Safety Aspects

of

HTR-technology

NRC visit in Germany

July 23rd, 2001 – July 26rd, 2001

Safety Aspects of HTR-Technology

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NACOK – **Na**tural Convention in **Co**re with **C**orrosion	Institute for Safety Research and Reactor Technology (ISR) Research Centre Jülich, Heit, Froschauer	Transparencies
Fuel element R & D and industrial production in Germany	Tessag, NUKEM	Transparencies
HTR Fuel Manufacture, Irradiation and Accident Condition Testing	Institute for Safety Research and Reactor Technology (ISR) Research Centre Jülich, H. Nabielek	Transparencies
Experiment R2-K3 Temperature Distribution		
Fuel Pebbles operational Experiences Irradiation and Postirradiation Examination	Institute for Safety Research and Reactor Technology (ISR) Research Centre Jülich, G. Pott, H. Nabielek	Paper
Long Time Experience with the Development of HTR Fuel Elements in Germany	Institute for Safety Research and Reactor Technology (ISR) Research Centre Jülich, H. Nickel, H. Nabilek, G Pott, A. W. Mehner	Publication
Ceramic Coatings for HTR Graphitic Structures-Tests and Experiments with SIC-Coated Graphitic Specimens	Institute for Safety Research and Reactor Technology (ISR) Research Centre Jülich, B. Schroeder, W. Schenk, Z. Alkan, R. Conrad	Publication
Safety Aspects of HTR Technology Nuclear Graphite for the HTR	Institute for Safety Research and Reactor Technology (ISR) Research Centre Jülich, G. Hag	Transparencies and Publication
Heat Transfer, Fluid Flow and Power Feedback in Pebble-Bed Reactors	Institute for Safety Research and Reactor Technology (ISR) Research Centre Jülich, W. Scherer	Transparencies
AVR operation experiences, test programms, overview, highlights, lessons to be learnt	Institute for Safety Research and Reactor Technology (ISR) Research Centre Jülich, Edgar Wahlen, Peter Pohl	Transparencies

Title	Issued by	Document Type
Decommissioning of the AVR Reactor, Concept for the total Dismantling	Institute for Safety Research and Reactor Technology (ISR) Research Centre Jülich, C. Marnert, M Wimmers	Publication
Unloading of the Rector core and spent fuel Management of THTR 300	STEAG Kernenergie GmbH, Essen, S. Plätzer, M Mielisch	Publication
R & D on Intermediate Storage and Final Disposal of spent HTR Fuel	Research Centre Jülich, J. Fachinger, H. Brücher, R. Duwe	Publication
THTR operation experience, test programms, overview, highlights, lessons to be learned	Hochtemperatur-Kernkraftwerk GmbH, Dr. Ivar Kalinowski, Guenther Dietrich	Transparencies
The situation of the THTR in October 1989	VGB Kraftwerkstechnik R. Bäumer	Publication
Ausgewählte Themen aus dem Betrieb des THTR 300	VGB Kraftwerkstechnik R. Bäumer	Publication
THTR 300-Erfahrungen mit einer fortschrittlichen Technologie	Hochtemperatur-Kernkraftwerk GmbH, R. Bäumer, Hamm	Publication
THTR Commissioning and Operating Experience	Hochtemperatur Kernkraftwerk, Hamm, Dr. Ivar Kalinowski, R. Bäumer	Paper on the "Eleventh International Conference on the HTGR"
Core Physics and Pebble Flow, Examples from THTR Operation	Dr. Helga Kalinowski	Speech and Transparencies
Review of some Aspects of Radiological Interest During the Establishment of the Safe Enclosure of the THTR 300 plant	STEAG Kernenergie GmbH, Essen, W. Stratmann Noell-KRC-Energie- and Umwelttechnik GmbH, M. Bächler	Publication
The THTR-300 Coolant Gas Activity, an Indicator of Fuel Performance	Hochtemperatur-Reaktorbau GmbH, Mannheim, K. Röllig	Publication
Pebble Flow Experimental Results Review	Institute for Safety Research and Reactor Technology (ISR) Research Centre Jülich, A. Kleine-Tebbe	Report in Part
Decommissioning of the Thorium high Temperature Reactor (THTR 300)	Hochtemperatur-Kernkraftwerk GmbH, Hamm-Uentrop, G. Dietrich. W. Neumann, N. Röhl	Publication
Waste Management (spent HTR-fuel elements)	Institute for Safety Research and Reactor Technology (ISR) Research Centre Jülich, Prof. Kurt Kugeler	Transparencies

Title	Issued by	Document Type
Experience with the interim storage of spent HTR fuel elements and a view to necessary measures for final disposal	Institute for Safety Research and Reactor Technology (ISR) Research Centre Jülich, D. Niephaus, S. Storch, S Halaszovich	Publication
Examples of Safety Assessment - Shutdown Margins - Reactivity Feedback, Inherent Safety - Power Density Distribution (Design and Control)	TUEV-Nord	Transparencies
Safety Assessment of the HTR Module in Germany	TUEV-Nord	Transparencies and Publication
Review of the safety concept of the HTR 2 module reactor plant	TUEV, Hannover, H. Helmers, H. Knieper	Publication
THTR 300 MWe Prototype Reactor - Safety Assessment	RWTUEV Anlagentechnik, K. Hofmann, W. Trapp	Publication
Proposed safety criteria for high-temperature gas-cooled reactors	TUEV-Arbeitsgemeinschaft Kerntechnik West, K. Hofmann, Federeal Ministry of the Interior, Bonn, J.B. Fechner	Publication, Reprint from "Current Nuclear Power Plant Issues" IAEA
Hochtemperaturreaktor-Technologie, Genehmigungsentscheidende Sicherheitsaspekte beim THTR	Ministerium für Wirtschaft und Mittelstand Energie und Verkehr des Landes NRW, Düsseldorf, W. Hohmann	Paper
Empfehlung zum Sicherheitskonzept einer Hochtemperatur-Modul-Kraftwerksanlage	Empfehlung der Reaktor-Sicherheitskommission, 250. Sitzung am 24.01.1990	Bundesanzeiger, Nr. 81, vom 28.04.1990
Rules and Standards for High Temperature Reactors	Secretariat of Nuclear Safety Standards Commission (KTA) at BFS, Dr. Ivar Kalinowski	
Safety Standards of the Nuclear Safety Standards Commission (KTA) HTR-related Standards	GRS, Cologne	Cover sheets
Einführung zum Regelvorhaben KTA 3221 "Metallische HTR-Komponenten"	KTA	Paper
KTA 3221.1 Metallische HTR-Komponenten, Teil 1	KTA	KTA-Rule
KTA 3221.1 Metallische HTR-Komponenten, Teil 3	KTA	KTA-Rule

Title	Issued by	Document Type
KTA 3232 Keramische Einbauten in HTR-Reaktordruckbehältern	KTA	KTA-Rule
THTR Commissioning and Operating Experience	Hochtemperatur Kernkraftwerk, Hamm, R. Bäumer, Dr. Ivar Kalinowski	Publication
TÜVIS Sicherheitskriterien für Kernkraftwerke Sicherheitskriterien für Anlagen zur Energieerzeugung mit gasgekühlten Hochtemperaturreaktoren	RWTÜV, Essen	
Fortschritte in der Energietechnik	Institute for Safety Research and Reactor Technology (ISR) Research Centre Jülich, K. Kugeler, H. Neis, G Ballensiefen	Cover sheet
The Chernobyl Accident	Nuclear Society International, Moscow, 1993, L.M. Vecsler, V.I. Obodzinsky, V.K. Popov	Publication
Sicherheitstechnische Grundlagen für die Katastrophenschutzplanung am THTR-300	Institute for Safety Research and Reactor Technology (ISR) Research Centre Jülich, D. Diephaus, S. Storch, S Halaszovich	Part of Report
Additional papers which were not distributed during the visit		
Status of the pebble bed modular reactor	D. R. Nicholls Eskom, South Afrika	Nuclear Energy, 2000, 39, No. 4, Aug., 231-236
Safety concept and design of a modular gas cooled high-temperature reactor from the viewpoint of externally generated load cases	K. Peters, Interatom-Bensberg, Federal Republic of Germany U. Müller-Frank, Interatom-Bensberg, Federal Republic of Germany W. Steinwarz, Interatom-Bensberg, Federal Republic of Germany	Report
High temperature gas cooled reactor technology development. Proceeding of a Technical Committee meeting held in Johannesburg, South Africa, 13-15 November 1996	IAEA-TECDOC-968	Publication

Title	Issued by	Document Type
Safety related design and economic aspects of HTGRs. Proceeding of a Technical Committee meeting held in Beijing, China, 24 November 1998	IAEA-TECDOC-1210	Publication
Current status and future development of modular high temperature gas cooled reactor technology	IAEA-TECDOC-1198	Publication
Irradiation Damage in Graphite due to Fast Neutrons in Fission and Fusion Systems	IAEA-TECDOC-1154	Publication

Revised Agenda

Visit of the NRC-Delegation to Germany

on the Topic

Safety Aspects of HTR Technology

for Monday, 23 July to Thursday, 26 July 2001

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, Vogel)
- 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,
Institute for Safety Research and Reactor Technology, Building 16.5,
Seminar-Room
Begin: 10:00 a.m.

Main Topic:
Research at FZJ related to HTR

- Welcome to the Research Centre Jülich (Eisenbeiß)
- Information on the activities of the Research Centre Jülich
  (Eisenbeiß)
- Overview of research and development (R & D) at the FZJ related to HTR technology
  (Kugeler)
- Fuel element R & D and industrial production in Germany
  (Heit, Froschauer)
- Lunch at Seacasino, Faculty Club, *13.00 h*
- 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
  (Scherer)

Wednesday, 25 July 2001, Research Centre Jülich, FZJ,
Institute for Safety Research and Reactor Technology, Building 16.5,
Seminar-Room
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, Barnert)

- Aspects of waste management
  (Kugeler)

- Lunch at Seacasino, Faculty Club, *13.00 h*

- Visit to intermediate storage facility, *14.15 – 15.30 h*
  (Halaszovich)

- Short Visit to AVR, *15.30 – 16.00 h*
  (Storch, Marnet, Wahlen, Pohl)

Optional:

- Visit to experimental hall no. IV: experimental work on Self-Acting Removal of Decay Heat and Natural Convection in Core with Corrosion, *16.00 – 16.30 h*
  (Barnert, Nießen, Schröder, Kugeler)

Thursday, 26 July 2001, GRS, Schwertnergasse 1, 50667 Köln

Begin: 10:00 a.m., Room 611

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

Safety Aspects of HTR-Technology

List of Participants

Date: 23. July 2001

Location: GRS, Cologne

Name	Affiliation
Carlson	US-NRC
Cubbage	US-NRC
Faulkner	US-NRC
Murray	US-NRC
Perin	US-NRC
Rubin	US-NRC
Shoop	US-NRC
Barnert	FZJ
Kugeler	FZJ
Nickel	FZJ
Schöning	Westinghouse-HRB
Brinkmann	Framatome-ANP
Nitzki	TUEV-Nord, Hannover
Vogel	TUEV-Nord, Hannover
Bönigke	GRS
Kersting	GRS

Safety Aspects of HTR-Technology

List of Participants

<u>Date</u>: 24. July 2001

<u>Location:</u> FZJ, Jülich

Name	Affiliation
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Nickel	FZJ
Nabielek	FZJ
Pott	FZJ
Schenk	FZJ
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Scherer	FZJ
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Froschauer	Nukem
Schöning	Westinghouse-HRB
Bönigke	GRS
Kersting	GRS

Safety Aspects of HTR-Technology

List of Participants

Date: 25. July 2001

Location: FZJ, Jülich

Name	Affiliation
Carlson	US-NRC
Cubbage	US-NRC
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Nickel	FZJ
Nießen	FZJ
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Schröder	FZJ
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Schöning	Westinghouse-HRB
H. Kalinowski	BfS
I. Kalinowski	BfS
Dietrich	RWE-Power
Bönigke	GRS
Kersting	GRS

Safety Aspects of HTR-Technology

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Location: GRS, Cologne

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September 25, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

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

On July 23–26, 2001, a six-member NRC delegation met with German scientists and engineers. The purpose of the meetings was to open up channels of communication for follow-up exchanges of technical information and to improve the Agency's knowledge and understanding of advanced high-temperature gas-cooled reactor (HTGR) design and technology. The visit was arranged in connection with the NRC staff action plan to expand staff expertise and understanding of world-wide experience in technology specifically applicable to the Pebble Bed Modular Reactor (PBMR) and the Gas Turbine-Modular Helium Reactor (GT-MHR). The delegation consisted of Stuart Rubin and Donald Carlson, Office of Nuclear Regulatory Research (RES), Amy Cubbage and Undine Shoop, Office of Nuclear Reactor Regulation (NRR), Alex Murray, Office of Nuclear Materials Safety and Safeguards (NMSS), and Howard Faulkner, Office of International Programs (OIP). Two (2) days were spent in Cologne and 2 days were spent at the Jülich Research Center (formerly the Jülich Nuclear Research Center) FZJ. Mr. Edmund Kersting, Head of International Programs, the Company for Reactor Safety (GRS), organized the visit on behalf of the NRC.

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

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

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

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

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

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S. Collins, NRR
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NAME	ACubbage		AMurray by JGiitter		HFaulkner		RZimmerman		AThadani	
DATE	08/22/01*		09/07/01*		08/23/01*		------------		09/25/01*	

Memorandum dated: September 25, 2001

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

Distribution w/o atts.:

CAder, RES	WBorchardt, NRR	MGamberoni, NRR
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HGraves, RES	DJackson, NRR	SSteele, NMSS
NLauben, RES	JStrosnider, NRR	WGleaves, NMSS
RMeyer, RES	FGrubelich, NRR	VPerin, NMSS
CTinkler, RES	EImbro, NRR	JGiitter, NMSS
SArndt, RES	HAshar, NRR	MWeber, NMSS
JUhle, RES	RCaruso, NRR	SBall, ORNL
CGingrich, RES	TCheng, NRR	
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DSARE R/F	EConnell, NRR	

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Visit of the NRC-Delegation to Germany

on the Topic

Safety Aspects of HTR Technology

for Monday, 23 July to Thursday, 26 July 2001

Attachment 2

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
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- Overview on safety assessment of HTR-Module in Germany
  (Nitzki)
- Know-how transfer to ESKOM for a PBMR
  Safety analysis report HTR-Module
  Access to the total HTR-know-how, Consultancy work
  (Schöning, Brinkmann, Kugeler)

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

Main Topic:

Research at FZJ related to HTR

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

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

Main Topic:

“Operational Experiences of AVR and THTR and Visits of Experimental Facilities”

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

Optional:

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

Thursday, 26 July 2001, GRS, Schwertnergasse 1, 50667 Köln

Begin: 10:00 a.m., Room 610

Main Topic:
Regulatory Aspects and Safety Assessment

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

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

INTRODUCTION

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

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

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

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

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

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

HIGH-TEMPERATURE REACTOR RESEARCH AT THE JÜLICH RESEARCH CENTER

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

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

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

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

In recent years, increasing attention has gone to the study of advanced reactor safety features that go beyond current high-temperature reactor (HTR) design and technology [9]. These ongoing R&D efforts fall under the heading of what Professor Kugeler calls "catastrophe-free" nuclear technology. Included is the developmental testing of silicon carbide coatings to cover and seal (i.e., isolate) the graphite surface of pebble fuel elements and graphite reflector blocks from chemical attack. If successful, such ceramic coatings would prevent self-sustaining graphite oxidation in the case of a potential air-ingress event such as might result, for example, from a postulated large break in the reactor pressure vessel. Towards eliminating the possibility of such large vessel breaks, the "catastrophe-free" developmental work further includes the design and scaled over-pressure testing of burst-protected reactor pressure vessels made of prestressed steel. Also under consideration are mitigation features that utilize sand or other granulates to block the continued ingress of air after a postulated vessel break and thereby prevent or limit graphite oxidation.

HIGH-TEMPERATURE REACTOR DESIGN ACTIVITY IN GERMANY

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

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

In addition to the HTRs that operated, a number of additional designs were developed in Germany to varying degrees. These designs ranged in power level from 10 MWt to 3000 MWt. One of the designs featured in the discussions was the HTR-Modul. The HTR-Modul is 85 MWe, with a design similar to the proposed PBMR except that the HTR-Modul incorporates a steam generator in the power conversion system. This HTR-Modul design was characterized as having a low power density, active and passive safety features, a reactor protection system, and a confinement envelope. Safety features of the HTR-Modul design include a strongly negative temperature coefficient of reactivity that ensures passive initial shutdown upon loss of forced cooling, passive decay heat removal through the reactor vessel wall after a loss of coolant, a primary pressure boundary incorporating the “leak before break” design criterion, and location of the steam generator and helium cross duct at an elevation below the reactor core. For a postulated break of the helium cross duct, the resulting “diving bell” geometry was said to trap hot helium above the break and thereby delay by several days the onset of air circulation and graphite oxidation in the core. For postulated events without active scram, the core is predicted to passively shut down and then return to critical after a day or two but reach no more than 1% of full power in the absence of forced helium circulation.

SAFETY ASSESSMENT OF THE HTR-MODUL

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

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

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

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

In October 1989, the TÜV safety assessment report was provided to the Reactor Safety Commission (RSK) for their review. The RSK stated that the HTR-Modul design has favorable safety-related characteristics even in the range beyond the design basis, and they concluded that the design of the HTR-Modul met the safety requirements imposed on nuclear facilities in Germany at the time [14].

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

PEBBLE FUEL ELEMENT RESEARCH, DEVELOPMENT, AND INDUSTRIAL PRODUCTION

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

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

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

German pebble fuel element R&D and production activities included many fuel variations: initially high- and later low-uranium enrichment (for fuel cycle reasons), fuel materials (Th, U, O, C) and coated particle design (BISO particle and later TRISO particle) and manufacturing processes. Many of the fuel element design variations were irradiated in materials test reactors (MTRs) as well as the AVR at Jülich. The AVR served as a large-scale (non-materials test reactor) irradiation facility for the evolving German pebble fuel element designs.

Development and irradiation testing on the fuel which was to become the reference low enriched uranium dioxide (LEU) fuel element design for future German HTRs (e.g., HTR-Modul) occurred over a period of about 10 years from the late 1970s to the late 1980s. This fuel was manufactured and irradiation tested and successfully used as the standard AVR reload fuel design beginning in the early 1980s until the reactor was permanently shutdown 1988.

