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

RES
90 R

#____ REC'D 8/17
2 JSM
1 LDS
__ MMD
3 BCM
__ JOT
__ LPP
__ TGF
FILE: Advanced Reactors

August 21, 2001

NOTE TO COMMISSIONER ASSISTANTS

OCM/RM	OCM/GJD	OCM/EM	OCM/JM
X Dennis Rathbun	X Brad Jones	X Janet Schlueter	X Lynne Stauss
__ Mike Tschiltz	__ Cynthia Jones	__ Kathryn Winsberg	__ Margie Doane
__ Alan Levin	__ Tom Hiltz	__ James Beall	__ Brian McCabe
__ Julio Lara	__ Joe Olencz	__ Jeffry Sharkey	__ John Thoma
__ Keith McConnell	__ Jose Ibarra	__ Linda Lewis	__ Heather Astwood
__ Mark Thaggard	__ Donna Smith	__ Judy Ledbetter	__ Lorna Kipfer
__ Bob McOsker	__ Noble Green		__ Tojuana Fortune
__ Marian Zobler			
__ Karen Henderson			
__ Clare Kasputys			
__ Barbara Gabriel			
__ Linda Herr			
__ Terry Agneu			

FROM: John W. Craig
Assistant for Operations, OEDO

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

Staff from RES, NRR, and NMSS participated in technical meeting in Germany on July 23-26, 2001. These meetings focused on safety aspects of high-temperature gas-cooled reactor (HGTR) design and technology. The objective of the visits was 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 first and last days were spent at the GRS offices in Cologne and the other two days were spent at the Julich Research Center, Julich. Discussions were held on operating and test experience with pebble bed HTGRs, reports and documents were exchanged, and insights were gained on a broad range of technical topics. Additional documents were requested and international agreements are being planned to expand NRC's technical understanding of HTGR technology.

Attached for your information, is the full agenda (Attachment 1); a list of the German participants and their affiliations (Attachment 2); a summary of the presentations, discussions and observations (Attachment 3); and a list of handouts and other documents that were provided during the course of the visit (Attachment 4).

Attachments: As stated

cc: W. Travers, EDO (w/o attachments)
C. Paperiello, DEDMRS (w/attachments)
W. Kane, DEDR (w/o attachments)
P. Norry, DEDM (w/attachments)
S. Reiter, CIO (w/o attachments)
J. Craig, AO (w/attachments)
I. Schoenfeld, OEDO (w/attachments)
T. King, RES (w/o attachments)

SECY (w/attachments)
OGC (w/attachments)
OCA (w/attachments)
OPA (w/attachments)
OIP (w/o attachments)
CFO (w/o attachments)
EDO R/F (w/attachments)

FF/22

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

DRAFT

Amy Cubbage

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

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

During the period July 23-26, 2001, a six-member NRC delegation participated in a productive 4-day visit to Germany for technical meetings on safety aspects of 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.

The discussions opened up channels of communication on the design and technology of HTGRs. Discussions were held on operating and test experience with pebble bed HTGRs. Non-propriety 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 AVR spent fuel intermediate storage facility, the hot cells for irradiated fuel examination, and 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, and (12) 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 copy of the full agenda is provided in Attachment 1 and a list of the German participants and their affiliations is provided in Attachment 2. Attachment 3 provides a summary of the presentations, discussions and observations during the four-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.

Attachments:
1. Agenda for the Visit to Germany
2. List of German Participants
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 copy of the full agenda is provided in Attachment 1 and a list of the German participants and their affiliations is provided in Attachment 2. Attachment 3 provides a summary of the presentations, discussions and observations during the four-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.

Attachments: 1. Agenda for the Visit to Germany
2. List of German Participants
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

Distribution w/o atts.: See attached list

G:\REAHFB\RubinS\TRIP REPORT 8_16BREV.wpd

OAR in ADAMS? (Y or N) ______ ADAMS ACCESSION NO.: ______________ TEMPLATE NO. RES-______
Publicly Available? (Y or N) ______ DATE OF RELEASE TO PUBLIC ______________ SENSITIVE? ______

To receive a copy of this document, indicate in the box: "C" = Copy without enclosures "E" = Copy with enclosures "N" = No copy

OFFICE	REAHFB:RES	REAHFB:RES	DSSA:NRR	DSSA:NRR	FCSS:NMSS
NAME	SRubin:mmk	DCarlson	ACubbage	NShoop	AMurray
DATE	/ /01	/ /01	/ /01	/ /01	/ /01
OFFICE	BCA:OIP	C:REAHFB:RES	D:DSARE:RES	DD:RES	D:RES
NAME	HFaulkner	JFlack	TKing	RZimmerman	AThadani
DATE	/ /01	/ /01	/ /01	/ /01	/ /01

