel, the "diving bell" geometry will initially limit diffusion mixing of outside air with hot helium in the system. For cases with no additional breaks postulated in the pressure boundary above the cross duct, the diving bell geometry was found to provide a "grace period" (i.e., time delay) before the onset of natural convection flow of air through the core. After the grace period, natural circulation of air through the core begins, subjecting the graphitic materials of the core supports, pebble fuel elements, and reflector blocks to oxidation-induced corrosion.

4. Oxidation measurements have been performed in Germany on heated beds of uncoated graphite pebbles in flowing air. For example, results with pebbles at 900°C indicate graphite corrosion rates of approximately 200 milligrams of reacted O_2 per square centimeter per hour with air flowing at 0.046 meters per second. The reported corrosion rates cover a range of air flow velocities and graphite temperatures from 600°C to 1200°C, with more limited data up to 1600°C. Research has shown that graphite corrosion can be prevented or greatly reduced by coating the graphite surfaces with silicon carbide. Work on developing the silicon carbide coating processes and on irradiation and durability testing of coated pebbles and other coated graphite structures is ongoing. The use of coatings on graphitic surfaces was not included the HTR-Modul design concept and does not appear to be included in the PBMR and GT-MHR design concepts at this time.

5. Specific grade designations were established and assigned to the reactor graphites that were formerly manufactured for the German reactors. These grades were derived for the specific feed sources of coke that existed at the time that the graphite R&D was conducted for the German HTGR applications. Extensive irradiation testing programs had been conducted in Germany for these grades to establish their material and physical properties for use in reactor design and analyses. However, today none of the formerly widely tested graphite grades is available. For future HTGR projects, development and irradiation testing of new graphites will be required.

6. Pebble flow through the THTR core was significantly different than had been predicted from the scale model tests, which had been conducted in air at uniform temperature. The initial core loading pattern produced a temperature profile with much higher temperatures at the core centerline than at the periphery. The pebbles near the peripheral core region moved much more slowly than predicted. By the time these outer pebbles reached the bottom of the core, the burnup was greater than predicted. This resulted in lower-than-expected local coolant temperatures due to the lower pebble power from the higher-burnup fuel pebbles. This in turn resulted in relatively higher sliding friction between pebbles, further slowing the pebble movement. The increased temperature gradient at the core exit produced higher thermal stresses in the helium cross duct, which led to the failure of some insulation attachment bolts. The actual behavior of the pebble flow was difficult to model. The THTR pebble flow experience is expected to provide important input to the review of a range of safety and design analyses which are based on pebble flow behavior.

7. Due to the absence of instrumentation within the pebble bed core, a special test was conducted at the AVR with melt-wire pebbles. The test indicated unpredicted local hot spots. In the test, approximately 20% of the 200 "melt-wire" pebbles that passed through the core at full power were found to have experienced maximum coolant temperatures above 1280°C. This was well above what had been predicted. FZJ is now preparing a new evaluation and analysis of the AVR melt-wire test results and the resulting report will be provided to the NRC staff when it is completed. The report is expected to provide insights with regard to: (a) validating or correcting the code-predicted maximum fuel operating temperatures in a pebble bed reactor design, and (b) assessing the need for similar tests and measurements for future pebble bed reactors.

8. The German safety analyses and safety evaluations for HTR design basis events involve a traditional deterministic approach with conservative assumptions. These include such aspects as the assumed failure of the first RPS trip signal, consideration of the worst single failure, and no credit for non-safety related equipment. Code calculations utilize conservative inputs for physical and material properties and initial conditions. Shutdown decay-heat removal and fission-product retention must be shown. Postulated events in each event category are developed based on the design-specific features and equipment.

9. The safety evaluation of the German HTR-Modul design concluded that the safety design functions of passive shutdown, passive decay heat removal, fission product retention within the fuel, and protection of the core against chemical attack (oxidation) would perform as analyzed. With regard to the potential for graphite oxidation events, the safety analysis cited the "leak before break" design criterion in assuming only individual small breaks (65 mm diameter) in the reactor pressure boundary piping. Events resulting in larger amounts of air entering the reactor core were excluded from the design basis because of the estimated low combined probabilities of the component failures involved and the time available for mitigative actions.