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

PEBBLE FUEL ELEMENT IRRADIATION AND ACCIDENT SIMULATION TESTING

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

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

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

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

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

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

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

HIGH-TEMPERATURE REACTOR NUCLEAR GRADE GRAPHITE

Dr. Gerd Haag, Institute for Safety Research and Reactor Technology, Jülich Research Center, discussed the subject of nuclear graphite for the HTR, including graphite research and development and industrial production [19] [20]. The microscopic carbon structure of graphite components may be viewed at the level of the coke particles, the alignment of crystallites within the coke particles and the arrangement of individual atoms within the crystallites. However, the behavior and material properties of graphite components when exposed to an irradiation environment can be understood only when investigated at the level of the individual atoms within the crystallites (i.e., the lattice structure of the carbon atoms). Neutron irradiation causes individual atoms in graphite to be knocked out of their latices into interstitial positions between the latices. These carbon atom relocations cause the change in dimensions (growth, shrinkage) in graphite components as well as changes in its material properties. A single 1-2 MeV neutron can displace on the order of 20,000 carbon atoms in graphite crystallite. Initially, shrinkage occurs in an irradiation environment but with increased fast fluence expansion will occur. Depending on the isotropy of the graphite the amount of shrinkage in the orthogonal dimensions under fast fluence can be very similar (isotropic) or fairly different (anisotropic).

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

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

In view of these observations, Dr. Haag provided a number of recommendations related to nuclear grade graphite: (1) graphite for the PBMR reflector components should be produced from material sources on a "best guess" basis using still existing procedures and experience, (2) data for stress analyses (e.g., irradiation induced growth strains and stresses, coefficient of thermal expansion for calculating thermal strains and stresses) should be deduced from the properties developed for similar materials that were previously tested extensively in the

German irradiation programs, (3) an international database for graphite should be established and should be composed of data from the US, UK, Japan, Germany and France, and should be supported by possible users, and (4) for future HTR projects, development and irradiation testing of new graphites should resume as soon as possible.

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

PEBBLE BED REACTOR CORE HEAT TRANSFER AND FLUID FLOW

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

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

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

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

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

AVR OPERATING EXPERIENCE, TESTING, AND LESSONS LEARNED

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

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

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

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

THTR OPERATING EXPERIENCE, TESTING, AND LESSONS LEARNED

The 300 MWe Thorium High Temperature Reactor (THTR) was designed during the late 1960s and early 1970s as a demonstration plant toward the planned commercialization of large-scale pebble-bed HTRs in Germany. The long time span between the start of THTR construction in 1972 and initial power operation in 1986 was necessitated largely by design and analysis changes for addressing the evolving regulatory requirements related to external events. Meanwhile, in the early 1980s, development efforts in Germany started a gradual shift away from large-scale HTRs toward more inherently safe modular designs with lower power density, like the HTR-Modul design of the late 1980s. This shift parallels the shift in HTGR development in the United States, from the Fort St. Vrain reactor (and larger HTGR designs such as the Fulton plant) of the 1970s to the lower power modular HTGR designs of the mid-1980s to early 1990s, leading up to the GT-MHR design development program.

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

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

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

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

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

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

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

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

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

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

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

THTR CORE PHYSICS AND PEBBLE FLOW

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

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

Achieving the optimal pebble flow and loading pattern for the reactor took considerable effort and needed to be continuously monitored. The core diameter-to-height ratio of 1:1 of the THTR was found to promote the increased velocity in the central core region. The ratio was changed to 1:3 (long slender core) for the later designs to achieve passive decay heat removal characteristics and to allow control and shutdown using reflector control elements only. This change is also expected to improve pebble flow so that the flow across the core is closer to the model predictions. An additional reason why the THTR core did not follow the predicted behavior is because all the experiments used to develop the predicted behavior used air, which results in a pebble flow friction coefficient significantly lower than that in helium. These differences had a significant impact and rendered the tests unreliable for predicting the actual core pebble flow and the resulting neutronic behavior.

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

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

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

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

THTR SAFETY ASSESSMENT

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

During operation of the THTR, several operational and design problems were observed, but these issues were not considered to be of high safety significance by operations, design, or regulatory organizations. These included breakage of fuel elements caused by the insertion of the in-core control rods, failure of bolts in the thermal insulation of the hot-gas ducts due to an elevated temperature gradient at the core exit, difficulties with the fuel handling system that initially limiting refueling activities to less than 40% power, and larger than anticipated quantities of graphite dust in the primary system. Despite these operational and design problems, the THTR demonstration plant was considered a technical success and was viewed as generally providing confirmation of the safety and the feasibility of an HTR based on the pebble bed reactor core concept.

THTR LICENSING SAFETY ISSUES

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

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

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

INTERMEDIATE STORAGE FACILITY TOUR

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

CASTOR-type storage casks are used. Several casks were being delivered during the visit. Remote operations place two canisters – one on top of the other – inside each cask. An end closure with two metallic O-ring seals is then inserted. After bolting, the operators pressurize the space between the seals with helium, typically to 5–7 bar of pressure (1.01325 bar = 14.72 psia). Sensors continuously monitor the helium pressure between the seals and alarm on low pressure (i.e., as indicative of a leak; typically at a pressure of 3 bars). Filled casks are vertically oriented in an array that provides adequate spacing for air cooling. The NRC delegation viewed the cask storage area. This consisted of a vault-like building with reinforced concrete walls (nominally 1.3 m thick) surrounding the cask array. Approximately 120 casks were visible containing spent pebbles from the AVR. The IAEA maintains cameras at various locations for safeguards purposes.

AVR SITE TOUR

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

EXPERIMENTAL FACILITY TOUR

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

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

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

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

HIGH-TEMPERATURE REACTOR CODES AND STANDARDS

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

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

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

Dr. Kalinowski provided the delegation with a complete list of the KTA standards including the HTR safety standards which were established by the KTA subcommittee for HTR standards. The HTR safety standards include standards for metallic HTR components, standards for monitoring radioactivity in HTRs and standards for reactor core design for HTRs including calculation of the material properties of helium, heat transfer in spherical fuel elements, loss of pressure through friction in pebble bed cores, thermal-hydraulic analytical models for stationary and quasi-stationary conditions in pebble bed cores, and systematic and statistical errors in the thermal-hydraulic core design of the pebble bed reactor.

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

SPENT NUCLEAR FUEL AND ASPECTS OF WASTE MANAGEMENT

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

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

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

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

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

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

The presentation also discussed final storage (disposal) options. FZJ has investigated using interstitial steel balls within the pebbles and silicon carbide filling as methods for increasing the conductivity and performance of waste disposal packages. Samples were passed around. Box, drum, and pressure-resistant disposal packages have been investigated and have been analytically shown to meet dose criteria. Analytical curves also compared the doses from disposal of the graphite fuel pebble with the same quantities of radionuclides in glass; the fuel pebble doses were lower. Some test data indicated a cesium leach rate of 100 Bq/day from a fuel pebble immersed in simulated groundwater. Curves were shown comparing fuel pebble toxicity to the uranium ore. These implied a time period of around 100,000 years before the HTR fuel toxicity equaled that of the natural ore. No specifics were given. Additional toxicity/time curves were presented for partition and transmutation. These displayed a reduction of the time period to around 1000 years for comparable toxicity to the uranium ore. FZJ acknowledged that additional water immersion, leaching testing, and disposal analyses need to be performed.

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

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

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

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

TRANSFER OF KNOW-HOW FROM GERMANY TO ESKOM

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

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

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

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

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

KEY FINDINGS

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

1. The manufacture of high quality fuel that consistently achieves fuel performance within expectations during irradiation and accident testing requires meticulous adherence to proven manufacturing equipment, processes and procedures and precise adherence to established quality measures for all aspects of fuel manufacture. Exact compliance in these areas is essential. Based on Germany’s 25 years of experience with fuel development and manufacture, a maximum initial defect fraction of $6x10^{-5}$ has been specified.

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

3. The Natural Convection in Core with Corrosion (NACOK) experiments were conducted at Jülich to assess air ingress into an HTR-Modul reactor for a postulated break in the lower hot gas duct. From the experiments it was found that after a depressurization accident caused by a postulated break in the helium cross duct near the bottom of an HTR-Modul pressure vessel, the “diving bell” geometry will initially limit diffusion mixing of outside air with hot helium in the system. For cases with no additional breaks

postulated in the pressure boundary above the cross duct, the diving bell geometry was found to provide a “grace period” (i.e., time delay) before the onset of natural convection flow of air through the core. After the grace period, natural circulation of air through the core begins, subjecting the graphitic materials of the core supports, pebble fuel elements, and reflector blocks to oxidation-induced corrosion.

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

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

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

7. Due to the absence of instrumentation within the pebble bed core, a special test was conducted at the AVR with melt-wire pebbles. The test indicated unpredicted local hot spots. In the test, approximately 20% of the 200 “melt-wire” pebbles that passed through the core at full power were found to have experienced maximum coolant temperatures above 1280°C. This was well above what had been predicted. FZJ is

now preparing a new evaluation and analysis of the AVR melt-wire test results and the resulting report will be provided to the NRC staff when it is completed. The report is expected to provide insights with regard to: (a) validating or correcting the code-predicted maximum fuel operating temperatures in a pebble bed reactor design, and (b) assessing the need for similar tests and measurements for future pebble bed reactors.

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

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

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

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

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

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

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

CONCLUSIONS

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

Appendix A

SAFETY ASSESSMENT OF THE HTR-MODUL

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

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

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

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

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

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

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

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

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

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

Appendix B

PEBBLE FUEL ELEMENT RESEARCH, DEVELOPMENT, AND INDUSTRIAL PRODUCTION

Pebble Fuel Design

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

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

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

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

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

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

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

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

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

Pebble Fuel Element Manufacture

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

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

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

Each kernel is coated into a TRISO particle using a fluidized bed coater qualified for a 5 kg batch (lot) size. Each coating layer is added via a chemical vapor deposition (CVD) processes in a sequential layering process. The CVD process decomposes gaseous species at temperature in a high surface area medium (the kernels, as the fluidizing bed). The kernels act as nucleation sites for the decomposition which grows the various layers. Each coating is made from a mixture of a carrier gas (typically Argon) and a coating gas which depends on the layer involved. The silicon carbide layer is coated using H_2 as the carrier gas and CH_3SiCl_3 as the coating gas. As each layer in turn is added, the particle diameter increases from the 500 µm UO_2 kernel size to the 1000 µm diameter of the finished coated particle. The UO_2 fuel kernels result in limited heavy metal contamination inside the coater and represents the source for heavy metal contamination outside the SiC layer in the finished particles. The Nukem fuel plant had a particle fuel capacity of approximately 2 MTHM/yr. Finished particles are then characterized. The last step is to provide a 100 µm overcoat of pyrolytic carbon. The overcoat provides a protective layer for the finished particles to prevent damage and breakage during the high-pressure pressing in the graphite matrix in manufacture of the pebbles.

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

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

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

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

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

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

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

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GRS Company Profile

Safety Aspects of HTR-Technology

NRC visit to Germany

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Tasks, Objectives and Competence of GRS

Central technical and scientific expert organisation for nuclear safety and waste management in Germany

Task

Assess and improve the safety of technical facilities

Objective

Protect man and the environment from the hazards of technology

Competence

- Interdisciplinary knowledge
- Advanced methods
- Qualified data

Company Locations and Technical Branch Offices

9707202

Governing Bodies of GRS

Shareholders' meeting

The shareholders are:

- the Federal Republic of Germany (46.1%)
- the Free State of Bavaria (3.85%)
- the *Land* of North Rhine-Westphalia (3.85%)
- the technical inspection agencies (TÜVs) and the Germanischer Lloyd (3.85 each, together 46.2%).

Supervisory board (12 members)

Chairman: Staatssekretär Rainer Baake

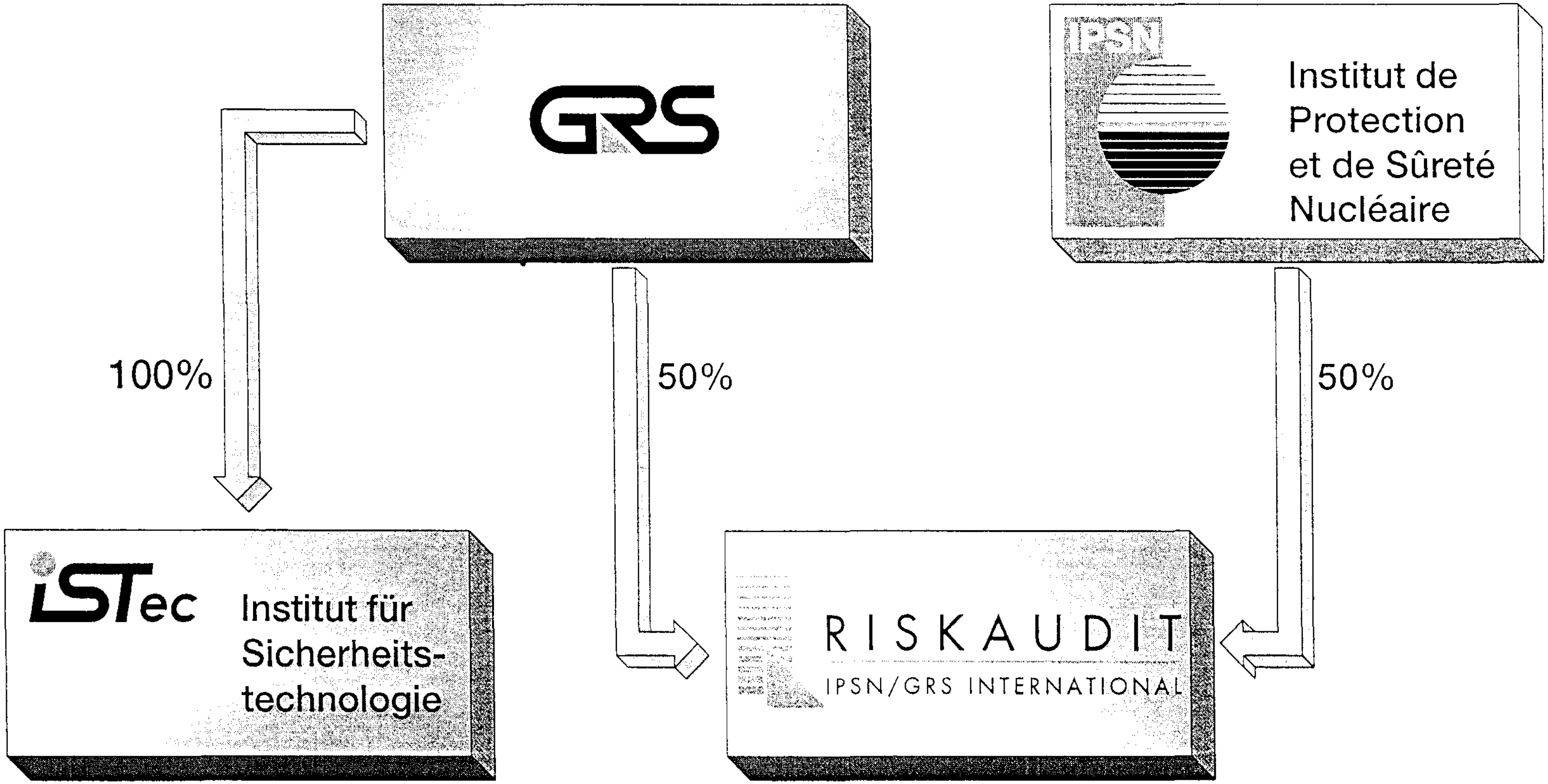
Vice-chairman: Prof. Dr.-Ing. Bruno O. Braun

Managing directors

Prof. Dr. Dr.-Ing. h.c. Adolf Birkhofer

Dr. Walter Leder

Subsidiaries of GRS



Customers 1999

GRS is exclusively financed through contracts.

BMU/BfS 45%

Analyses of fundamental nuclear safety issues and ad-hoc assessments of current issues of power plant operation.

BMBF/ BMWi 32%

Research and development in the field of nuclear facilities and waste management, including the final storage of radioactive waste.

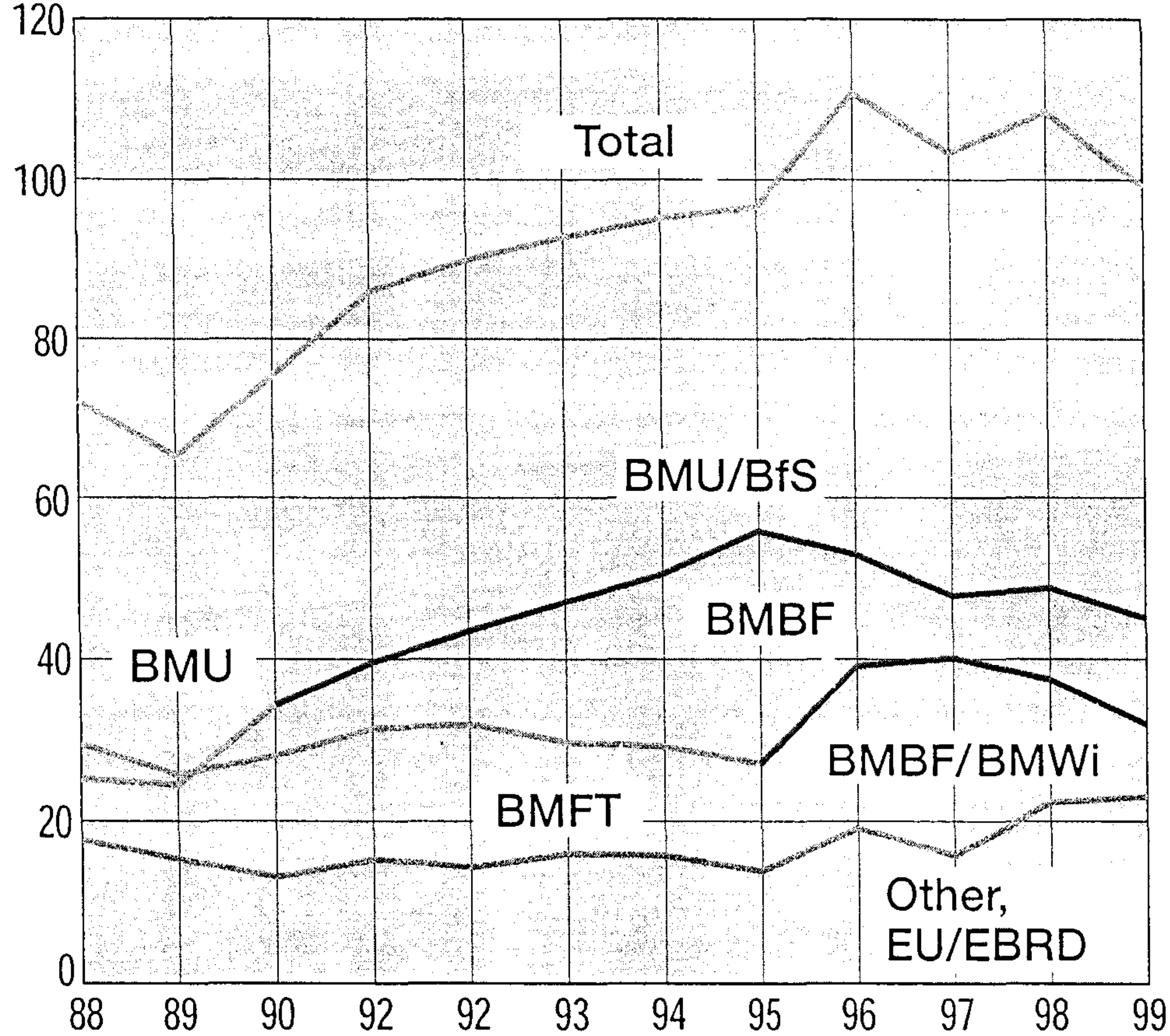
Other 17%

Safety analyses and assessments of nuclear facilities in Germany and abroad, providing expert advice to foreign supervisory and licensing authorities.