Memorandum dated: / /01

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

Distribution w atts.:

TKing, RES	WBorchardt, NRR	ELeeds, NMSS	SBall, ORNL
CAder, RES	WKoo, NRR	SSteele, NMSS	
MMayfield, RES	YOrechwa, NRR	WGleaves, NMSS	
SNewberry, RES	FEltawila, NRR	VPerin, NMSS	
JRosenthal, RES	NTrehan, NRR	MWeber, NMSS	
NChokshi, RES	DJackson, NRR		
DDorman, RES	JStrosnider, NRR		
JMuscara, RES	FGrubelich, NRR		
HGraves, RES	EImbro, NRR		
NLauben, RES	HAshar, NRR		
RMeyer, RES	RCaruso, NRR		
CTinkler, RES	TCheng, NRR		
SArndt, RES	FAkstulewicz, NRR		
JUhle, RES	HLi, NRR		
CGingrich, RES	EConnell, NRR		
REAHFB R/F			
DSARE R/F			

List of German Participants

Name	Affiliation	E-Mail
Barnert, Heiko	Jülich Research Center	h.barnert@fz-juelich.de
Bonigke, Guenther	GRS	bon@grs.de
Brinkmann, Gerd	Framatome-ANP	gerd.brinkman@framatome-anp.de
Dietrich, Guenther	HKG	guenther.dietrich@rwepower.com
Eisenbeiss, Gerd	Jülich Research Center	g.eisenbeiss@fz-juelich.de
Froschauer, Karl	NUKEM	karl.froschauer@nukem.tessag.com
Haag, Gerd	Jülich Research Center	g.haag@fz-juelich.de
Halaszovich, S	Jülich Research Center	st.halaszovich@fz-juelich.de
Heit, Werner	NUKEM	
Helmers, Helmut	TÜV-Nord, Hannover	hhelmers@tuev-nord.de
Hofmann, Knud	Rhein-Westf TÜV, Essen	
Hohmann, Wifried	Ministry of Economy..., D-dorf	wolfried.hohmann@mwmev.nrw.de
Kalinowski, Helga	BfS	hkalinowski@bfs.de
Kalinowski, Ivar	BfS/KTA-GS	ikalinowski@bfs.de
Kersting, Edmund	GRS	kee@grs.de
Kleine-Tebbe, A		
Kugeler, Kurt	Jülich Research Center	k.kugeler@fz-juelich.de
Leder, Walter	GRS	
Marnet, C		
Nabielek, Heinz	Jülich Research Center	h.nabielek@fz-juelich.de
Nickel, Hubertus	Jülich Research Center	h.nickel@fz-juelich.de
Nitzke, Volker	TÜV-Nord Hannover	vnitzke@tuev-nord.de
Pohl, Peter		
Pott, Guenther	Jülich Research Center	g.pott@fz-juelich.de
Scherer, Winfried	Jülich Research Center	w.scherer@fz-juelich.de
Schenk, Werner	Jülich Research Center	w.schenk@fz-juelich.de
Schöning, Josef	Westinghouse Reactor, Mannh	josef.schoening@freenet.de
Schroeder, Bruno	Jülich Research Center	b.schroeder@fz-juelich.de
Storch, S		
Vogel, Gerhard	TÜV-Nord, Hannover	gvogel@tuev-nord.de
Wahlen, Edger		

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 ® & D) 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 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 twenty 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 if the future licensing organization (FLO), in NRR, the interest of Exelon in the 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 the some of the challenging design, technology, safety and policy review issues that these initiatives raised. Finally, he discussed 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 the U.S., especially as it related to HTGRs, including the PBMR.

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

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) 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 ACRS. 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 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). 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].

In recent years, increasing attention has gone to the study of advanced reactor safety features that go beyond current 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 the graphite surface of pebble fuel elements and graphite reflector blocks. 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.

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 twenty years commencing in 1967. The THTR was a demonstration plant that operated for less than four 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 three-year test program, one 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.

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 reactor design similar to the PBMR except that the HTR-Modul design incorporates a steam generator in the power conversion system. This HTR-Modul design was characterized as having a low power density, passive safety features, and a confinement envelope.

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, TÜV-Hannover, discussed the safety evaluation performed by TÜV for the HTR-Modul design [11a] [11b]. 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.