10. Testing was performed at Jülich to understand the how various steel surfaces are affected by the lack of oxide film formation in a helium atmosphere. Without proper design and surface preparation, some steel joints and bearing surfaces exposed to helium were found to fuse or self-weld.

11. Significant operating experiences occurred at the THTR. These included: pebble breakage (without measurable increases in reactor coolant activity) due to control rod insertion into the pebble-bed core; core-bypass helium flows nearly three times the predicted design values; pebble flow patterns significantly different than what had been predicted; core exit temperature gradients significantly larger than had been predicted resulting in breakage of a number of insulation attachment bolts; graphite dust problems greater than had been expected and; shortcomings in the online refueling system instrumentation and controls used to monitor pebble flow in the refueling system. Despite these operating problems, overall operational performance for the THTR demonstration plant was viewed as a success within the German nuclear power community. However, faced with political efforts seeking to shutdown the AVR and the higher estimated operating costs and potential financial risks that had been identified, the parties supporting THTR were not willing to continue to operate the plant. As a result, the reactor was permanently shutdown in 1989 only 4 years after licensing. A decommissioning program has been initiated at the facility.

12. Decommissioning of the AVR and THTR is based upon a SAFESTOR approach. Significant quantities of activated and contaminated non-fuel graphite (containing carbon-14 and tritium) will likely require disposal at some time. The AVR had 500 tonnes of non-fuel graphite. In addition, the design and layout of these plants did not appear to fully consider the need to minimize radiological exposure of workers during the decommissioning activities.

13. Spent fuel pebbles from the AVR and THTR are being loaded into metallic casks, similar in concept to those used in the United States. Specific power density and weight of heavy metal are lower; however, the packaged volume of spent fuel elements from HTRs is potentially higher, by roughly and order of magnitude, than that from light-water reactors for the same electrical output. The casks used in Germany do not use helium backfilling and slight oxidation of the graphite pebbles has been observed due to the small amount of oxygen in the cask/can system. The fuel pebbles are not held by fasteners or springs in the casks and are free to move.

14. Several key German organizations with extensive and expert technical knowledge and large archives of technical documents on German HTGR design and technology, have entered into agreements with ESKOM to support ESKOM's design and development of the PBMR and its licensing in the RSA. Most of these agreements are with ESKOM and provide access to extensive research, development, design, testing and operating data, and safety analyses and safety evaluations of German high-temperature pebble bed-reactors. Since these agreements involve direct assistance to ESKOM in support of PBMR licensing, NRC cooperation with the involved organizations in support of PBMR pre-application review in these technical areas would likely raise a conflict of interest for these organizations. Additionally, since these agreements prohibit ESKOM or the other involved receiving organizations from providing the information to third parties, NRC may not be able to obtain the reference technical information through Exelon until such time

that they sign their own separate agreements with the involved German organizations. However, some of the German organizations have indicated that the technical information provided to ESKOM and PBMR could also provided to NRC under a separate agreement.

CONCLUSIONS

The German nuclear power industry believes they have demonstrated that HTGRs can be successfully designed, constructed and licensed, and operated with acceptable safety performance. German safety and regulatory authorities have concluded that the HTR-Modul design (a modular pebble bed HTGR similar to the PBMR) would have been able to meet the safety criteria for licensing in Germany. Actual operating experience of German HTGRs suggests that startup problems with new HTGR plant designs can be expected. German experiments, tests, safety evaluations, licensing experiences, and subsequent plant operating experiences have provided important lessons in HTGR design, technology, safety analysis and regulation. These lessons should be considered in the NRC's HTGR pre-application and licensing review activities. The information obtained from Germany on the safety aspects of HTGR design and technology will be extremely beneficial in supporting the NRC's safety reviews of new HTGR designs.