EU/EBRD 6%

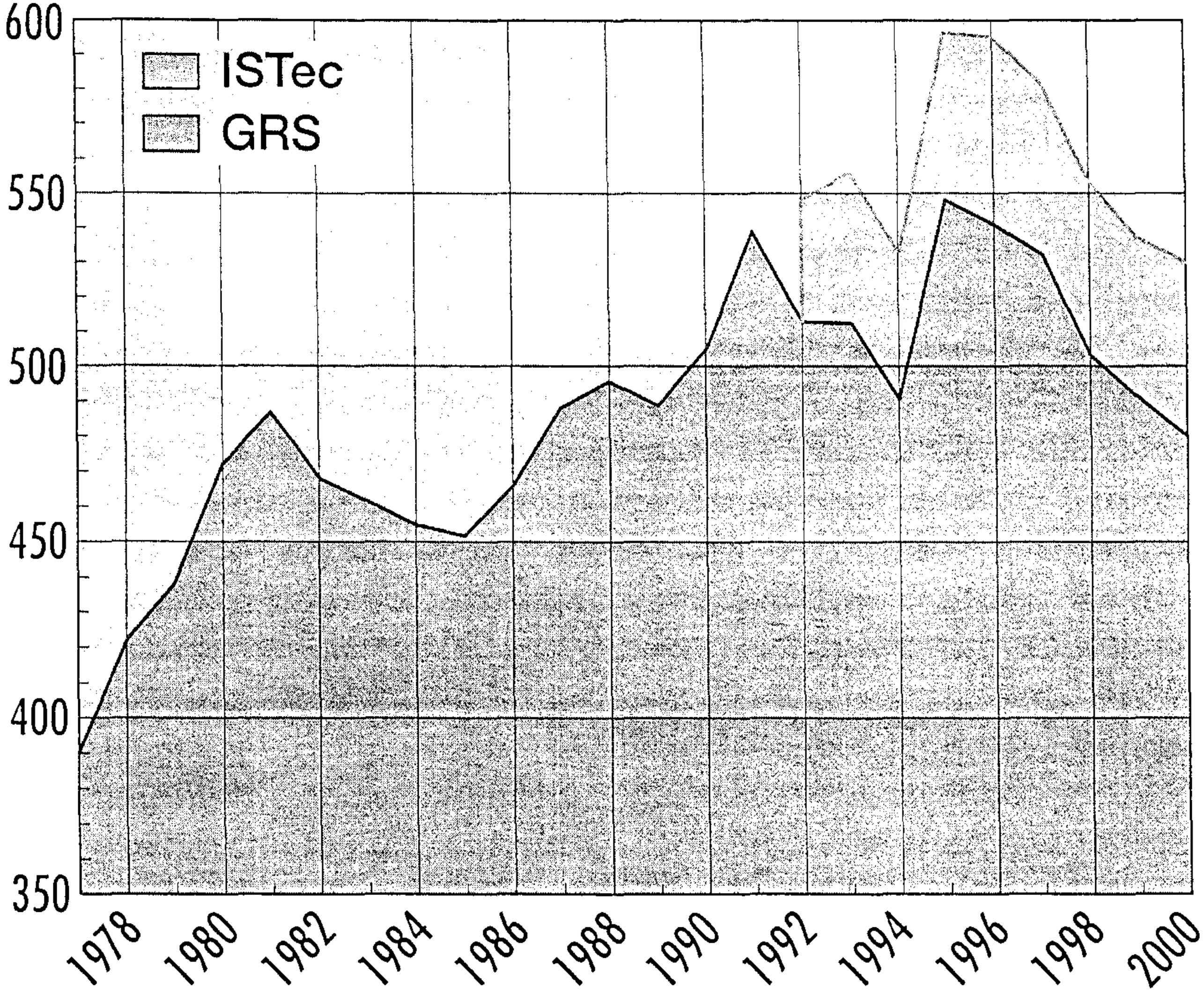
Volume of Contracts, Customers and Staff

In 1999, contracts to the amount of approx. DM 99 m were awarded to GRS for its scientific and technical work. This work was performed by more than 500 staff members, of which about 350 are scientists and engineers of such disciplines as:



- mechanical engineering
- electrical engineering
- physics
- nuclear engineering
- process engineering
- safety engineering
- civil engineering
- chemistry
- geochemistry
- geophysics
- mathematics
- informatics
- biology
- jurisprudence
- meteorology

Number of Staff Over the Years



In 2000, GRS had more than 480 staff members, of which about 280 are scientists and engineers of such disciplines as:

- mechanical engineering
- electrical engineering
- physics
- nuclear engineering
- process engineering
- safety engineering
- civil engineering
- chemistry
- geochemistry
- geophysics
- mathematics
- informatics
- biology
- jurisprudence
- meteorology

Major Activities
Research and Development

- Development and verification of scientific software for the simulation of nuclear power plant behaviour under accident conditions
- Development of advanced methods for probabilistic risk assessment
- Development of simulators for investigations into the behaviour of complex technical systems and their man-machine interfaces
- Methods for the early diagnosis of mechanical failures (e.g. vibration analysis, loose-part monitoring)
- Development of methods for the assessment of the uncertainties of computer predictions
- Methods to assure and assess the quality of safety-relevant software
- Computer models for the performance assessment of final repositories
- Experiments concerning geological and geo-technical influences final repository safety
- Development of advanced safety concepts
- Development of information and documentation systems

Major Activities
Analyses, Assessments and Expert Opinions

- Safety analyses and assessments of nuclear facilities
  (e. g. nuclear power plants, waste repositories)
- Probabilistic risk analyses of complex technical systems
- Analyses and assessments of transport safety
- Safety analyses of facilities for the disposal of chemotoxic waste
- Analyses and assessments of specific technical safety issues
  (e. g. incident control, reactor physics, material issues, fire protection, safety of digital I&C, software reliability)
- Analyses and assessments of nature conservation issues
  (e.g. impact of mining-related radioactive contamination, restoration of contaminated industrial sites, questions regarding occupational radiation exposure)
- Monitoring and evaluation of national and international operating experience from nuclear and other technical facilities
  (e. g. common-mode effects, human factor analyses, precursor studies)

Joint R&D Activities of GRS within the German Reactor Safety Research Programme

Universities

R&D contributions to resolve individual phe-nomena

Research Centres

Basic and develope-mental research for particular subjects, e.g.

FZK: Severe Accidents

FZJ: Passive Safety Systems

FZR: Reactor Dynamics

GRS

Basic and application-oriented R&D as a ba-sis for sound scienti-fic-technical safety statements

- Development and supply of codes and methods for safety evaluations of incidents and accidents
- Integral view and evaluation of R&D results

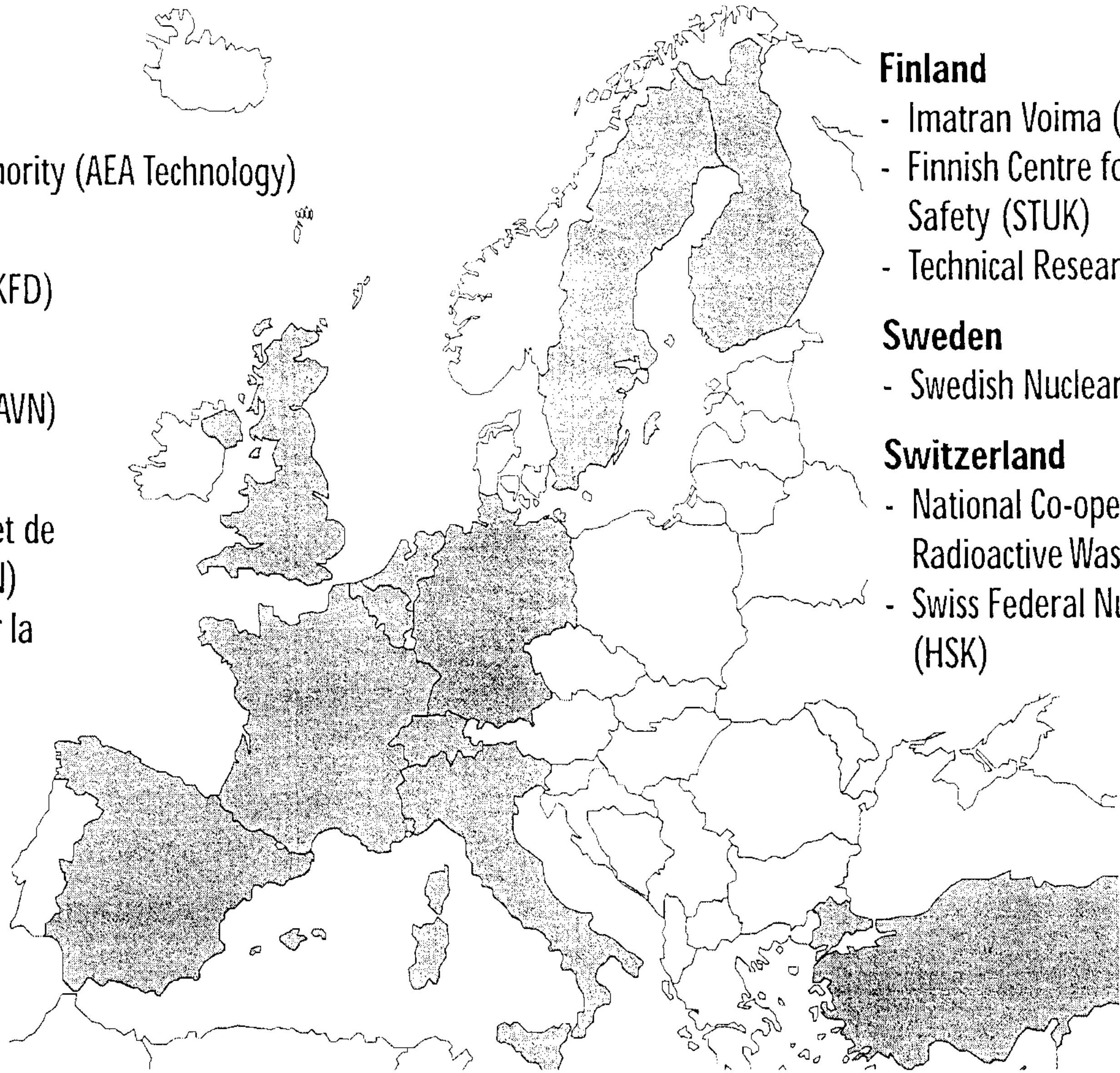
MPA, BAM, IzfP

Basic and application-oriented R&D in the area of component safety and material te-sting

Industry

Development and ope-ration-oriented rese-arch for design and optimisation of com-ponents and systems of reactor plants

GRS and its Partners in Western Europe



United Kingdom
- UK Atomic Energy Authority (AEA Technology)

Netherlands
- Kernfysischer Dienst (KFD)

Belgium
- AIB-Vincotte Nuclear (AVN)

France
- Institut de Protection et de Sûreté Nucléaire (IPSN)
- Agence Nationale pour la Gestion des Déchets Radioactifs (ANDRA)

Spain
- Empresa Nacional des Residuos Radioactivos SA (ENRESA)
- Consejo de Seguridad Nuclear (CSN)

Finland
- Imatran Voima (IVO)
- Finnish Centre for Radiation and Nuclear Safety (STUK)
- Technical Research Centre of Finland (VTT)

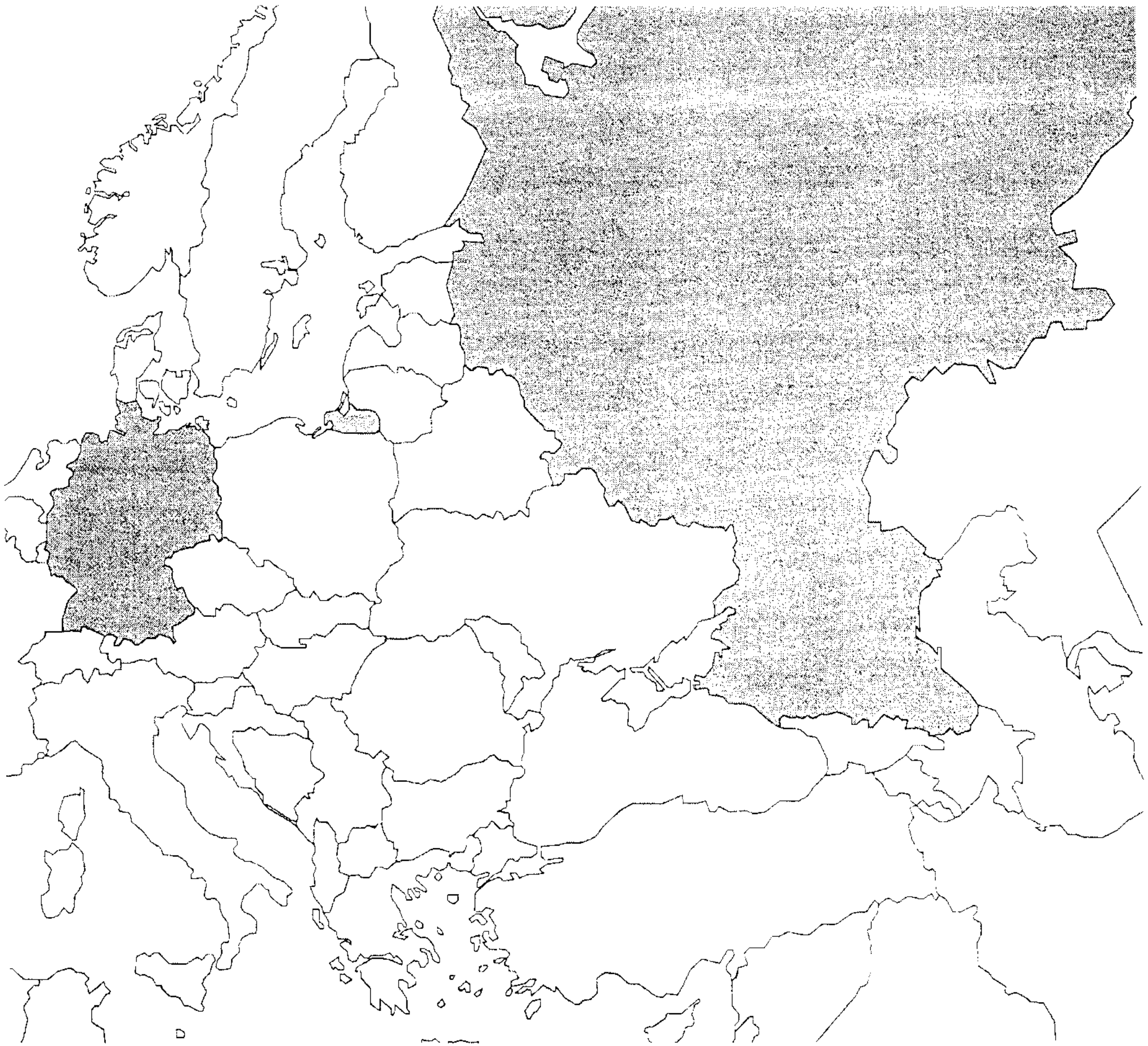
Sweden
- Swedish Nuclear Power Inspectorate (SKI)

Switzerland
- National Co-operative for the Storage of Radioactive Waste (NAGRA)
- Swiss Federal Nuclear Safety Inspectorate (HSK)

Italy
- Agenzia Nazionale per la Protezione dell' Ambiente (ANPA)

Turkey
- Türkiye Atom Enerjisi Kurumu (TAEK)

GRS and its Partners in Eastern Europe (1)



Russia

- Russian Research Centre Kurchatov Institute, Moscow
- Rosenergoatom (REA)
- Balakovskaya AES
- The Russian State Committee on Nuclear Safety and Radiation Protection
- Institute for Power Engineering NIKIET
- Experimental Design Bureau Gidropress (OKB GP), Podolsk
- Atomenergoproject AEP, Moscow
- Russian Academy of Sciences, Nuclear Safety Institute (IBRAE), Moscow
- Gosatomnadzor of Russia
- Scientific and Engineering Centre on Nuclear and Radiation Safety (SEC NRS, expert organisation of Gosatomnadzor of Russia)