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. During the review, several design-analysis changes as well as design changes were made by the applicant to address deficiencies identified by TÜV relative to the technical requirements. 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. 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 [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 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. Fuel technology development there was divided among three organizations: Brown Boveri Corporation (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 was 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 manufactured to precise design and manufacturing specifications, irradiation performance tested conducted to simulate the normal operating core environmental envelope and the fuel must be tested for all postulated off normal conditions via post irradiation heat up tests. The quality of the fuel manufacture with respect to heavy metal contamination and defective particles 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 is then placed in HNO_3 which will leach out the heavy metal from any fuel particles with defective SiC layers (not from particles with intact SiC layers). 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 tests result indicate a manufacturing defect rate of about $5X10^{-5}$.

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 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 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. 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 of 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, (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. Under conditions of depressurized loss of forced cooling, 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. This effective conductivity 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.

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 developed 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 fluid flow and helium density 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, including the LEU TRISO fuel. 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 one day and stabilized at a very low core power. The response to a complete loss-of-coolant accident was also simulated in an experiment with the AVR running at depressurized conditions and at low power to simulate decay heat. 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 these 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 significantly 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 between the late 1960s to 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. Some of the broken pebbles got jammed at the core outlet or in the fuel cycling 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. This resulted in a number of problems, including excessive pressure in the pneumatic fuel lifting system which necessitated reducing reactor power to 40 percent during on-line fueling.

(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 scale 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. The gradient led to larger than expected thermal stresses in the hot gas ducts and breakage of several 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.

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 Bundesamt fur Strahlenschutz (BfS) and formerly of HKM 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. The utility operating the THTR did not have the resources to overcome the political forces. As a result, the reactor was decommissioned in 1989 only 4 years after licensing.

From Mr. Hohmann's perspective, the following are the lessons to be learned from the THTR experience: (1) In-core control rods are forbidden; (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 safety advantages that the LWR cannot provide.

INTERMEDIATE STORAGE FACILITY TOUR

The delegation was taken on a tour of an fuel intermediate 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 pebble 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 horizontal 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. 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 simulated graphite pebbles. 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) {insert text here from Findings....} [27].

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 [28]. 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 of a complete list of the KTA standards [XX] 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. 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 [29].

The irradiation time for fuel pebbles in the reactor averages approximately three 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 HTR-M, 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 one year after discharge, the heat load was stated as about 760 watts. Most of the loaded casks contain fuel over ten 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.

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. No estimates for the quantities of graphite involved or anticipated values for the PBMR were presented.

TRANSFER OF KNOW-HOW FROM GERMANY TO ESKOM

Dr. Josef Schöning, General Manager, HTR-GmbH, Dr. 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 [30]. 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 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. The following technical information is viewed by the delegation as important to the safety or operational assessment of modular HTGRs:

1. To manufacture high quality fuel that consistently achieves fuel performance within expectations during irradiation and accident testing, proven manufacturing equipment, manufacturing processes and manufacturing procedures must be utilized with meticulous adherence to manufacturing quality controls for all aspects of fuel manufacture. Exact compliance essential.

2. The Natural Convection in Core with Corrosion (NAKOK) experiments were conducted at the FZJ to assess air ingress into an HTR-Modul reactor. 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. The 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 core graphite materials such as the fuel elements to oxidation-induced corrosion.

3. Specific grade designations were established and assigned to the reactor graphites that were formerly manufactured for the German reactors. These gradès 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.

4. At THTR pebble flow through the core was significantly different than was seen in 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 temperature in the center axis of the core then at the outer reflector. The pebbles at the reflector moved much more slowly. By the time the outer pebbles reached the bottom of the core the burnup was greater than predicted. This resulted in the coolant temperatures at the outer reflector being lower due to the lower pebble power output there. This in turn resulted in relatively higher sliding friction between pebbles, further slowing the pebble movement. The increased core exit temperature gradient increased thermal stresses and the failure of mechanical components. 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.

5. A special test conducted at the AVR indicated unpredicted local hot spots. In the test, approximately 20% of the 200 "melt-wire" pebbles that had passed through the core experienced significantly higher-than-expected maximum coolant temperatures. The melt-wires indicated maximum core coolant temperatures were over 1280^{o} C at full power. This is well above what had been predicted. FZJ is now preparing a report on the AVR test results. It will be provided to the 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.

6. The German safety analyses and safety evaluations for HTR design basis events involves 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, 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.

7. Extensive measurements have been performed in Germany on heated beds of graphite pebbles in flowing air. For example results with pebbles at 900^{o} C indicate graphite corrosion rates of approximately 200 milligrams of reacted O_2 per square Cm 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^{o} C to 1200^{o} C.