Appendix A

SAFETY ASSESSMENT OF THE HTR-MODUL

The HTR-Modul is a thermal power plant designed for the cogeneration of electricity and process steam. The plant is comprised of two nuclear steam supply systems (modules) in a common reactor building. Each module consists of one high- temperature reactor in a steel pressure vessel, one steam generator in a separate steel pressure vessel, one primary gas blower joined to the steam generator vessel, and a connecting pressure vessel containing coaxial hot-gas/cold-gas systems which connects the reactor to the steam generator. The capacity of each module is 200 MWt (80 MWe). The HTR-Modul fuel design was based on the standard reference low enriched uranium (LEU) fuel element, which is also the reference for the PBMR fuel design.

The TÜV performed a traditional deterministic assessment of the HTR-Modul design against the basic safety criteria of shutdown, decay heat removal, and retention of fission products. These safety criteria were satisfied in the HTR-Modul design by the following safety features:

- Shutdown: The HTR-Modul design includes two shutdown systems. The automatic reflector control rods for reactor control and hot shutdown and the manual small sphere absorber system (KLAK system) to ensure cold shutdown of the core. The absorber spheres were not considered necessary by the designer, but were required as an independent means of reactor shutdown. Due to the negative temperature coefficient of reactivity, the reactor can also be shutdown by turning off the primary coolant blower thereby interrupting the primary coolant flow. This inherent safety property of the reactor was not credited by the designer in the safety analysis report. The shutdown of the blower and insertion of the reflector rods are initiated simultaneously by the reactor protection system.

- Decay Heat Removal: Decay heat is removed from the core passively by heating up the surrounding structural components. Active heat removal is not necessary to avoid exceeding the fuel design temperature of 1620°C. During normal operating conditions, the energy losses from the reactor pressure vessel will heat up the concrete structures, and the reactor cavity is equipped with a surface cooler to protect these structures. Analyses were performed to demonstrate that there is no need for short-term availability of active decay heat removal. The design temperatures of the reactor cavity concrete structures and reactor components will not be exceeded until 15 hours after shutdown.

- Retention of Fission Products: The HTR-Modul design does not include a pressure-resistant, gas-tight containment. The confinement, consisting of the reactor building and its associated ventilation and filter system, was designed to facilitate activity control. The design concept of the HTR-Modul is such that fission products will be nearly completely contained in the fuel elements provided that the fuel design temperature of 1620°C is not exceeded. In a loss-of-coolant accident, the fission gas activity of the coolant and part of the plate-out activity on the primary system surfaces would be released to the reactor building and to the environment via the ventilation stack. The resulting radiation exposure to the environment was calculated to be far below the accident dose limits of the German Radiological Protection Ordinance.

LWR licensing basis events were screened for applicability by the TÜV and HTR-Modul specific scenarios were added. As a result, the list of licensing basis events for the HTR-Modul was revised and enlarged. The applicant revised the safety analysis report to include the revised listing. The following categories of design basis events were analyzed:

- Reactivity Accidents
- Disturbed Heat Removal Without Loss of Coolant
- Disturbed Heat Removal With Loss of Coolant
- Loss-of-Coolant Event
- External Events (does not include aircraft impact and external shock wave)

The event analysis was also revised by the applicant to address the following basic assumptions: (1) failure of the first initiation signal to activate the reactor protection system; (2) consideration of single failure and system unavailability due to maintenance; and (3) non-safety related systems are not credited. The revision to the safety analyses resulted in an increase of the fuel design temperature to 1620°C from 1600°C, and resulted in design changes to the reflector control rod system, the reactor protection system, and the seismic design of some structure and components.

Aircraft impact and external shock wave were considered extremely low probability events and were not classified as design basis events. "Risk-reducing measures" are provided in the HTR-Modul design to reduce the risk due to operation of the plant. The reactor building (essentially the confinement) and the safety related components in the reactor building were designed for loads from aircraft impact and external shock wave. The switch gear and emergency supply building are assumed to be partially or completely destroyed by the event which could result in failure of the reactor protection system and emergency power supply system. The applicant planned to design the reactor protection system such that the protective actions would be initiated when necessary due to plant behavior or as a result of damage to the reactor protection system itself. In addition to the above described risk-reducing measures, steps were required to establish an external supply of feedwater for the reactor cavity surface coolers and a power supply for the emergency control room.