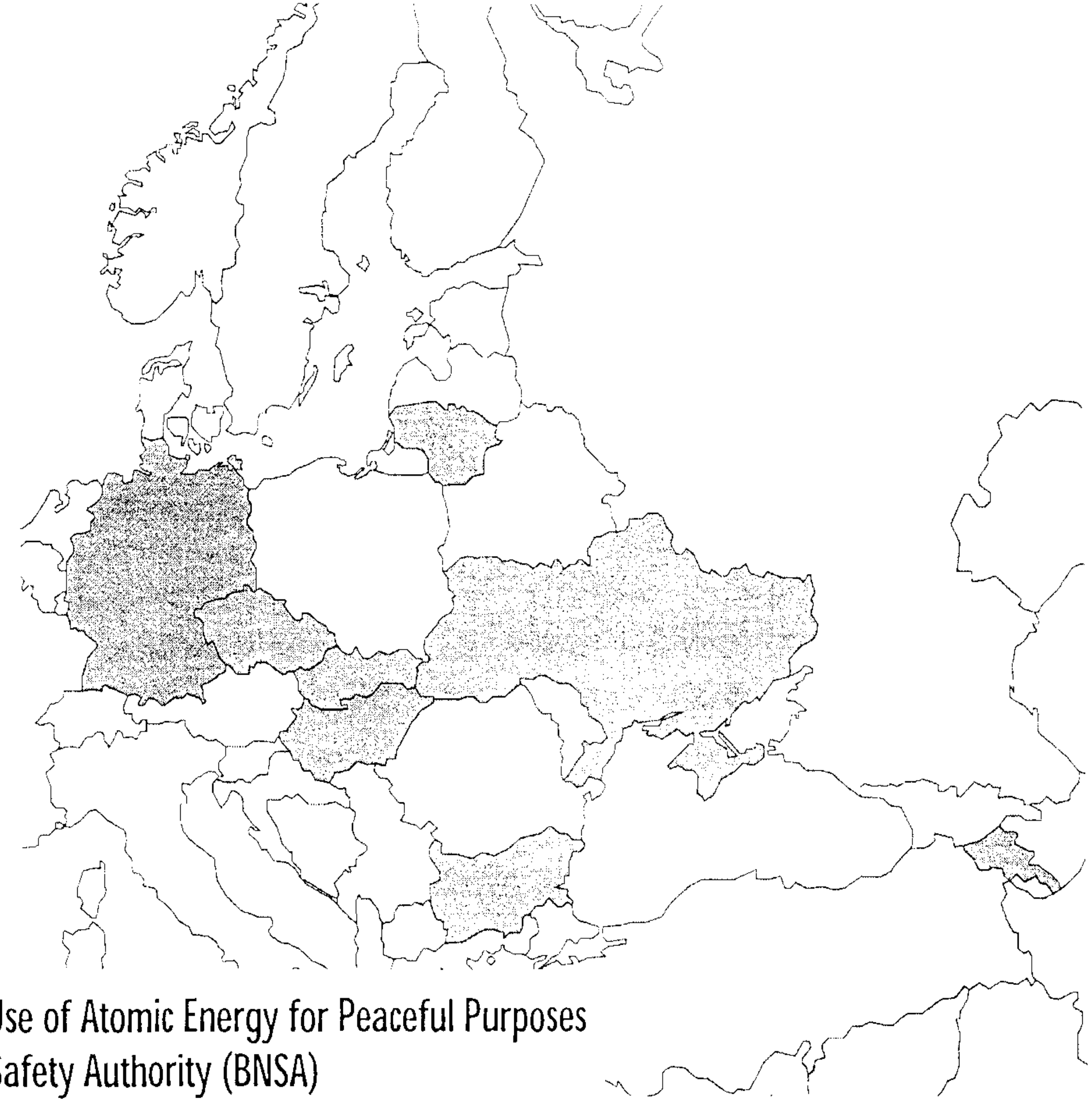
GRS and its Partners in Eastern Europe (2)

Czech Republic
- State Office for Nuclear Safety (SONS)
- Nuclear Research Institute Rez (NRI)

Slovak Republic
- Nuclear Regulatory Authority of the Slovak Republic

Hungary
- Hungarian Atomic Energy Commission
- Atomic Energy Research Institute (AERI)

Bulgaria
- Committee on the Use of Atomic Energy for Peaceful Purposes
- Bulgarian Nuclear Safety Authority (BNSA)



Lithuania
- Lithuanian Nuclear Power Safety Inspectorate (VATESI)
- Lithuanian Energy Institute (LEI)
- Nuclear Regulatory Authority (NRAUI)

Ukraine
- State Scientific-Technical Centre (SSTC, expert organisation of the NRA Ukraine)

Armenia
- Ministry of Energy and Fuel, Department "Armatomenergo"
- Armenian Nuclear Regulatory Authority

International Co-Operation of GRS in the Area of Reactor Safety Research

Contracts for bilateral co-operation between

France,
Great Britain,
USA,
Japan,
Republic of Korea

Bilateral research projects with CEEC&NIS within the frame of STC of BMWi

Russia,
Ukraine,
Czech Republic,
Slowak Republic,
Hungary, Bulgaria

Bilateral contracts for usage of GRS-Codes

with a total of 19 countries in Europe, Asia, America, South Africa

Multilateral contracts related to R&D projects of the 4th EU framework programme

11 EU member countries

Multilateral scientific technical co-operation within the frame of OECD-NEA

26 member countries in Europe, North America, Asia, Australia

The Regulatory System in Germany

Safety Aspects of HTR-technology

NRC visit in Germany

July 23rd, 2001

Edmund Kersting (GRS Köln)

Tel.: +49 221 2068 652
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Federal Republic of Germany

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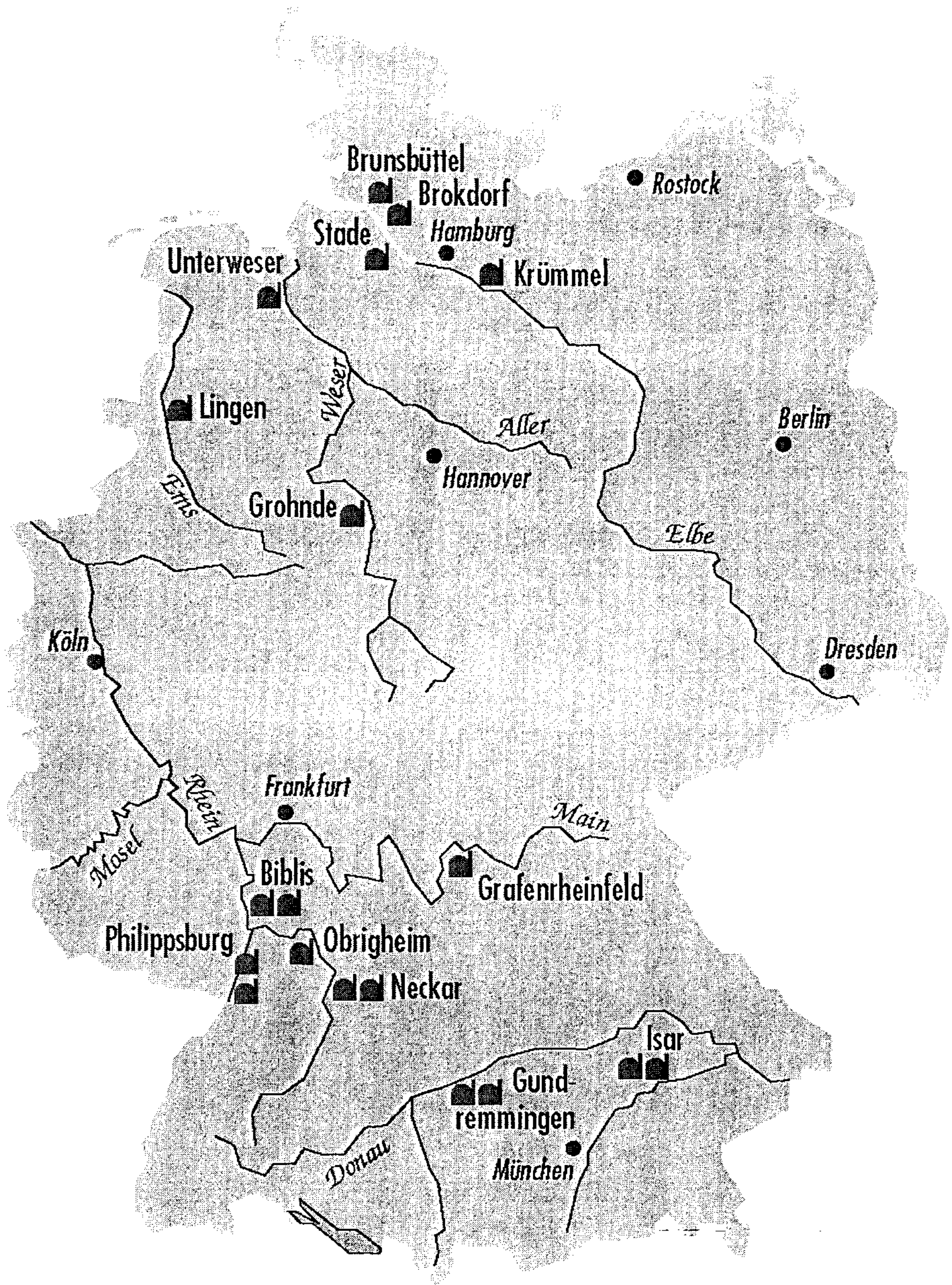

Nuclear Power Plants in Operation

Licensing Prerequisites for Nuclear Power Plants

- Amendment to the act (April 1994)
  - Additional provision against risks for the general public to limit consequences of severe accidents to the site - no need for major offsite emergency measures (Sec. 7, para. 2, No. 2a AtG)

- A Plant may only be licensed, if the following requirements are fulfilled
  - Reliability and qualifications of the applicant (Sec. 7, para. 2, No. 1 AtG)
  - Necessary knowledge of the operating personnel with respect to safe operation, possible hazards and protective actions (Sec. 7, para. 2, No. 2 AtG)
  - Necessary provisions against damage taken by design and operation according to the state of science and technology (Sec. 7, para. 2, No. 3 AtG)
  - Necessary financial provisions to cover all legal obligations for the compensation of damage (Sec. 7, para. 2, No. 4 AtG)
  - Necessary protection against disturbances or other 3rd party acts (Sec. 7, para. 2, No. 5 AtG)
  - Compatibility with overriding public interests, in particular to protect water, air and soil (Sec. 7, para. 2, No. 6 AtG)

The NRC Delegation

Howard Faulkner - Office of International Programs

Stuart Rubin - Advanced Reactors Group, RES

Donald Carlson - Advanced Reactors Group, RES

Amy Cubbage - Future Licensing Organization, NRR

Undine Shoop - Reactor Systems Branch, NRR

Alex Murray - Special Projects Branch, NMSS

Vanice Perin - Special Projects Branch, NMSS (NRC Observer)

Background and Purpose for the NRC Delegation Visit to Germany on the Safety Aspects of HTGR Technology

Stuart Rubin, RES

July 23, 2001

The NRC Delegation

Howard Faulkner - Office of International Programs

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Donald Carlson - Advanced Reactors Group, RES

Amy Cubbage - Future Licensing Organization, NRR

Undine Shoop - Reactor Systems Branch, NRR

Alex Murray - Special Projects Branch, NMSS

Vanice Perin - Special Projects Branch, NMSS (NRC Observer)

Core of group trying to form

Mission Purpose:

- Meet Leading HTGR Design and Technology Experts in the FRG and Learn of Their Areas of Special Expertise and Experiences

- Discuss and Obtain Information in the Many HTGR Design and Technology Areas That Are Important to the PBMR and GT-MHR Safety Review.

  Pebble/TRISO Fuel: Design, Manufacture, Performance, Qualification, etc.
  HTGR Heat Transfer and Fluid Flow Analysis, Methods, Testing, etc.
  Pebble Core Physics and Nuclear Design Analysis, Methods, Testing, etc.
  HTGR Nuclear Graphite, Properties, Behavior, etc.
  HTGR Accident Passive Decay Heat Removal Experiments, Analysis, etc.
  AVR & THTR Testing, Operating Experiences and Safety Lessons Learned
  Ex-Reactor Fuel Cycle Safety, Storage, Transportation
  Regulatory & Safety Assessments of FRG HTGRs, Regulations, Codes, etc.

- Obtain FRG Views and Information on the Key HTGR Design and Technology Safety Issues Which Should be Closely Examined in the PBMR and GT-MHR Safety Reviews.

UNITED STATES NUCLEAR REGULATORY COMMISSION

New Reactor Licensing

Amy Cubbage, NRR

July 23, 2001

NRR FUTURE LICENSING ACTIVITIES

- READINESS ASSESSMENT
- EARLY SITE PERMITS
- CONSTRUCTION
- PRE-APPLICATION REVIEWS
- REGULATORY INFRASTRUCTURE

FUTURE LICENSING AND INSPECTION READINESS ASSESSMENT

- Assess Postulated Scenarios, Review Durations, Resources
- Recommendations:
  - Staffing, Training, Contractor Support
  - Schedules
  - Rulemakings & Guidance Documents
- Complete Assessment September 2001

EARLY SITE PERMITS

- Early Site Permits (ESP)
  - Allows licensee to “bank” site for 10 - 20 years
- Regulations and Guidance
  - 10 CFR Part 52, Subpart A
  - Regulatory Guides, SRP, and Environmental SRP
- Current Schedule:
  - One ESP Application in 2002
  - Two ESP Applications in 2003
  - One ESP Application in 2004

CONSTRUCTION

- Construction Inspection Program Re-activation

  - Develop Guidance for Inspection of Critical Attributes
  - Initiate Development of Training for Inspection Staff

- Reactivation of Construction Permit (WNP-1)

PRE-APPLICATION REVIEWS

- AP-1000

  - Pre-Application Review Ongoing
  - Application for Design Certification Expected in 2002

- Pebble Bed Modular Reactor

  - Pre-Application Review Ongoing
  - Application for Early Site Permit Expected 2002
  - Application for Combined Operating License Expected 2003

- Gas Turbine - Modular Helium Reactor

  - Pre-application Expected to Begin September 2001

- International Reactor, Innovative and Secure

  - Pre-Application Expected to begin in 2002 or later

REGULATORY INFRASTRUCTURE

Current Activities:

- Rulemaking to Update 10 CFR Part 52
- Rulemaking on Alternative Site Reviews 10 CFR Parts 51 and 52
- Rulemaking on 10 CFR Part 51, Tables S3 and S4
- Financial-related regulations
- Proposed rulemaking website:

  http://ruleforum.llnl.gov

Future Activities:

- NEI white paper for generic regulatory framework
- NRC approaches for new technologies

NRC NEW REACTOR LICENSING WEBSITE

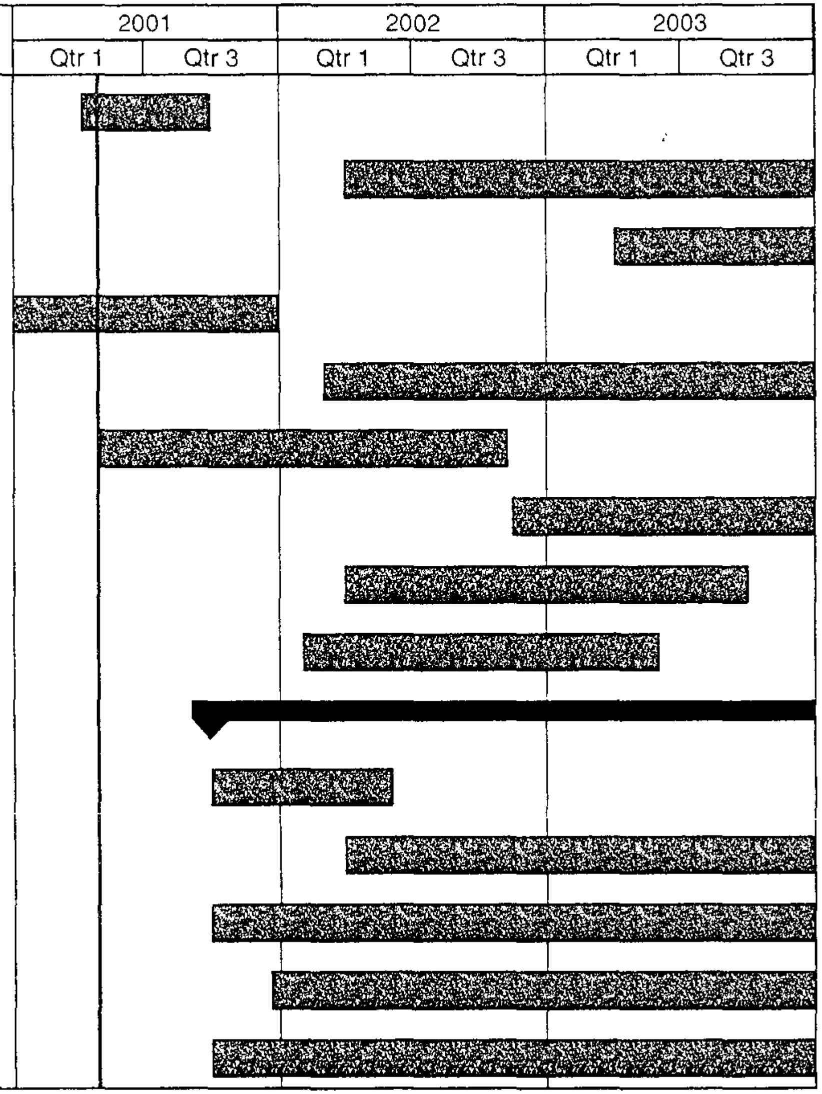
http://www.nrc.gov/NRC/REACTOR/FLO/index.htm

Path from www.nrc.gov:

- Nuclear Reactors
- What's New on this Page
- Future Reactor Licensing Activities

PROJECTED SCHEDULE

ID	Task Name
1	**Future Licensing and Inspection Readiness Assessment**
2	**Early Site Permit - 1st application**
3	**Early Site Permit - 2nd and 3rd applications**
4	**AP 1000 pre-application review - Phase 2**
5	**AP 1000 design certification review**
6	**PBMR pre-application review**
7	**PBMR combined license application (without ESP)**
8	**GT-MHR pre-application review**
9	**IRIS pre-application review**
10	**Regulatory Infrastructure**
11	Part 52 Rule
12	Clarify/modify environmental related regulations
13	Update regulatory and review guidance
14	New Plant Regulatory Framework
15	Review/clarify/modify financial related regulations



May 1, 2001

MEMORANDUM TO: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield

FROM: William D. Travers ***/RA/ by William F. Kane***
Executive Director for Operations

SUBJECT: STAFF READINESS FOR FUTURE LICENSING ACTIVITIES

This responds to the staff requirements memorandum (SRM) of February 13, 2001, in which the Commission directed the staff to "assess its technical, licensing, and inspection capabilities and identify enhancements, if any, that would be necessary to ensure that the agency can effectively carry out its responsibilities associated with an early site permit application, a license application, and the construction of a new nuclear power plant." In addition, the staff was directed to "critically assess the regulatory infrastructure supporting both Parts 50 and 52, and identify where enhancements, if any, are necessary." The Commission further directed the staff to integrate the tasks identified during this effort with the various related activities that are underway and provide the Commission with a schedule for completing these tasks, being thoughtful and judicious in committing resources.

Discussion

In the following discussion of the current activities and plans to address the Commission's SRM, each office's activities are addressed by topic. A preliminary schedule through 2003 for the items discussed in this paper is summarized in the attached figure. The Office of Nuclear Reactor Regulation (NRR) and the Office of Nuclear Regulatory Research (RES) are establishing organizational changes to prepare for these future licensing activities.