8. Significant operating experiences occurred at the THTR. These included: pebble breakage (without measurable increases in reactor coolant activity) due to control rod insertion; 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 problems which occurred over the few years of plant operation, overall performance for the THTR demonstration plant was viewed to be a success within the German nuclear community. The parties supporting the operation of the THTR did not have the resources to overcome the political forces seeking to shutdown AVR and they were not willing to operate the plant in light of the higher estimates of potential financial risks that had been identified. As a result, the reactor was decommissioned in 1989 only 4 years after licensing.

9. Decommissioning of the AVR and THTR is based upon a SAFESTOR approach. Significant quantities of activated graphite (containing carbon-14 and tritium) will likely require disposal at some time. In addition, the experience indicates that the design and layout of these plants did not effectively provide for ease and radiological protection of workers during the decommissioning activities.

10. 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 operation and safety analyses and safety evaluations of German high temperature pebble bed reactors. Because these agreements prohibit the ESCOM or the other involved organizations from providing the information to third parties, it will be difficult for NRC to obtain this detailed technical information from the German organizations that entered into these agreements.

Appendix A

SAFETY ASSESSMENT OF THE HTR-MODUL

The HTR-Modul is 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 for 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 by 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 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 to damage of 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.

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). Cesium diffusion starts to become significant at temperatures above 1600°C.

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; adds strength to the SiC layer.

Overall the purpose of the coatings are to prevent potential 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 reactor 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 200 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. It is not clear how defective pebbles would be found and removed from the HTR-Modul.

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 reactors).

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

1. Safety Aspects of HTR-Technology, Edmund Kersting, GRS
2. The Regulatory System in Germany, Edmund Kersting, GRS
3. New Reactor Licensing, Amy Cubbage, NRC/NRR
4. Background and Purpose for the NRC Delegation Visit to Germany on the Safety Aspects of HTGR Technology, Stuart Rubin, NRC/RES
5. NUREG-1338, Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor; P.M. Williams, T.L. King, J.N. Wilson (NRC/RES), 1989
6. Draft Update of the Draft Preapplication Safety Evaluation Report on the Modular High-Temperature Gas-Cooled Reactor; J. Dohohew (NRC/NRR), Project 672, February 26, 1996.
7. FZ-Jülich brochures: (a) Institute for Safety Research and Reactor Technology (ISR), (b) Expertise for the Future: Facilities of the Research Center Jülich, (c) High Tech on Historical Soil, (d) The Future is Our Mission
8. Large Test Facilities in HTR Development, Kurt Kugeler, Jülich Research Center
9. Concept of Inherently Safe Modular HTR, Kurt Kugeler, Jülich Research Center
10. Overview on the HTR Program in Germany, Josef Schöning, Westinghouse Reactor GmbH
11. Safety Aspects of HTR Technology, Volker Nitzki, TÜV Hannover
12. Safety Assessment of the Design of the Modular HTR-2 Nuclear Power Plant, TÜV Hannover, June 1998
13. Concept Licensing Procedure for an HTR-Module Nuclear Power Plant, Brinkmann and Will, 1990
14. Recommendation of the Reactor Safety Commission on the Safety Concept of a High-Temperature Modular Power Plant, 250th Meeting of the RSK, January 24, 1990
15. Pebble Bed Fuel Element Research and Development and Industry Production in Germany, Heit, Froschauer, NUKEM Nuclear GmbH, Germany
16. HTR Fuel Manufacture, Irradiation and Accident Condition Testing, Heinz Nabielek, Jülich Research Center, Germany

17. Long Time Experience with the Development of HTR Fuel Elements in Germany, H. Nickel, H, Nabielek, G. Pott, FZJ; A.W. Mehner, Advanced Nuclear Fuels GmbH, Duisburg, Germany, International HTR Fuel Seminar, Brussels, Belgium, 2001

18. Fuel Pebbles Operational Experiences Irradiation and Post-Irradiation Examination, G. Pott, H. Nabielek

19. Nuclear Graphite for the HTR- Research, Development and Industrial Production, Institute for Safety Research and Reactor Technology, Jülich Research Center, Germany, Gerd Haag, FZJ

20. Development of Reactor Graphite, G. Haag, FZJ, et al, Journal of Nuclear Materials, 1990

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

22. AVR Operational Experience, Overview; Wahlen, Pohl; July 2001

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, MW-NRR (in German, presented with translator)