The safety evaluation of the HTR-Modul design concluded that the safety design functions of passive shutdown, passive decay heat removal, fission product retention, and protection of the core against chemical attack (oxidation) would perform as analyzed. With regard to the potential for graphite oxidation events, the safety analysis cited the "leak before break" design criterion in assuming only isolated small breaks (65 mm diameter) in the reactor pressure boundary piping. Events resulting in larger amounts of air entering the reactor core were excluded from the design basis because of the estimated low combined probabilities of the component failures involved and the time available for mitigative actions.

Appendix B

PEBBLE FUEL ELEMENT RESEARCH, DEVELOPMENT, AND INDUSTRIAL PRODUCTION

Pebble Fuel Design

The basic concept consists of coated particle fuel. The center comprises the fuel, as a kernel, and is surrounded by multiple coatings that protect the fuel and retain the fission products.

The initial pebble fuel designs of HTR fuel in Germany for the THTR utilized BISO coated fuel particles based on the BISO fuel designed and manufactured in the US. This fuel involved pebbles with a central spherical fueled region consisting of coated particles randomly mixed in a graphite matrix surrounded by a fuel-free graphite outer shell. Highly sintered thorium and uranium oxide (10-to-1 thorium-to-uranium) at 93 % enrichment was initially utilized. All layers coating the fuel kernel in the BISO coated particle design involved pyrolytic carbon material.

The later reference fuel design for the HTR-Modul involves a TRISO particle that was used for reloads at the end of the AVR operating history. This fuel is also the reference design for the Pebble Bed Modular Reactor (PBMR). The HTR-Modul reference fuel has the same overall fuel element design as the THTR (i.e., a central 50 mm spherical fueled region consisting of coated particles randomly distributed in a matrix of graphite and binders surrounded by a 5 mm fuel-free graphite outer shell). However the coated fuel particles are of the TRISO particle design. The fuel kernel is highly sintered (near theoretical density) UO_2 with a uranium enrichment of 7-9 %.

For TRISO fuel particles the layers and the purpose of each layer was described as follows:

Inner Buffer Layer: Low density (i.e., ~50% porosity) pyrolytic carbon. The buffer layer provides void space for fission product gases, serves to accommodate the irradiation-induced swelling of the fuel kernel (including fission product recoil) and protects the other layers from damage due to these effects.

Inner Layer: High density pyrolytic carbon deposited from an argon/acetylene/propylene gas mixture. The inner layer retains most of the fission products; fixes the inner porous buffer layer; protects (seals) the next (SiC) layer from chemical attack from fuel kernel fission products; prevents hydrogen chloride, that is generated during the formation of the SiC layer, from entering fuel kernel.

Silicon Carbide (SiC) Layer: The layer is generated from the decomposition of trichloromethyl silane (CH_3SiCl_3) upon the fuel particle, in the presence of hydrogen gas. The SiC layer serves as the impervious barrier to the escape of gaseous or solid fission products (except ^{110m}Ag) from escaping the coated particles; Provides the largest contribution to the mechanical strength of the particle; and functions as a pressure vessel. The silicon carbide layer temperature of formation is important to the effectiveness of the coating (1550°C was mentioned as an optimum).

Outer Layer: High density pyrolytic carbon deposited from an argon/acetylene/propylene gas mixture. The outer layer serves to protect the SiC layer from chemical attack from outside the particle and adds strength to the SiC layer.

Overall the purpose of the coatings is to prevent fission products from escaping the fuel kernel during fuel manufacture, in-reactor irradiation, and potential accidents.

Pebble Fuel Element Manufacture

The fuel element manufacturing process consists of: UO_2 fuel kernel manufacture, coating of the fuel kernels, and manufacture of fuel elements.

The UO_2 fuel kernels, are prepared by a modification of the ammonium diuranate (ADU) process that uses vibrating nozzles to generate the initial spherical droplets. The manufacture of the fuel kernels begins with a uranyl nitrate solution. The solution is pre-neutralized and mixed with polyvinyl alcohol (PVA) and tetrahydrofurfyl alcohol. This forms the feed solution. A pump forces the feed solution through small diameter vibrating nozzles. This is termed vibrodropping. The diameter of each droplet (which determines the size of the fuel kernels) is very precisely controlled and is determined by the nozzle orifice diameter, pressure, and vibrating frequency. The free droplets fall through a small gaseous space and then a more concentrated solution of ammonium hydroxide. This continues the ADU precipitation reactions and the uranium/ADU particle assumes the shape of minimum energy – a sphere – as it falls through the ammonium hydroxide solution.