NRR is in the process of establishing the Future Licensing Organization (FLO), which will be responsible for coordinating the preparations for the review of new applications (i.e., early site permits, design certifications, and combined licenses), and to manage the AP1000

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Joseph M. Sebrosky, NRR, ADIP, FLO
301-415-1132

pre-application review and other activities listed below. FLO's near-term objectives are to identify (1) the steps that may need to be undertaken by the staff to prepare for licensing reviews, (2) the necessary resources and technical skills needed to perform these reviews, and (3) areas for improvements so that the reviews can be completed in a predictable time frame, based on past experience.

The establishment of FLO is a two-phase process. Initially, approximately 10 NRC staff members, some of whom have experience with standard and advanced reactor reviews and environmental reviews, have been temporarily assigned to (1) provide central points of contact within NRR for matters concerning future licensing efforts, (2) manage certain related initiatives currently underway (rulemaking activities, AP1000 pre-application review), (3) coordinate efforts to perform the readiness assessment, and (4) interact with Nuclear Energy Institute (NEI) working groups (e.g., NEI's siting task group), and other stakeholders. By the end of 2001, NRR plans to establish an organization that will continue these initial efforts and carry out the tasks established as a result of the readiness assessment.

RES is leading the staff's efforts with respect to the Department of Energy's (DOE) Generation IV program and initiatives on non-light-water-reactor (LWR) advanced designs. The goal of DOE's Generation IV program is to develop nuclear energy systems that would be available for worldwide deployment by 2030 that would have competitive economics, improved safety, improved environmental benefits, and enhanced proliferation resistance. The non-LWR advanced designs include modular high-temperature gas-cooled reactors (HTGRs) such as the Pebble Bed Modular Reactor (PBMR) being designed and developed in South Africa and the Gas Turbine-Modular Helium Reactor (GT-MHR) being designed and developed by General Atomics (GA). RES is in the process of establishing the Advanced Reactors Group (ARG) to serve as a focal point in RES for interactions with NRR, the Office of Nuclear Material Safety and Safeguards (NMSS), DOE, reactor designers, and potential applicants on matters related to advanced reactors. The ARG will be responsible for managing the advanced reactor technology, Generation IV, and non-LWR pre-application assessment work conducted by RES with the support of NRR and NMSS. The pre-application assessment work is also expected to provide input to the readiness assessment for Generation IV non-LWRs.

The Special Projects Branch in the Fuel Cycle Safety and Safeguards Division will serve as a central point of contact for coordination and review activities within NMSS. The primary role of NMSS will be to support future licensing efforts in areas of fuel fabrication, transportation, safeguards, and waste storage and disposal, with focus on any unique technical or regulatory issues associated with non-light-water-reactor advanced designs and increased enrichment levels.

Beyond the organizational infrastructure changes described above, a number of specific activities are already working or planned to begin as described below.

Future Licensing and Inspection Readiness Assessment

An early initiative has been to create the Future Licensing and Inspection Readiness Assessment (FLIRA) interoffice working group to address the ability of the NRC to support future application reviews under 10 CFR Parts 50 and 52. Approximately 11 NRC staff members from NRR, RES, NMSS, Office of Human Resources (HR), and the Office of the

General Counsel (OGC) will participate part time in the FLIRA working group. This group will operate under the direction of the director of the FLO.

The working group will provide an assessment of the following matters to the Commission in September 2001:

- postulated licensing scenarios for the future application reviews, durations of reviews (linked to milestones), and resource estimates in full time equivalent (FTE) and technical assistance support

- critical skills that must be available within the agency or that can be accessed through contractual agreements to perform these reviews

- necessary interfaces (intra- and inter-office, Advisory Committee on Reactor Safeguards (ACRS), stakeholders, Commission)

Early Site Permit Group

A group of experienced NRR staff has been identified to assess activities necessary to prepare for early site permit (ESP) applications (including pre-application inspections). This group will provide input to the FLIRA working group in a time frame consistent with their assessment schedule. One of the early issues that has been identified is access to key technical expertise that may reside within the agency. This access may be limited because of other competing priority projects (e.g., possible review of a license application for a high-level waste repository) and therefore, additional resources (including contractor support) may be needed.

In the interim, the staff has developed a scenario for receiving one early site permit in 2002, two in 2003, and one in 2004. This scenario is based on oral statements by industry representatives and staff assumptions. In advance of site approval applications, the staff expects to interact with prospective applicants to ensure that siting information has been developed with appropriate quality standards and representations of site conditions. This activity would also involve pre-application inspections of potential sites.

Pre-application and License Reviews

FLO is currently managing the Phase 2 portion of the Westinghouse AP1000 pre-application effort, in which the staff has been requested to provide feedback that will provide information to Westinghouse that will assist them in deciding whether to apply for design certification. The staff plans to issue its recommendation to the Commission on this portion of the review by the end of calendar year 2001.

A design certification application for the AP1000 is possible in 2002. The AP1000 assumption is based on a letter from Westinghouse dated December 12, 2000. Westinghouse stated that it would be prepared to submit its application in early calendar year 2002, but the date may be affected by the results of the AP1000 pre-application review. The preliminary schedule and rough resource estimates for this effort assume no hearing and minimal re-review of most of the AP600 design control document, on which the AP1000 design is based.

In a letter dated December 5, 2000, Exelon Generation Company (Exelon) requested early interactions with the staff on the feasibility of licensing the PBMR design in the United States. RES has taken the lead to develop a plan for pre-application activities on the PBMR, which is described in SECY-01-0070, "Plan for Pre-Application Activities on the Pebble Bed Modular Reactor (PBMR)," dated April 26, 2001.

Based on discussions at an April 30, 2001, meeting with Exelon, an application for a combined license for the PBMR is possible in late calendar year 2002. Exelon representatives also indicated that a design certification application for the PBMR may be submitted late in the combined license application phase. The staff has assumed that this could occur in 2005.

Westinghouse has recently requested a preliminary meeting with the staff to discuss the IRIS (International Reactor Innovative and Secure) design and plans for development testing (the meeting is planned for May 7, 2001, at NRC headquarters). Following this meeting, the staff should be in a better position to plan for future activities on IRIS. However, in the interim, the staff is assuming additional pre-application activities in the 2002 and 2003 time frame. A design certification application for Westinghouse's IRIS design is not expected for several years.

In a March 22, 2001, letter, General Atomics requested exploratory discussions with NRC on how to proceed with the licensing of its GT-MHR design. Because these discussions are in the early stages, the staff does not yet have detailed schedule information; however, based on statements made by GA representatives, pre-application activities may be requested as early as 2002.

Some of the pre-application and license reviews discussed in this section will need fuel cycle infrastructure, licensing, and certification review support. For example, the designs will have to be assessed for unique technical, environmental, and regulatory activities in the areas of fuel material enrichment and fabrication; transportation, storage, and safeguards of fresh and spent fuel; and waste disposal. This fuel cycle support would have to be in place before startup and operation of the plants.

Regulatory Infrastructure

Rulemaking efforts are currently underway to update 10 CFR Part 52 to address lessons learned from the experience of certifying three nuclear plant designs and clarify the processes for future application reviews. In a September 3, 1999, letter, the NRC solicited stakeholder comments and suggestions on a proposed update to 10 CFR Part 52. The staff received a response to this solicitation from NEI on April 3, 2001. In order to respond to these comments, the staff intends to delay its target date for the proposed rulemaking in this area from July 2001 to September 2001 to address the issues that were identified. Related rulemakings are also being planned, one of which is discussed in a December 18, 2000, memorandum to the Commission. In that memorandum, the staff provided a schedule for rulemaking associated with alternative site reviews. Additional rulemakings in the environmental area that are being considered include revisions to Tables S-3 and S-4 of 10 CFR Part 51 to address higher burnup fuel considerations and non-LWR advanced designs.

The staff will address the need to update regulatory and review guidance for future licensing applications, i.e., Standard Review Plans (SRPs), Regulatory Guides, and referenced codes and standards, and identify where enhancements are needed. The staff will have a better understanding of the extent of this effort and the necessary schedule and resources after the FLIRA working group assessment has been completed, although the staff does not expect this effort to begin until FY 2004.

During the 2001 Regulatory Information Conference, and at a public meeting with the staff on April 5, 2001, NEI proposed to replace deterministic regulations with risk-informed, performance-based regulations for future plants, where appropriate. NEI plans to submit a petition for an advance notice of proposed rulemaking (ANPR) for this initiative in December 2001. The NEI-proposed scope of work for the New Plant Regulatory Framework involves the actions needed to develop a conceptual framework of regulations, including general design criteria and general operating criteria. The scope does not include the work needed to develop and implement the associated infrastructure of design-specific regulatory guides and SRPs that would be needed to enable implementation of the framework for licensing purposes.

In an April 5, 2001, meeting with the staff, NEI also discussed the need to review issues such as antitrust reviews, decommissioning funding assurance, and financial qualification need to be reviewed because of the possibility of nuclear power plants being built as merchant plants. In addition, NEI suggested that Price-Anderson secondary protection, NRC rules governing annual fees, and operator staffing should be reviewed. The staff plans to begin preliminary work on this effort later this year.

The staff also needs to begin development of the regulatory infrastructure with respect to certain advanced technology assessment. Resources for code development have been included to provide the NRC with an independent capability to analyze the safety of non-LWR designs. This work would include code development (thermal-hydraulic, severe accident, fuels) and related testing to validate the codes. Additional advanced technology assessment in instrumentation and controls and human factors will begin. These efforts are being conducted by RES and are expected to begin in FY 2002.

In order to prepare for future applications NRR will reactiviate the construction inspection program revision effort suspended in 1994. This effort will include review and revisions of applicable inspection manual chapters and development of the associated inspection guidance and training for inspection of critical attributes of construction processes and activities.

Coordination and Communication with Stakeholders

The staff intends to communicate with stakeholders to ensure there is a clear understanding of upcoming activities related to future applications and to solicit stakeholder input. The staff is currently evaluating which communication tools should be used, and is considering the use of a public workshop to solicit stakeholder input and the creation of a web site to keep stakeholders informed of future licensing activities. The staff has had discussions with NEI, which is establishing four working groups to address current 10 CFR Part 52 licensing matters, early site permits, financial considerations, and the proposed new plant regulatory framework. NEI is being encouraged to provide information about new applications to support the staff's readiness reviews. Public meetings have been held with NEI and are being scheduled in the following months.

The staff will be meeting with the ACRS during its June workshop on advanced reactor designs, and plans to meet with the ACRS as necessary to support this effort. The staff has been holding technical and scheduling meetings with Westinghouse and Exelon, and will ask these potential applicants to provide input to support the related assessment activities. DOE has established the Near-Term Deployment Group to "identify technological and institutional gaps between the current state of the art and the necessary conditions to deploy new nuclear plants in the United States before 2010."

Resources

At the time the FY 2002 budget was developed, there was no indication of industry interest in future licensing activities and essentially no resources were included for these initiatives in the FY 2002 Budget Estimates and Performance Plan currently under consideration by Congress. Only recently has the industry shown significant interest in new construction. As a result the staff has estimated resources necessary to accomplish the activities identified in this memorandum for fiscal years 2001 and 2002. The staff is confident that it can complete the effort necessary for FY 2001 within the estimated FY 2001 resources. The estimate for FY 2002 is more uncertain, as the timing and pace of effort will be affected by the scope, timing and quality of submittals by applicants and industry organizations. In addition, technological or regulatory issues could arise that affect resource requirements and schedules. The staff will have a better understanding of resource needs for future licensing activities after the FLIRA working group completes its assessment.

Agency resources for FY 2001 are expected to be approximately 12 FTE and $270K in contractor support. These resources are necessary to perform the FLIRA working group readiness assessment, manage review initiatives currently underway or scheduled to begin during this period, and implement the PBMR review plan documented in SECY-01-0070. This effort will be accomplished by reprioritizing work using the planning, budgeting, and performance management (PBPM) process. Westinghouse and Exelon will be charged fees in accordance with 10 CFR 170 for NRC resources expended for the AP1000 and PBMR pre-application reviews, respectively.

The FY 2002 preliminary estimate of additional resources needed is approximately $15 - 18 million (including salary and benefits for approximately 50-60 FTE). This estimate is currently being reviewed and evaluated, in particular the estimated support cost needs. It includes direct and indirect costs for the program offices to accomplish the efforts described previously in this paper and supporting office costs such as legal advice, recruitment and retention incentives, training initiatives, security clearances, space alterations, and additional information technology equipment and support. The staff is in the process of developing the FY 2003 budget. The FY 2003 resource estimates for the future licensing activities will be included in the budget to be submitted to the Commission in June 2001.

While there is uncertainty associated with the specific activities that will be proposed by industry and the schedules on which they will be proposed, the staff is confident that sufficient future work will occur to warrant some hiring activities at the present time. In order to backfill for the staff members displaced by the future licensing activities through September 2001, the staff plans to begin the process of hiring additional staff. In the unlikely event that all industry initiatives associated with the licensing of future plants does not occur, the impact of the additional staff can be accommodated through normal attrition.

Agency Coordination

The Office of the Chief Financial Officer has reviewed this paper for resource implications and has no objections. The Office of the General Counsel has reviewed this paper and has no legal objections.

Conclusions:

The staff will be interacting with stakeholders in future review and licensing activities to ensure that it has a clear understanding of upcoming application plans. The staff will inform the Commission of the results of its readiness assessment and its recommendations when the assessment is completed in September 2001. At that time, the staff will recommend appropriate activities, including refined schedules and resource estimates, that are necessary to address these recommendations.

Attachment: As stated

cc: SECY
OGC
OPA
OCA
CFO
CIO

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Attachment: As stated

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DATE	04/ 18 /01	05/ 01 /01			

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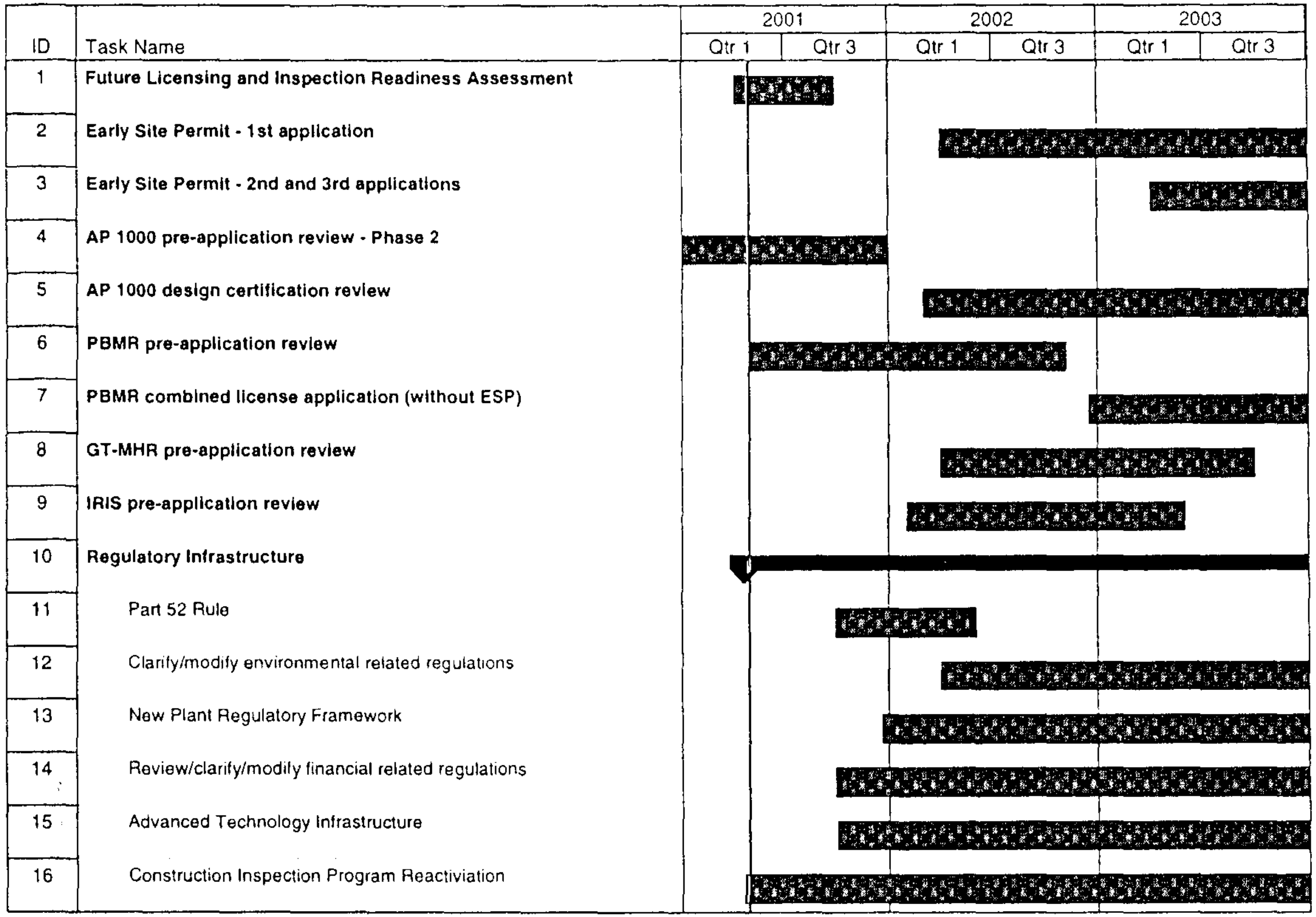
Figure - Estimated Future Licensing Timeline in Calendar Years



ATTACHMENT

POLICY ISSUE
(Notation Vote)

April 25, 2001 SECY-01-0070

FOR: The Commissioners

FROM: William D.Travers
Executive Director for Operations

SUBJECT: PLAN FOR PREAPPLICATION ACTIVITIES ON THE PEBBLE BED MODULAR REACTOR (PBMR)

PURPOSE:

To request Commission approval to proceed with preapplication activities on the PBMR.

BACKGROUND:

On November 14, 2000, representatives from Exelon Generation Company informally expressed their desire for early (preapplication) interactions with the staff directed toward establishing the feasibility of licensing a PBMR in the United States. Exelon indicated that these interactions would also help them determine the viability of the PBMR project. The PBMR is a modular high-temperature gas-cooled reactor (HTGR) being developed in the Republic of South Africa (RSA). Subsequently, in a letter dated December 5, 2000, Exelon formally requested such early interactions (Attachment 1). An initial meeting with Exelon was held on January 31, 2001, at NRC HQ to discuss the PBMR design and technology and the preapplication plans for the PBMR. Based upon the initial meeting, Exelon has indicated that it is their desire to have the preapplication phase completed by July 2002. Subsequently, the Commission issued a Staff Requirements Memorandum (SRM), dated February 13, 2001, which requested the staff to assess its readiness for new nuclear plant construction including the pebble bed reactor. A response to this SRM addressing the staff's readiness for licensing and the necessary changes to the licensing process is in preparation and will be forthcoming under separate cover.