27. (a) Ceramic Coatings for HTR Graphitic Structures - Tests and Experiments with SiC-Coated Graphitic Specimens, B. Schroeder et al, FZJ/T.U.-Aachen (article);
(b) Institute for Safety Research and Reactor Technology (ISR), Julich Research Center (FZJ), NACOK: Natural Convection in Core with Corrosion

28. Summary, KTA 3321; Ivar Kalinowski

29. Waste Management - Spent AVR Fuel Elements, Kurt Kugeler, Jülich Research Center

30. 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

Additional documents not explicitly referenced in Attachment 3:

31. THTR-300 Coolant Activity, an Indicator of Fuel Performance, K Rollig

32. TÜV, erstellt fuer BMI, Sicherheitskriterien fuer Anlagen zur Energieerzeugung mit gasgekuelten Hochtemperaturreaktoren, Entwurf September 1980 (59 pages)

33. GRS, Gesellschaft fuer Reaktorsicherheit mbH, Sicherheitsuntersuchungen fuer Hochtemperaturreaktoren: Untersuchungen zu ausgewaelten risiko-bestimmenden Ereignisablaeufen fuer den Thorium-Hochtemperatur-Reaktor THTR-300 in Hamm-Uentrop - Abschlussbericht, GRS-A-1412 (March 1988) ~ 300 pages in German, Abstract in English, Translation of Title: Safety Studies for High-Temperature Reactors: Studies of Selected Risk-Determining Event Profiles for the Thorium High-Temperature Reactor THTR-300 in Hamm-Uentrop - Final Report

34. GRS, Gesellschaft fuer Reaktorsicherheit mbH, Risikoorientierte Analyse fuer Hochtemperaturreaktoren (Phase 1) - Abschlussbericht, GRS-A-1734 (December 1990) ~300 pages in German, Abstract in English, Translation of Title: Risk-Oriented Analysis for High-Temperature Reactors (Phase 1) - Final Report

35. Gerd Brinkman, et al, Concept Licensing Procedure for an HTR-Module Nuclear Power Plant, (N.E.D. 1990)

36. Bundesanzeiger, 28. April 1990, RSK 250. Sitzung am 24. Januar 1990, Empfehlung zum Sicherheitskonzept einer Hochtemperatur-Modul-Kraftwerksanlage (also as English translation: 250th Meeting of the Reactor Safety Commission, January 24, 1990, Recommendation of the Reactor Safety Commission on the Safety Concept of a High-Temperature Modular Power Plant)

37. G. Dietrich et al, HKG Hamm-Uentrop, Decommissioning of the Thorium High Temperature Reactor (article)

38. K. Hofmann, TÜV Essen, J.B. Fechner, BMI, Proposed Safety Criteria for High-Temperature Gas-Cooled Reactors, IAEA-CN-39/26, reprint from Current Nuclear Power Plant Safety Issues Vol. II, IAEA 1981

39. D. Niephaus et al, FZ Julich, Experience with Interim Storage of Spent HTR Fuel Elements and View to Necessary Measures for Final Disposal (article)

40. W. Stratmann, M. Baechler, Review of Some Aspects of Radiological Interest During the Establishment of the Safe Enclosure of the THTR-300 Plant (article)

41. C. Marnet, M. Wimmers, U. Birkhold, Decommissioning of the AVR Reactor, Concept for the Total Dismantling (article)

42. H. Nickel, H. Nabielek, G. Pott, A.W. Mehner, Long Time Experience with the Development of HTR Fuel Elements in Germany, HTR-TN International HTR Fuel Seminar, Brussels, Belgium, February 1-2, 2001

43. V. Kaminski, H. Reutler (Interatom GmbH), Instandhaltung der Primaerkreislaufkomponenten des HTR-MODUL (Maintenance of the Primary Circuit Components of the HTR-Module), paper from the conference Jahrestagung Kerntechnik '86

44. Sicherheitstechnische Grundlagen fuer die Katastrophenschutzplannung am THTR-300, (Safety Technology Fundamentals for Catastrophe Protection Planning at the THTR-300), KFA Julich, 1984

45. K. Kugeler, H. Neis, G. Ballensiefen, Fortschritte in der Energietechnik fuer eine wirtschaftliche, umweltschonende und schadenbegrenzende Energieversorgung - Prof. Dr. Rudolf Schulten zum 70. Geburtstag, (Progress in the Energy Technology for an Economical, Environment-Preserving, and Damage-Limiting Energy Supply - In Honor of Prof. Dr. Rudolf Schulten upon his 70th Birthday), Forschungszentrum Julich GmbH, Institut fuer Sicherheitsforschung und Reaktortechnik, Mongraphien des Forschungszentrums Julich, Band 8/ 1993