The ammonium hydroxide solution needs to have adequate height to allow sufficient conversion to ADU so that the sphere is mechanically stable when it reaches the bottom of the column or precipitation chamber. At the bottom of the column, the kernels (also called gel spheres because of their softness) are allowed to "age" and complete the ADU reactions. This forms an ADU kernel of adequate strength for handling. The ADU spheres are removed, washed to remove residual chemicals, and dried at moderate temperatures. A calciner converts the ADU to uranium oxide (UO_{2+x}), and reduction with hydrogen completes the conversion to uranium dioxide. A high temperature sintering step increases the density of the kernel to near theoretical density. The fuel kernels are sorted by sieving to ensure 100% meet the specified size and sphericity. The finished fuel kernels are measured,and classified by size and roundness within the specified tolerance band. The reference German fuel for the AVR design had a sintered fuel kernel mean diameter of 500 μm. The PBMR fuel is based on this reference.

Each kernel is coated into a TRISO particle using a fluidized bed coater qualified for a 5 kg batch (lot) size. Each coating layer is added via a chemical vapor deposition (CVD) processes in a sequential layering process. The CVD process decomposes gaseous species at temperature in a high surface area medium (the kernels, as the fluidizing bed). The kernels act as nucleation sites for the decomposition which grows the various layers. Each coating is made from a mixture of a carrier gas (typically Argon) and a coating gas which depends on the layer involved. The silicon carbide layer is coated using H_2 as the carrier gas and CH_3SiCl_3 as the coating gas. As each layer in turn is added, the particle diameter increases from the 500 μm UO_2 kernel size to the 1000 μm diameter of the finished coated particle. The UO_2 fuel kernels result in limited heavy metal contamination inside the coater and represents the source for heavy metal contamination outside the SiC layer in the finished particles. The Nukem fuel plant had a particle fuel capacity of approximately 2 MTHM/yr.

Finished particles are then characterized. The last step is to provide a 100 μm overcoat of pyrolytic carbon. The overcoat provides a protective layer for the finished particles to prevent damage and breakage during the high-pressure pressing in the graphite matrix in manufacture of the pebbles.

With the standard design, one coater can process five kilograms (U) of fuel batch size and apply all four coatings in 8–10 hours. A larger coater has been tested for processing 10 kg (U) batches in the same 8–10 hour period but has not been licensed for LEU TRISO particle fuel manufacture based upon German State license (criticality) restrictions. This 5 kg coater is to be used for PBMR fuel manufacture. Safety analyses have shown that the 5 kg/batch coater can accept up to 10% assay material. The coaters use argon as the carrier gas for the pyrolytic carbon layers. Temperatures of 1200–1600°C are achieved by electrical heaters in the base and funnel area walls of the coater. Most of the surfaces in the coating system are graphite or graphite lined. The coater also has insulation, cooling water jackets, and thermocouples around the fluidized bed walls.

The finished TRISO particles are mixed with an approximately 50/50 mixture of graphite powder and binder material to form the fueled zone of the pebble fuel element. These are formed in spherical rubber molds, initially in a pre-molding at low pressure. The pressure must be applied isostatically (uniform) to avoid particle failures from nonuniform external pressures. (The fuel particles are not strong when subjected to high non-isostatic external pressure.) The pre-molded fuel elements are then covered in a fuel free zone of graphite power and pressed a second time at high pressure (300 bar). The completed fuel elements are heat treated at up to 1950°C to remove all volatile material and convert the binder/graphite/fuel particle mass into a monolith. This temperature is sufficiently distant from the 2000°C plus at which the SiC layer would begin to decompose into its constituents. After the final molding and heat treatment, the pebbles are machined to the precise diameter and finish. Finished pebbles are then characterized.