CONTACT: Thomas L. King, RES
301-415-5790

DISCUSSION:

Consistent with my memorandum of November 14, 2000, on advanced reactors, RES has taken the lead (in coordination with NRR and NMSS) to develop a plan for preapplication activities on the PBMR. This plan is provided as Attachment 2 and it involves technology assessment, regulatory framework, and regulatory process assessment activities. It is estimated that approximately 18 months would be required to complete the plan.

As part of the technology assessment activities, the staff would familiarize itself with HTGR designs, technology, and safety issues generic to any HTGR design and identify NRC scientific and technology research needs. As part of the regulatory framework and regulatory process assessment activities, the staff would become familiar with the PBMR design, assess regulatory requirements applicable to the PBMR and Exelon's proposed approach to licensing, and identify key licensing issues and regulatory policy issues needing resolution. These activities would build upon the staff's previous domestic and international HTGR and fuel cycle experience and its advanced light water reactor (ALWR) design and regulatory reviews.

Commission approval is requested to begin the PBMR preapplication activities described in the plan. With respect to the PBMR, we believe that the plan is consistent with the Commission's SRM and is responsive to Exelon's request. However, certain activities will be completed later than Exelon has requested. For example, assuming a start date in late April 2001, completion of the preapplication activities would more likely be in Fall 2002 instead of July 2002 as requested by Exelon.

Early interactions with potential applicants are encouraged by and consistent with the Commission's policy statement on advanced reactors. Because of the active interest in the PBMR and requests of Exelon, this plan is being forwarded to the Commission in advance of the broader readiness assessment plans being developed in response to the SRM of February 13, 2001.

RESOURCES:

The activities, schedule, and resource needs are based upon the staff's previous experience with a pre-application review of a DOE-sponsored modular HTGR conducted in the late 1980s. The technology assessment, regulatory framework, and regulatory process assessment activities described in the attached plan would build upon that work and other previous advanced reactor work.

The U.S. Department of Energy (DOE) also considers an NRC safety and technology assessment of HTGRs, like the PBMR, as providing fundamental input for evaluating their advanced reactor program. Accordingly, DOE has recently inquired into the feasibility of NRC conducting such an assessment and has indicated that they would be willing to fund a portion of the work. DOE funding would support technology assessment and transfer activities that are generically applicable to HTGRs, including the PBMR. It is expected that most of the work for DOE would benefit the staff by developing the understanding, expertise and capabilities it would need to conduct future licensing reviews of HTGRs, including the PBMR. However, the DOE funding scope would not include safety and technology assessment work that is applicable only to the PBMR.

It is estimated that the total HTGR technology assessment and transfer activities to be funded by DOE would be approximately $1.4 million ($800K for contractor support and $600K for 3 FTE). DOE funding would begin in FY 2001, through a reimbursable agreement between DOE and NRC, if the Commission approves proceeding with this work. DOE has indicated that it would make $500K available ($300K for contractor support and $200K for 1 FTE) to initiate the work in FY 2001. DOE will provide the remainder of the funding totaling $500K for contract support and $400K for 2 FTE, subject to the availability of funds, in FY 2002.

The non-DOE funded work in support of PBMR preapplication activities in FY 2001 totals 1 FTE and will be realigned from within RES, NRR, and NMSS resources. The FY 2002 non-DOE funded work totals $200K and 3 FTE. Although the resources are not planned for in the FY 2002 budget, resources for FY 2002 and beyond will be addressed during the upcoming FY 2003 planning, budgeting, and performance monitoring (PBPM) process by RES, NRR, and NMSS.

Exelon would be assessed fees under 10 CFR Part 170, consistent with the Commission's 1995 fee policy for advanced reactor designs, for NRC's pre-application activities that are specific to the PBMR. Additionally, 10 CFR Part 170 fees would be assessed for the review of any license application for an HTGR, such as the PBMR.

COORDINATION:

The Office of the General Counsel has no legal objection to this paper. The Office of the Chief Financial Officer has reviewed this paper for resource implications and has no objections.

RECOMMENDATION:

We request that the Commission (1) approve proceeding with preapplication activities on the PBMR, including the DOE-sponsored HTGR technology assessment and transfer activities, described in Attachment 2, and (2) note that a meeting with Exelon has been scheduled for April 30, 2001.

William D. Travers
Executive Director
for Operations

Attachments: (1) December 5, 2000, Exelon letter
(2) Plan for Preapplication Activities on the PBMR

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200 Exelon Way
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Kennett Square PA 19348

Exelon Generation

Attachment 1

December 5th, 2000

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attn: Mr. William Travers

Subject: Pebble Bed Modular Reactor Review Requirements

Dear Mr. Travers

As you are aware, Corbin McNeill, the co-CEO of Exelon Corporation, has expressed interest in the Pebble Bed Modular Reactor (PBMR) technology. Exelon and several partners are currently trying to determine the technical, economic, and licensing feasibility of the PBMR design worldwide, including here in the United States.

The NRC's "Statement of Policy for Regulation of Advanced Nuclear Power Plants" (July 8, 1986) encourages the earliest possible interaction between the agency and applicants to provide licensing guidance. In line with this policy, Exelon and our partners request to formally engage with the NRC Staff for exploratory discussions on how we could most efficiently proceed with licensing the PBMR. We expect these discussions to help us determine if the PBMR is a viable project, in advance of our decision to be taken later. We would expect to identify review assumptions, policy issues to be considered, and to establish an estimate of cost and schedule for preliminary NRC PBMR technology education and review. It is our intent that subsequent phases could be identified during these initial discussions. We would like to target completion of a first meeting by January 12, 2001.

If you have any questions, please do not hesitate to contact us.

Very truly yours,

James A. Muntz

James A. Muntz
Vice President
Nuclear Projects

xc: C. A. McNeill, Jr.
E. F. Sproat, III
D. Nicholls (Eskom)
P. H. Readle (BNFL)
J. Colvin (NEI)
Honorable B. Richardson (DOE)
W. D. Magwood (DOE)

EDO --G20000567

Attachment 2

Plan for Preapplication Activities on the PBMR

INTRODUCTION

In a letter dated December 5, 2000, to William Travers, Exelon Generation Co. requested pre-application interactions with NRC directed toward assessing the viability of licensing a pebble bed modular reactor (PBMR) in the United States. The PBMR is a modular high-temperature gas-cooled reactor (HTGR), utilizing helium as the coolant and having online refueling capability, similar to HTGRs developed in Germany in the 1970s and 1980s. The current design is being developed in the Republic of South Africa (RSA) where a full-scale prototype PBMR module may be built and demonstrated. In addition to being a non-light water reactor, the design concept of the PBMR being developed in the RSA has other features which together are characteristic of (and unique to) modular high-temperature gas-cooled reactors. These characteristics make the PBMR approach to protecting public health and safety very different from reactor designs currently licensed in the United States. Chief among these features are:

- passive decay heat removal processes that are to be demonstrated under postulated accident conditions
- coated UO_2 fuel particles that are designed to contain the fission products and to be demonstrated at very high (accident) temperatures
- low power density (an order of magnitude below that for light water reactors (LWRs)) with large thermal capacity that are to be demonstrated to provide for slow transient behavior
- no conventional containment building
- a significantly reduced emergency planning zone (EPZ)
- multi-modular site concept with incremental power generation

Concurrently, DOE has informally inquired into the feasibility of the NRC staff conducting an independent assessment of HTGR technology and safety in order to assist in assessing their advanced reactors program. The proposed assessment (which would be conducted with DOE funding) would examine the design and the safety basis for HTGRs (including the PBMR) from a generic perspective. The assessment would include DOE support for the development of key analytical tools and NRC staff expertise in order for the NRC to conduct qualitative and quantitative safety assessments of HTGR reactors such as the PBMR. It is expected that most of the work for DOE would benefit the staff by developing the understanding, expertise and capabilities it would need to conduct a future licensing review of an HTGR, including the PBMR.

The Commission's Policy Statement on Advanced Reactors encourages early interactions on such advanced designs so as to facilitate the resolution of safety issues early in the design process. Additionally, many of the Commission's current reactor regulations are specific to LWRs and, as such, would not be applicable to the PBMR. Likewise, due to the different technology and approach to safety employed by the PBMR new requirements will be necessary in some areas. Accordingly, preapplication activities with DOE and Exelon are proposed to identify key safety and policy issues, propose a path to their resolution and establish a regulatory framework providing guidance on applicable requirements for the PBMR. It is proposed that these preapplication activities be structured to include: (1) a preliminary assessment of HTGR (including PBMR) technology and safety, and (2) a preliminary assessment of the regulatory framework and regulatory process for the PBMR. These preapplication activities would also help NRC to be prepared to review the PBMR in a timely fashion, if and when an actual application is received. The objectives of these activities would be as follows:

<u>HTGR Technology Assessment</u>:

- conduct early interactions with DOE on the NRC preliminary technology assessment scope and content to meet both NRC and DOE needs
- familiarize a nucleus of staff with the design and technology of HTGRs and their approaches to safety
- assess analytical tools and establish an independent staff capability to quantitatively assess the safety performance of HTGRs
- identify key generic technology issues with safety implications
- identify research needs to address these issues

PBMR Regulatory Framework and Process

- conduct early interactions with Exelon on its PBMR design and technology
- conduct early interactions with Exelon on its proposed licensing approach
- identify a resolution approach for key PBMR safety and technology issues
- evaluate the applicability of current regulatory criteria to the PBMR
- identify and solicit Commission guidance on PBMR policy issues
- support ongoing efforts to identify NRC infrastructure, research, and resource needs to support a PBMR licensing review, and reactor and fuel facility inspections.

HTGR technology issues and areas which are unique to the PBMR being developed in the RSA (and therefore not included in the scope of the DOE modular HTGR technology assessment scope) would be assessed directly through interactions with Exelon. These design-specific assessments will identify key issues with safety, technical and policy implications.

The outcomes of these technology assessment, regulatory framework and regulatory process assessment activities would be staff familiar with HTGRs, including the PBMR; identification of key safety and policy issues, and research needs; and preliminary guidance for the staff and potential applicants sufficient to establish the expectations for licensing. Documentation would include SECY papers to the Commission for information or for guidance on policy issues, letter reports to DOE and letters to Exelon providing feedback on technical and process issues (i.e., a preapplication safety evaluation report on the PBMR design itself would not be written).

PROPOSED PLAN

This paper describes a plan for preapplication activities, which involve technology, safety, regulatory framework and process assessment activities. These activities are directed toward HTGR technology transfer and preparing the agency for a possible application to license a HTGR, such as the PBMR, in the United States consistent with the above objectives. It is based upon experience in the past with preapplication reviews, including an earlier preapplication review of a DOE-sponsored modular HTGR, and would build upon that previous work. The plan describes preapplication activities that would be conducted over an approximately 18 month period and consists of technology assessment and transfer, and regulatory framework and process assessment elements described below. The plan also describes conduct of interactions and documentation and office coordination, resources and schedule.

Technology Assessment and Transfer

- familiarization with the design, safety, fuel cycle, and research issues via:
    - interaction with foreign partners and domestic organizations, including Exelon, with HTGR design, safety or operating experience
    - interaction with the RSA regulatory organization
- identification of reactor and materials safety and policy issues
- technology assessment, infrastructure and contractor support
- development and implementation of staff training

Familiarization with Design, Safety, Fuel Cycle, and Research Issues

Initial staff technology assessment and transfer efforts will be directed toward becoming familiar with HTGR (including PBMR) design, technology, safety and fuel cycle issues and research needs. This will be accomplished first through discussions and interactions with Exelon and others with PBMR and HTGR experience. An initial meeting was held with Exelon on January 31, 2001, at NRC-HQ to discuss the PBMR design, safety issues, and proposed Exelon schedule and approach for preapplication interactions related to technology assessment. Additional follow-on meetings will be scheduled on an as-needed basis to discuss specific topics and issues. In parallel with interactions with Exelon, the staff will contact others with HTGR and, to the extent possible, PBMR-specific experience to obtain their insights and views on HTGR and PBMR-specific safety issues and technology. These contacts are discussed below and include international as well as domestic organizations.

The NRC has a number of agreements with foreign countries that provide a mechanism to cooperate on a wide variety of safety matters. Some of our foreign partners have HTGR experience and some also have currently operating HTGRs (which utilize Helium coolant and coated particle fuel designs). Specifically, Germany has had many years experience with small (~45 MWt) and large (~750 MWt) HTGRs of pebble bed (i.e., coated particle/fuel sphere) design. Although the German HTGRs are no longer operating, their experience is relevant to the PBMR. Japan currently has an operating research HTGR (~30 MWt), although not of the pebble bed design. It does, however, utilize coated particle fuel and helium coolant and operates at high temperatures. China has recently begun initial startup of a small (~10 MWt) pebble bed research HTGR, from which experience should be obtained. In addition, they are developing a larger (200 MWt) modular pebble bed reactor design. The United Kingdom operates 14 advanced gas reactors (AGRs). Although they are different from HTGRs and the PBMR (e.g., the coolant is CO_2 and the fuel is not a coated particle design), they are graphite moderated and some experience may be relevant to HTGRs including the PBMR. Russia has had some HTGR development efforts in the past and is currently engaged in a joint effort with General Atomics (sponsored by DOE) to develop a modular HTGR (although not a pebble bed design) for plutonium (Pu) disposition. In addition, IAEA has some activities (in both the development and safety areas) looking at the design and safety of the PBMR. The NRC staff would also build upon and utilize their work in our activities. Finally, we would plan to discuss with the South African regulatory authorities their views on the PBMR design, safety issues, and research conducted (or to be conducted) to address the issues. In calendar year 2001, we would intend to arrange interactions with our international partners to discuss their experience with HTGRs and their views on safety issues.

Domestically, there remains some HTGR expertise, primarily at Los Alamos National Laboratory (LANL), Oak Ridge National Laboratory (ORNL) and General Atomics (GA). Preliminary discussions have been held with LANL and ORNL regarding the feasibility of drawing upon their expertise. Relevant experience at the other DOE labs will also be determined. Access to expertise at GA may be limited because GA is an NRC licensee and has indicated an interest in having preapplication interactions with NRC on their modular HTGR design. In addition, for the past several years the Massachusetts Institute of Technology has had an effort to design a modular pebble bed HTGR. Their experience will also be sought. Finally, previous NRC experience with earlier generation HTGRs (e.g., Ft. St. Vrain and the NRC review of a DOE- sponsored modular HTGR in the late 1980s and early 1990s) would be utilized to help identify safety and technology issues, research needs, and approaches to their resolution.

Identification of Safety and Policy Issues

HTGRs, such as the PBMR, involve characteristics that make their approach to protecting public health and safety very different from reactor designs currently licensed in the United States. For example, among the four basic layers of defense-in-depth for ensuring public health and safety against potential adverse consequences - prevention, protection, mitigation and emergency planning - modular HTGRs typically result in a shift in emphasis from mitigation features to highly reliable protection features. That is, HTGRs aim to achieve high reliability and protection through the use of fuel capable of withstanding high temperature, simple and passive decay heat removal and reactor shutdown processes as compared to high reliability through active standby engineered safety systems in LWR designs. Mitigation is provided through different concepts for fission product containment and through long response times of the reactor in the event of an accident. These and other differences between HTGRs and current generation LWRs are expected to lead to a number of safety, technology and policy issues. Issues such as high temperature materials performance; the qualification of accident analysis codes and methods; the qualification and performance of the coated particle/fuel spheres, the siting source terms, and the range of events that must be considered for design and siting purposes, are expected to be among the key safety, technology and policy issues that will need to be assessed.

Technology Assessment, Infrastructure and Contractor Support

Along with the identification of key technology, safety, and policy issues associated with HTGRs, including the PBMR, the staff will also identify the technology assessment and infrastructure needs to be ready to review an actual application. This will include needed in-house and contractor expertise, analytical tools, and the resources to obtain them. It is expected that the expertise needs will be in areas unique to HTGR technology and include:

- fuel design, fabrication and performance
- high-temperature materials performance
- helium turbine technology
- accident analysis
- HTGR risk analysis

A complete identification of infrastructure needs is, to some extent, dependent upon the identification and nature of the safety issues. However, in regard to analytical tools, it is important for the agency to have an independent capability to verify the plant response to accidents, particularly those related to loss of coolant, decay heat removal, and reactivity insertion. Such independent capability is valuable in providing a deeper understanding of plant behavior under a wide range of off-normal conditions, which can result in insights that contribute to the quality and thoroughness of the staff review and determine confidence in information provided by the applicant. Independent analyses have, in the past, led to the identification of significant advanced reactor safety issues that may otherwise have gone undetected (e.g., AP-600 fourth stage depressurization valve under-sizing). Currently, NRC does not maintain any analytical tools, data bases, or activities on HTGRs. The most recent efforts in this regard were approximately seven years ago when the agency had under way a preapplication review of a DOE-sponsored modular HTGR (MHTGR) design in accordance with the Commission's Advanced Reactor Policy Statement.

A draft preapplication safety evaluation on the MHTGR was issued in 1989 for comment (NUREG-1338); however, although a final NUREG was prepared in the early 1990s, it was never issued because DOE canceled the program. In developing NUREG-1338, the staff utilized contractor support and analytical tools from Oak Ridge National Laboratory (ORNL) and Brookhaven National Laboratory (BNL). Since that time, ORNL has remained active in the HTGR field and currently supports DOE-sponsored work on HTGRs for Pu disposition. Accordingly, there is expertise at ORNL (including analytical tools) that the agency could draw upon in the preapplication phase to assist the staff in the identification of issues and approaches for the preapplication review, as well as familiarizing the staff with the available analytical tools, their basis, and how to use them. In this regard, ORNL has available the GRSAC code (a three-dimensional thermal-hydraulic code with point kinetics reactor physics) that it is using in assisting DOE; this is an improved version of a code used in the staff's review of the DOE modular HTGR ten years ago. Other expertise and codes are also available and would be reviewed for applicability and possible use. Any needed improvements in the analytical tools will be identified and plans developed for their implementation.

Staff Training

One outcome of the technology assessment and transfer work would be the development of a small nucleus of staff familiar with HTGR technology and the unique attributes of the PBMR such that they can participate and facilitate an actual application review, if and when an actual application is received. This nucleus would include staff from RES, NRR, and NMSS.

To help achieve this outcome, development and implementation of a training program will also be included in the technology assessment and transfer work. The training program will consist of information on basic HTGR design, technology, safety features, operation, and experience. Contractor assistance will be used to develop and give the training, which will be targeted to be available in approximately one year. DOE has indicated that they would be willing to fund development and conduct of the program for their staff and the NRC staff.

Regulatory Framework and Process Assessment

- approach to licensing
- identification of regulatory requirements, safety and policy issues and a proposed approach for resolution

Approach to Licensing

Exelon has proposed an approach to licensing the PBMR in the United States. The approach includes building a single module in the United States under the combined license provision of 10 CFR Part 52 and, based upon that experience and the results of a test program using a prototype to be built in South Africa, subsequently certifying the design. Licensing and certification of a PBMR design may raise process questions regarding issues such as:

– with fuel quality an integral part of the safety case, should the fuel fabrication be tied to the design certification?
– is an application required for each module?
– is a decommissioning trust fund required for each module?
– application of Price-Anderson

Early interaction to identify and address such issues with Exelon would be part of the plan.

Regulatory Requirements, Safety and Policy Issues

An important output from the preapplication interactions with Exelon will be the identification of applicable requirements, safety and policy issues. This will involve looking at the requirements in 10 CFR (and their supporting regulatory guides) and identifying those that are unique to LWRs (and thus not applicable to the PBMR), as well as looking at the PBMR design and the technology and safety issues and identifying unique aspects that are not covered by current requirements.

The interactions with Exelon and our foreign partners, the domestic experience described above as well as the experience with the Ft. St. Vrain reactor, the review of a DOE-sponsored modular HTGR in the late 1980s, and the ALWR reviews would be utilized in reviewing the applicability of the requirements and in identifying unique issues associated with the PBMR.

It is expected that the technology, safety and regulatory assessments will lead to the identification of certain safety and policy issues that would need to be resolved in order to proceed with an actual licensing review. It is likely that the issues that stem from the preapplication activities will include:

- how to ensure fuel quality over the life of the plant
- acceptability of the use of fuel enrichments greater than 5%
- what accidents should the plant be designed for?
- containment vs. confinement
- an acceptable approach to the source term
- control room design and staffing
- transportation and on-site spent fuel storage
- extent of necessary prototype testing
- reduced emergency planning zone.

Policy issues would be provided to the Commission for guidance. A combination of traditional engineering and a risk-informed approach to addressing the issues would be utilized.

It is expected that an approach for resolving such safety and policy issues could be provided to the Commission in approximately 18 months. As an interim step, a preliminary set of the key safety and research issues associated with HTGRs including the PBMR would be provided to the Commission for information in approximately 12 months.

Conduct of Interactions and Documentation

Meetings with DOE and Exelon on specific topics related to HTGR and PBMR design, safety, technology, regulatory requirements and licensing process issues will be held. Following each meeting DOE and Exelon will be requested to document the information presented, any additional information identified by the staff, and their request for NRC feedback. On specific technical issues, requirements and process issues a response from the Director, RES, or the Executive Director for Operations, as appropriate, will be sent back to DOE and Exelon via letter. ACRS/ACNW and stakeholder input will be sought on technical and requirement issues prior to preparation of the EDO response. An approach for resolving policy issues would be provided to the Commission for guidance and would include consideration of ACRS/ACNW and stakeholder input. After Commission guidance is received, it would be provided to DOE/Exelon.

Coordination, Resources and Schedule

The preapplication activities will be a joint RES/NRR/NMSS effort. Although RES will have the lead, this effort will involve close coordination with and support from NRR and NMSS. The staff will also interact with ACRS and other stakeholders. Interoffice coordination and responsibilities will include:

- RES Role (overall lead for project)
  - organize, coordinate, conduct, and document meetings
  - organize and participate in ACRS presentations and stakeholder workshop
  - draft SECY papers, letter reports to DOE and letters to Exelon
  - preliminary identification of issues, research needs, applicable requirements, etc.

- NRR Role (overall lead for process issues related to the actual application)
  - participate with RES on preparing papers and participate in meetings, giving presentations and identifying technical issues
  - concur on correspondence to Exelon, DOE, ACRS, EDO, or the Commission

- NMSS Role (overall lead for fuel fabrication, transportation, waste and safeguards issues)
  - participate with RES on team preparing papers and participate in meetings, giving presentations, and identifying technical issues
  - concur on correspondence to Exelon, DOE, ACNW, EDO, or the Commission involving fuel fabrication, transportation, waste or safeguards issues

- OGC Role (overall advice on legal matters)

NRC staff work would focus on the review of applicable requirements and the identification of important accident scenarios, infrastructure, research, and resource needs. Contractor work would focus on review of HTGR analytical tools, training, and engineering analysis support.

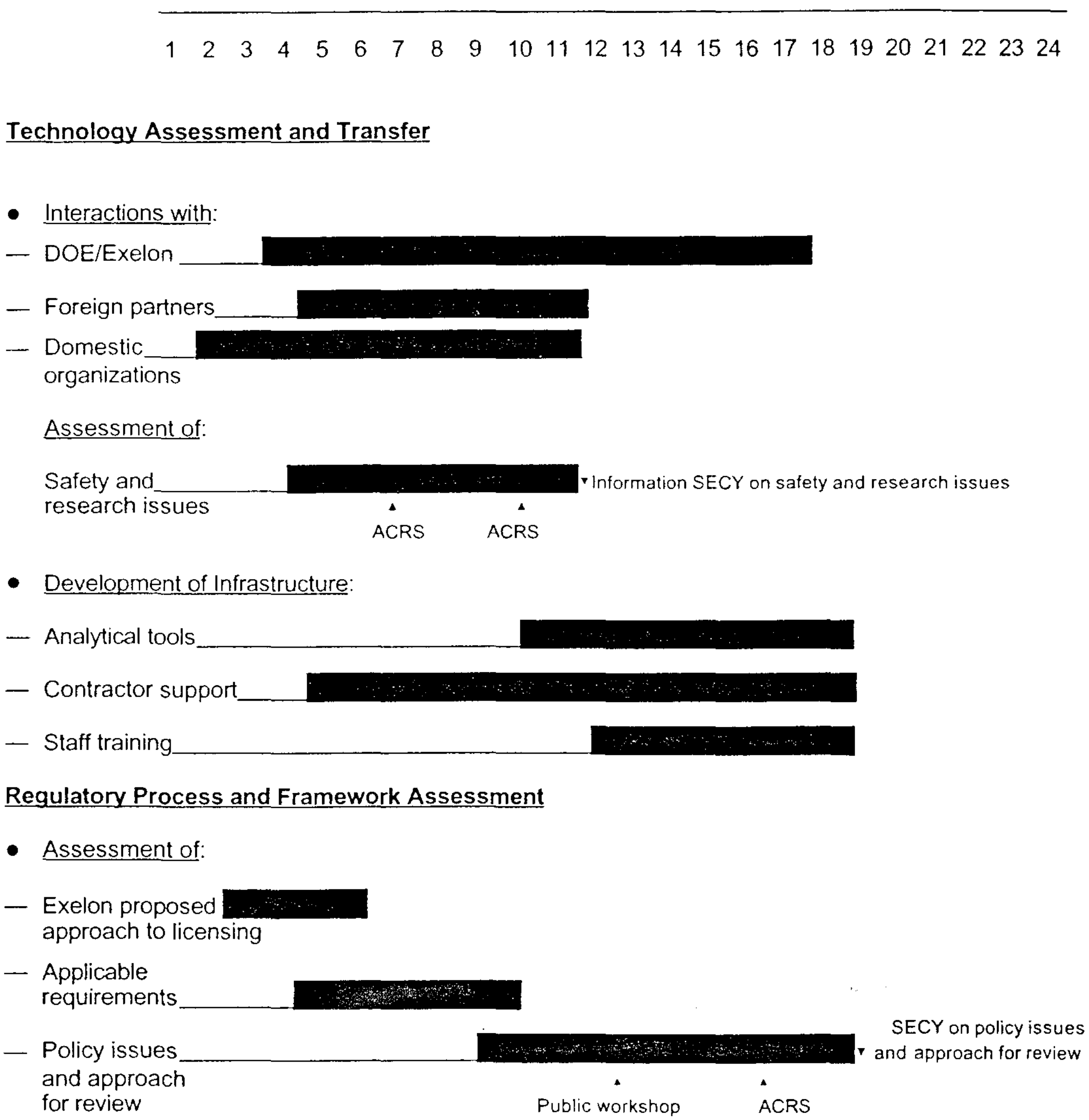
A preliminary schedule for the activities described above is shown in the attached figure. It is recognized that this schedule is dependent upon many factors, however, it represents the approximate time (18 months) necessary to accomplish the preapplication activities.

To accomplish the preapplication activities, it is expected that approximately 7 FTE will be necessary over the 18 month period. This will include 4 FTE in RES, 2 FTE in NRR and 1 FTE in NMSS. Also, it is estimated that $1000K will be needed over the 18-month period for contractor support in providing training, reviewing analytical tools and providing calculational assistance to the staff. DOE funds to cover the technology assessment and transfer activities are estimated to amount to $800K and 3 FTE over the 18-month period. Exelon would be assessed fees under 10 CFR 170 consistent with the Commission's 1995 fee policy for advanced reactor designs, for NRC's preapplication activities that are specific to the PBMR.

Preliminary Schedule for

PBMR Preparatory Activities

(in months)



June 19, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Andrew L. Bates, Acting Secretary **/RA/**

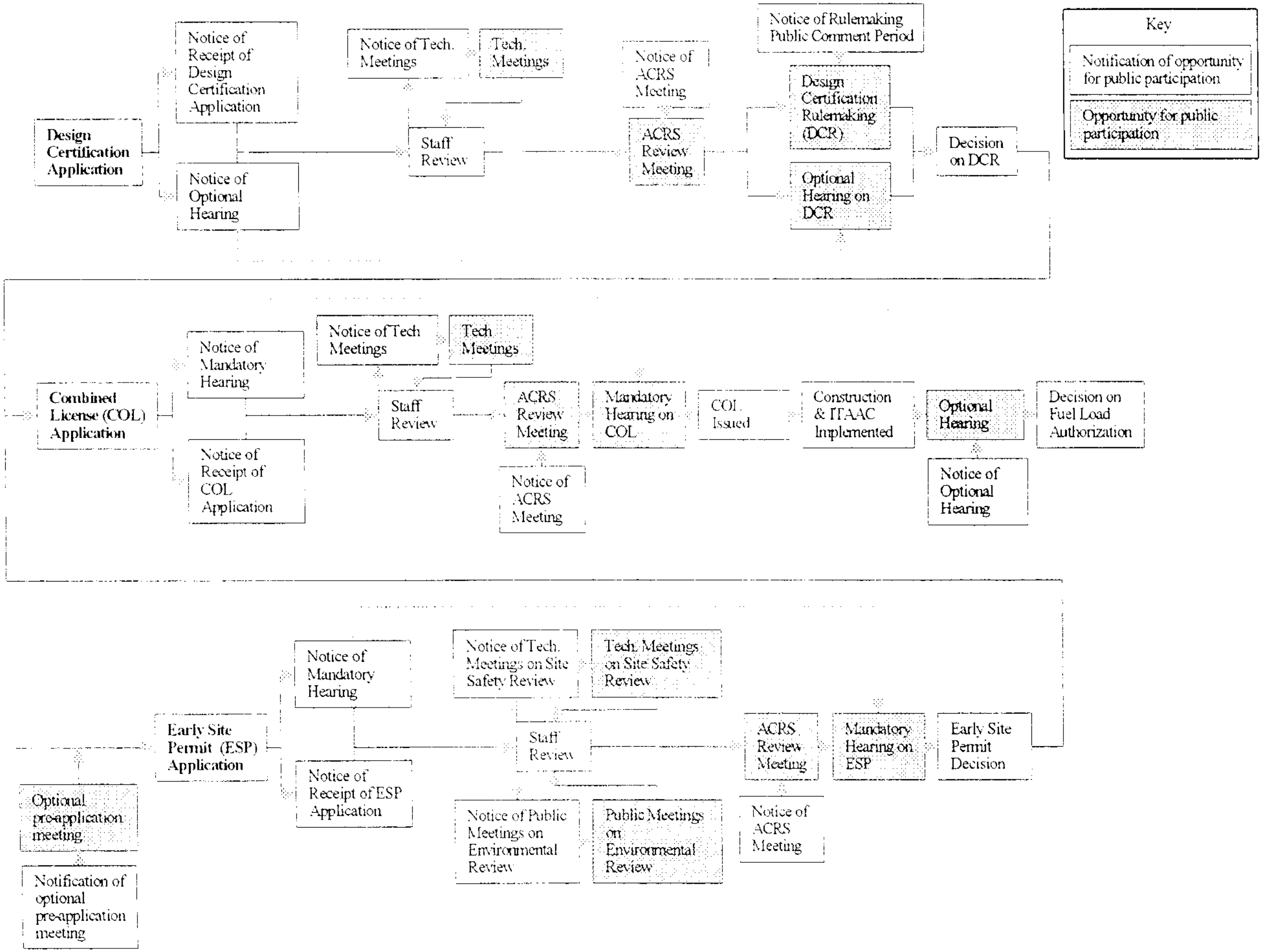
SUBJECT: STAFF REQUIREMENTS - SECY-01-0070 - PLAN FOR PREAPPLICATION ACTIVITIES ON THE PEBBLE BED MODULAR REACTOR (PBMR)

The Commission has approved the staff's recommendation to proceed with preapplication activities on the pebble bed modular reactor (PBMR), including the DOE-sponsored high-temperature gas-cooled reactor (HTGR) technology assessment and transfer activities, described in Attachment 2 of SECY-01-0070. The staff should keep the Commission informed of its progress on preapplication activities and promptly notify the Commission when policy issues are identified for resolution. The staff should also identify issues that the NRC is not able to resolve without Congressional action, and promptly bring these issues to the Commission's attention.

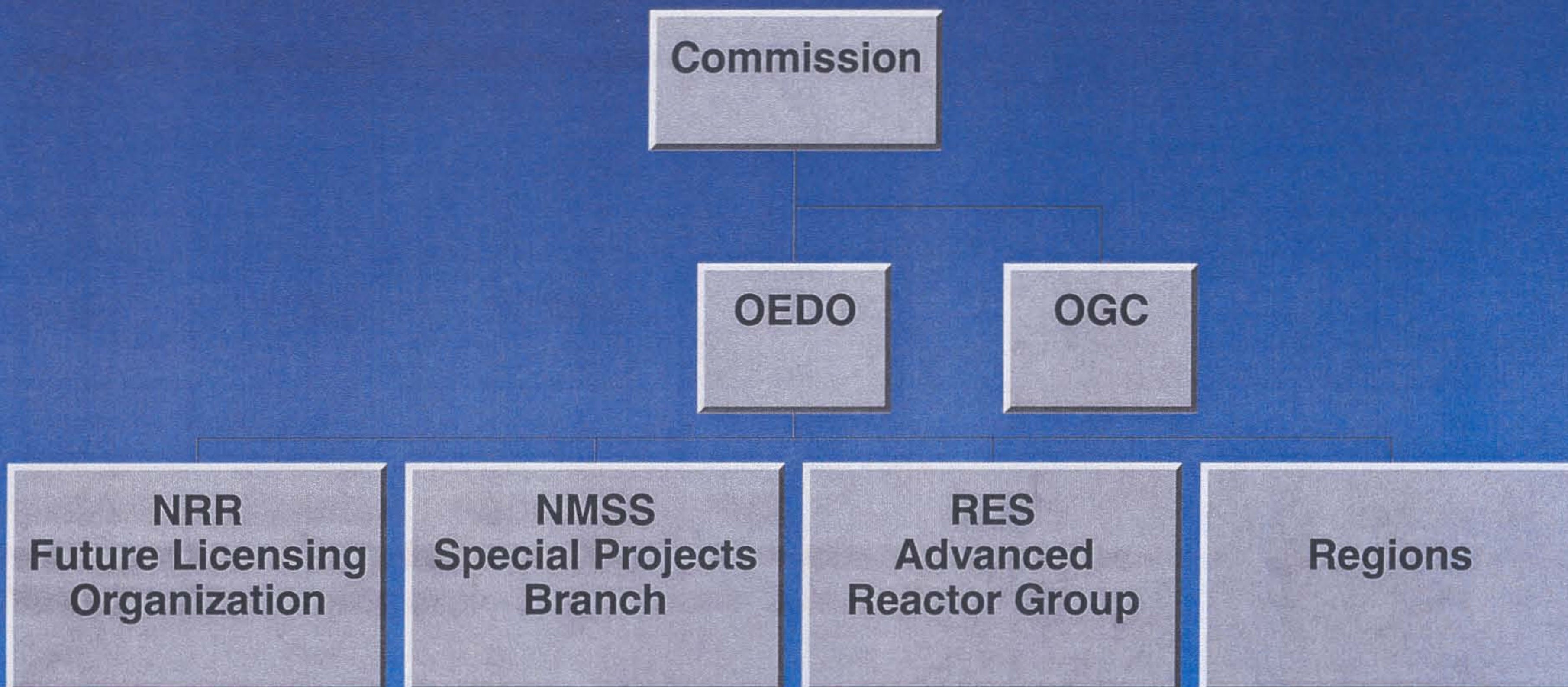
Experience with previous design certification reviews has shown that the staff evaluation of the applicant's testing programs can be a critical issue, particularly if the tests are conducted outside of the U.S. In light of the unique features of the PBMR and the possibility that testing could include work outside the U.S., the staff should include review of information relevant to the planned testing program within the scope of its preapplication activities.

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
OGC
CFO
OCA
OIG
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

10 CFR Part 52 Process



NRC Organization



U.S. Nuclear Regulatory Commission

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New Content–Who We Are

Who We Are

The U.S. Nuclear Regulatory Commission (NRC) is an independent agency established by the Energy Reorganization Act of 1974 to regulate civilian use of nuclear materials. Headed by a five-member Commission, the NRC has its headquarters in Rockville, Maryland, and four regional offices throughout the United States.

The NRC's regulatory mission covers three main areas:

- Reactors - Commercial reactors for generating electric power and nonpower reactors used for research, testing, and training
- Materials - Uses of nuclear materials in medical, industrial, and academic settings and facilities that produce nuclear fuel
- Waste - Transportation, storage, and disposal of nuclear materials and waste, and decommissioning of nuclear facilities from service

Our Mission

The NRC's mission is to regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, to promote the common defense and security, and to protect the environment.