NUKEM manufacturing experience of TRISO particle pebble fuel elements for the THTR involved about 1000 batches of kernels, about 4000 batches of coated particles and about 500 lots of finished pebble fuel elements (~1M pebble fuel elements). Overall yields (input uranium to uranium in the final fuel pebbles) were greater than 95 % for these products.

Fuel quality is primarily verified by destructive analyses on selected samples from batches. Experience has developed a set of procedures and processes requiring verbatim compliance for generating the fuel with known quality; typical failure numbers of $1X10^{-4}$ to $1X10^{-5}$ were cited for defective pebbles, with one or two defective particles per pebble. This is generally better than the failure rates found during prior NRC efforts on HTR fuels.

According to Dr. Heit, the key to consistent manufacturing quality and consistency and fuel performance within expectations during irradiation and accident simulations is the proven manufacturing equipment and manufacturing process procedures, and a special and detailed quality assurance program for all aspects of fuel manufacture and fuel produced. The way to reproducing the consistent success that was eventually achieved by the German program in the 1980s must involve a deliberate and meticulous characterization of each aspect of manufacture in the fuel manufacturing development process and fuel products leading up to the proven performance and qualification of the final fuel facility production lines and fuel that will

consistently meet all fuel product specifications. Exact compliance with the final fuel manufacturing procedures is essential. However, Dr. Heit indicated that improvements could be made with fuel manufacturing process.

Dr. Heit also stated that the irradiation fuel proof testing for the production fuel must be fully representative of the production fuel that will be made for the HTR plants. To achieve this consistency, both the production fuel elements and the fuel elements used for the proof tests must be manufactured using TRISO particles which are based on a split statistical sample taken from the same (number of) batches of TRISO particles made by the same fuel manufacturing lines (e.g., fluidized bed coaters).

The design drawings for the manufacturing equipment and the manufacturing process procedures and related documented still exist in Germany, although the manufacturing equipment itself has been sold to the Chinese for the manufacture of the HTR-10 fuel. German organizations also have retained personnel who have knowledge and experience in the manufacture of TRISO fuel particles and pebble fuel elements.

LIST OF HANDOUTS AND DOCUMENTS PROVIDED

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8. Large Test Facilities in HTR Development, Kurt Kugeler, Jülich Research Center

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10. Overview on the HTR Program in Germany, Josef Schöning, Westinghouse Reactor GmbH

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15. Pebble Bed Fuel Element Research and Development and Industry Production in Germany, Heit, Froschauer, NUKEM Nuclear GmbH, Germany

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21. Heat Transfer, Fluid Flow and Power Feedback in Pebble-Bed Reactors, W. Scherer, Jülich Research Center

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23. THTR Operation Experience, Test Programs, Overview, Highlights, Lessons Learned; Guenther Dietrich, HKG, Ivar Kalinowski

24. Core Physics and Pebble Flow, Examples from THTR Operation, Helga Kalinowski, BfS (formerly HKG)

25. THTR 300 Prototype Reactor Safety Assessment; K. Hofmann, W. Tapp

26. High-Temperature-Reactor Technology – Licensing Basis Safety Aspects of the THTR, W. Hohmann, MWMT-NRW (in German, presented with translator)

27. NACOK: Natural Convection in Core with Corrosion, Institute for Safety Research and Reactor Technology (ISR), Jülich Research Center

28. Ceramic Coatings for HTR Graphitic Structures – Tests and Experiments with SiC-Coated Graphitic Specimens, B. Schroeder et al, Jülich Research Center and T.U.-Aachen (article)

29. Summary of KTA 3321, Ivar Kalinowski

30. Collection of draft and final KTA Safety Standards pertaining to gas-cooled high-temperature reactors (most in German, some in English), courtesy of Hubertus Nickel, Jülich Research Center

31. Waste Management – Spent AVR Fuel Elements, Kurt Kugeler, Jülich Research Center

32. Appendix: Know-How on Pebble Bed HTR owned by FZJ being of relevance for the PBMR-Project of ESKOM (FZJ-ISR-RC-5001/2000), Heiko Barnert, Jülich Research Center

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