NUREG-133

Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

P.M. Williams, T.L. King, J.N. Wilson

D.E. Carlson

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

February 26, 1996

Mr. Donald E. Erb, Acting Project Manager
Office of Nuclear Energy
U.S. Department of Energy
Washington, D.C. 20585

SUBJECT: DRAFT COPY OF PREAPPLICATION SAFETY EVALUATION REPORT (PSER) ON THE MODULAR HIGH-TEMPERATURE GAS-COOLED REACTOR (MHTGR)

Dear Mr. Erb:

Enclosed is a draft of the final PSER which documents the staff's preapplication review of the MHTGR design. In accordance with your letter of July 17, 1995, the PSER does not contain Applied Technology information and, thus, does not carry that designation.

The enclosed PSER contains minor revisions to the draft PSER that was submitted to you in my letter of June 30, 1995, for a review for Applied Technology information. The revisions are based on an internal review after the June 30, 1995, letter. The enclosed PSER was submitted to the Commission in SECY-95-299, "Issuance of the Draft of the Final Preapplication Safety Evaluation Report (PSER) for the Modular High Temperature Gas-Cooled Reactor (MHTGR)," on December 19, 1995.

The enclosed PSER is (1) Volume 1, which contains the documentation of the staff's preapplication review of the MHTGR design and the conclusions of the staff on the design from this review, and (2) Volume 2, which contains the appendices, without copies of the documents that are referenced in the PSER and available through the NRC Public Document Room. These documents, which would be in Appendices C through J of Volume 2, are not essential for the staff's discussion of MHTGR licensability and policy issues and are, therefore, not included in the enclosed PSER to reduce its size. These documents were provided to you in our letter of June 30, 1995, and will be provided in the final PSER.

Please provide comments to the Nuclear Regulatory Commission (NRC) on the technical discussions and conclusions in the enclosed PSER within 6 weeks of the receipt of this letter. These comments will be considered for inclusion into the PSER before it is submitted to the Commission as the final PSER for the staff's preapplication review of the MHTGR. Section 4.2.9 of the PSER states that the Department of Energy (DOE) should provide in its design approval application for the MHTGR the basis for designating design information as being required to be withheld from the public. That PSER section states further that DOE should include in the application an explanation as to how information designated as Applied Technology falls within the scope of the Atomic Energy Act. In your comments, you are also requested to address the discussion on the Applied Technology designation in Sections 1.8 and 4.2.9 of the enclosed PSER.

Visit of the NRC Delegation to Germany

SAFETY ASPECTS OF HTR TECHNOLOGY

23 to 26 July 2001

Contributions to be presented by

TÜV Hannover/Sachsen-Anhalt e.V.

Topics - 1

Monday, 23 July 2001

Overview on Safety Assessment of the HTR Module in Germany

- The contribution of the TÜV to technical safety in Germany
- The role of TÜV Hannover/Sachsen-Anhalt e.V. in nuclear technology
- The licensing process for the German HTR-2 NPP

Topics - 2

Thursday, 26 July 2001

Safety Assessment of the HTR Module in Germany

- The task as defined in the contracts
- Overview of the plant concept
- The methodology applied in safety assessment of the HTR-2 NPP
- The most important results

Origin of TÜV in Germany

- TÜV founded in the second half of nineteenth century by industrial companies operating steam vessels and engines
- Aim: Reduction of steam vessel and engine failures
- Status: Independent and neutral association; regionalized structure

⇨ Effect: Distinct reduction of steam vessel and engine failures

⇨ Consequence: Enlargement of TÜV tasks

TÜV today - 1

- The TÜV are a service companies engaged in safety assessment and inspections of technical equipment
- Task: To protect people and the environment from the hazards caused by erection and operation of technical equipment
- TÜV are free from manufacturers‘, licensees‘ and buyers‘inter ests; they are independent and self-governing institutions of trade and industry
- Statute: Expertise - Independence - Neutrality

TÜV today - 2

- Field of activities: TÜV are service companies carrying out
  - ⇨ sovereign tasks for the authorities
  - ⇨ consultation tasks for the authorities
  - ⇨ consultation tasks for industrial companies

free market activities

- Spectrum of activities:
  - ⇨ Car inspections
  - ⇨ Safety of conventional plants
  - ⇨ Biotechnology
  - ⇨ Environmental protection
  - ⇨ Quality management systems
  - ⇨ Material investigations
  - ⇨ Nuclear safety

TÜV NORD GRUPPE

- TÜV NORD GRUPPE:
  Merger of TÜV Hannover/Sachsen-Anhalt e.V. and
  TÜV Nord e.V.

- Turnover: 800 Million DM per year

- 4000 employees

- Offices in 8 federal states in Germany:
  Lower Saxony, Mecklenburg-Vorpommern,
  North-Rhine Westphalia, Sachsen-Anhalt, Schleswig-Holstein,
  Berlin, Bremen and Hamburg

The Division Energy and Systems Technology

Founded:	1957
Staff today:	170 scientists and engineers with expertise in: civil engineering, electrical engineering, process engineering, mechanical engineering, nuclear physics, chemistry, biology
Organization form:	Matrix structure: 4 specialists‘ departments, 1 project department, efficient project management, strict separation of responsibilities, but team work

Date:
01/07/2001

TÜV Hannover/Sachsen-Anhalt e.V.

Division Energy and Systems Technology

TÜV NORD GRUPPE

ET – Energy and Systems Technology
Dipl.-Ing. Helmers
Assistant Manager: Dr. Nitzki

- Administration and Computer Application
- Quality Management – Dr. Plagemann

ETP – Projects
Dr. Knieper

Nuclear Power Plants, Material Test Reactors

- KKE, KWL, Biblis – Dipl.-Ing. Liebing, Dr. Bieling, Dr. Hergemöller
- KWG – Dipl.-Ing. Gutheil, Dipl.-Ing. Steinle
- KWW – Dipl.-Ing. Haas
- TRIGA, SUR 100, FMRB – Dr. Hergemöller

Nuclear Fuel Cycle

- BFL – Dipl.-Ing. Degener
- TBLG – Dr. Rabe
- TBLA – Dr. Müller, Dr. Rabe, Dipl.-Ing. Schulz
- SZL – Dr. Rabe
- SZN/SZS Spent fuel casks – Dr. Müller, Dr. Rabe, Dr. Bieling, Wetzel, Dr. Stein
- PKA – Dr. Müller, Dipl.-Ing. Schulz
- ALG – Dr. Stolpmann
- ZLN – Dipl.-Ing. Degener
- Radioactive Waste Management – Dr. Wehmeier

Non-Nuclear Projects, Innovative Projects – Dr. Stolpmann

ETB – Operation and Systems Analysis
Dipl.-Ing. Junge

- Systems Analysis – Dipl.-Phys. Krüger
- Power Supply and Instrumentation & Control – Dr. Grondey
- Transportation and Handling – Dipl.-Ing. Vollmer
- Services in Biotechnology – Dr. Pohl

ETK – Components and Materials Technology
Dr. Plagemann

- Component Safety – Dr. Plagemann
- Strength Analysis – Dipl.-Ing. Bode
- Non-Destructive Testing – Dr. Krüger

ETS – Radiation Protection
Dr. Drotleff

- Activity Release and Radioecology – Dr. Rinkleff
- Radiation Protection of Workers – Dr. Börchers
- Radiation Protection Measurements – Dipl.-Ing. Mende
- TÜV NORD RÖNTGEN-TECHNIK – Dipl.-Ing. Kreienfeld

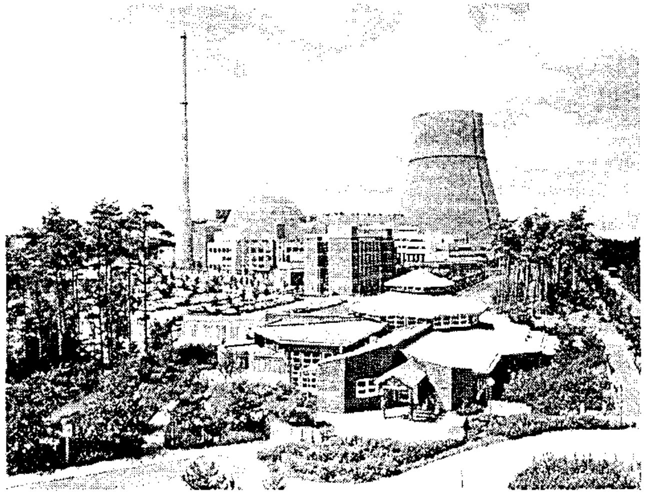
ETR – Nuclear Core Design and Waste Management
Dr. Nitzki

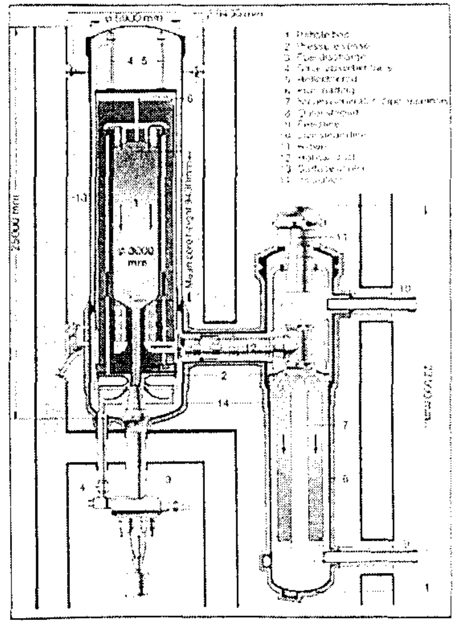
- Reactor Physics and Fluid Dynamics – Dr. Vogel
- Nuclear Fuel – Dr. Märtens
- Radioactive Waste Management – Dr. Kröger

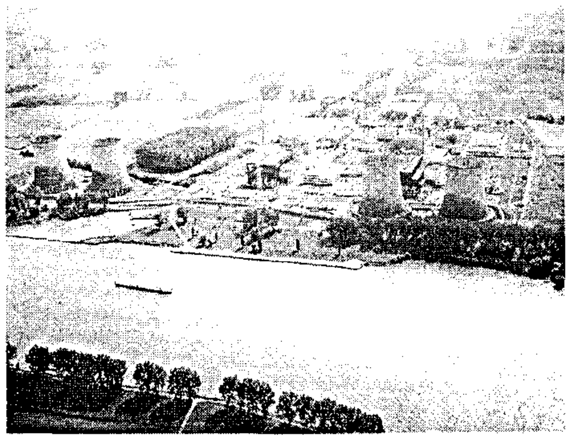
Main Projects of the Division

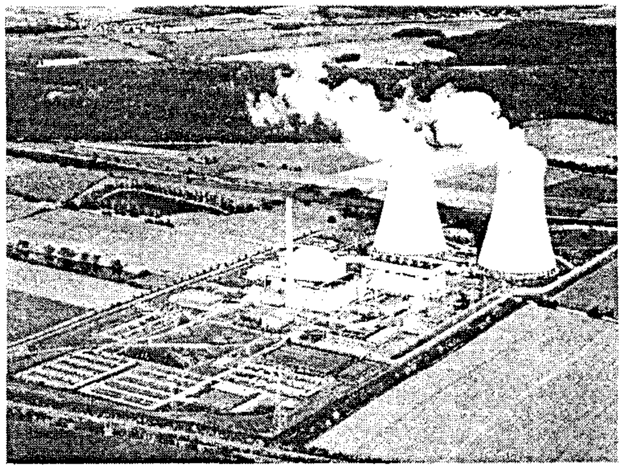
- Nuclear Power Plants
  - ⇨ Operating NPPs: Grohnde, Emsland, Biblis A and B
  - ⇨ Decommissioning: Würgassen, Lingen
- Fuel Fabrication: Fuel fabrication plant of Fragema/Siemens at Lingen (former Exxon Plant)
- Spent Fuel Storage: Ahaus, Gorleben, Greifswald and various On-Site Storage Facilities
- Interim and Final Waste Storage: Gorleben, Greifswald, Konrad
- Compliance of waste properties and acceptance criteria
- Compliance of spent fuel transportation casks and transportation requirements
- Others, e.g. German HTR-2 Modular Reactor and Pebble Bed Modular Reactor of ESKOM, South Africa

Main Projects: Nuclear Power Plants

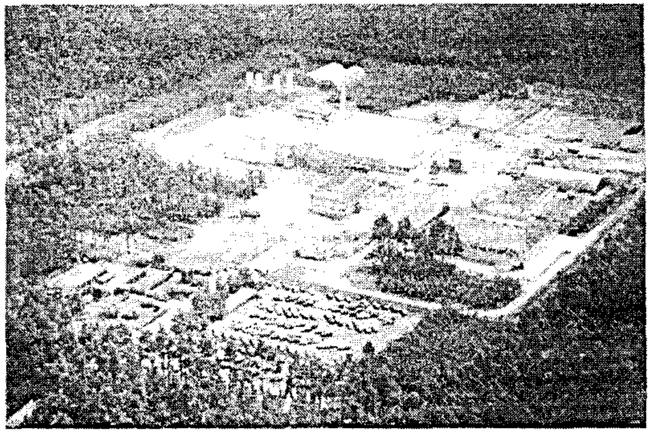


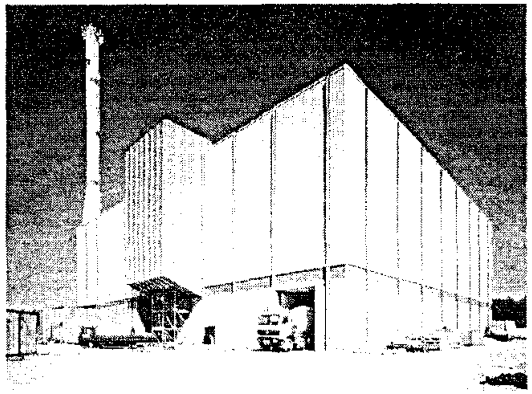


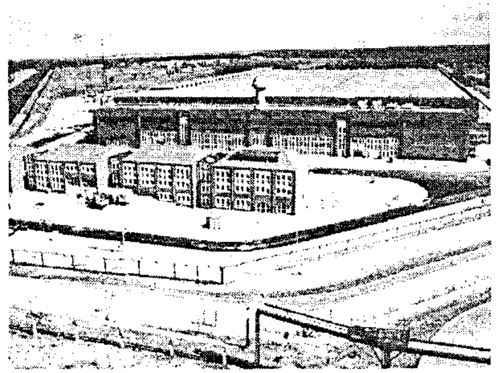


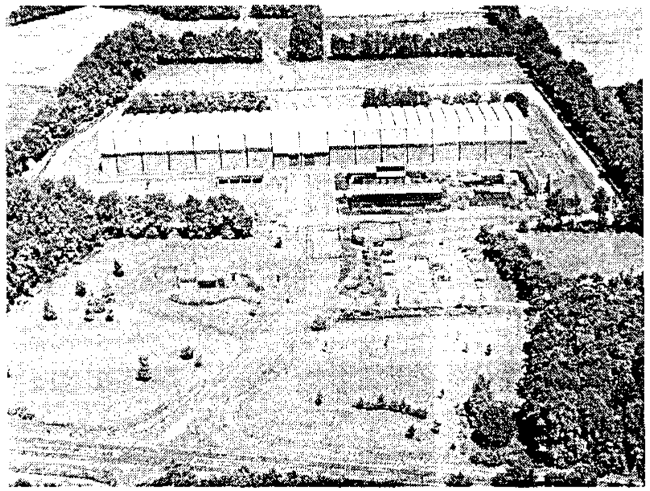


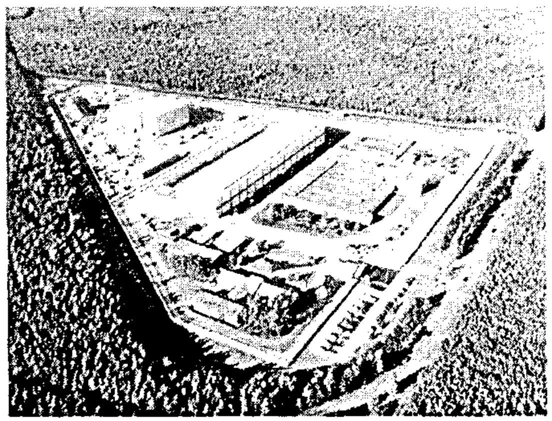
Main Projects: Nuclear Fuel Cycle

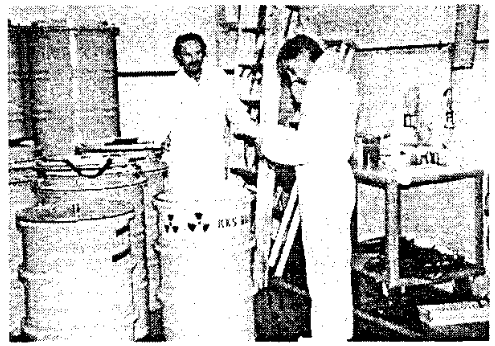






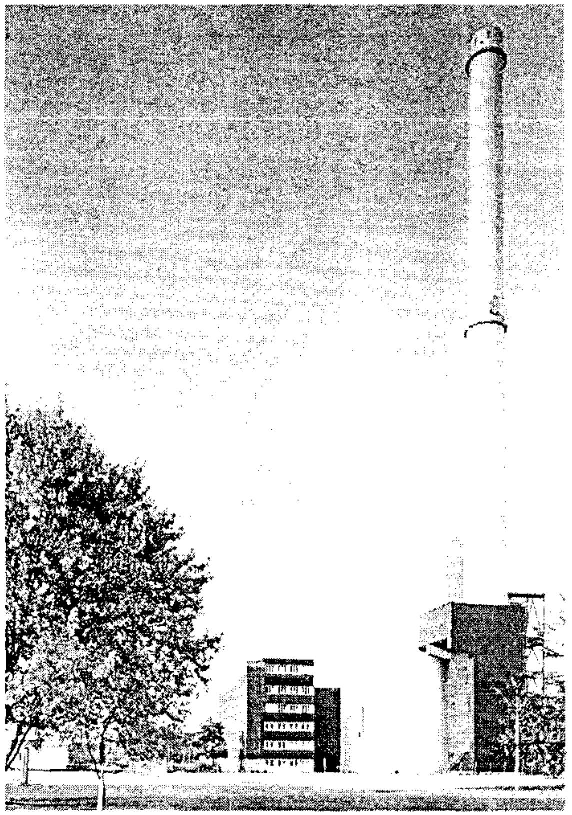


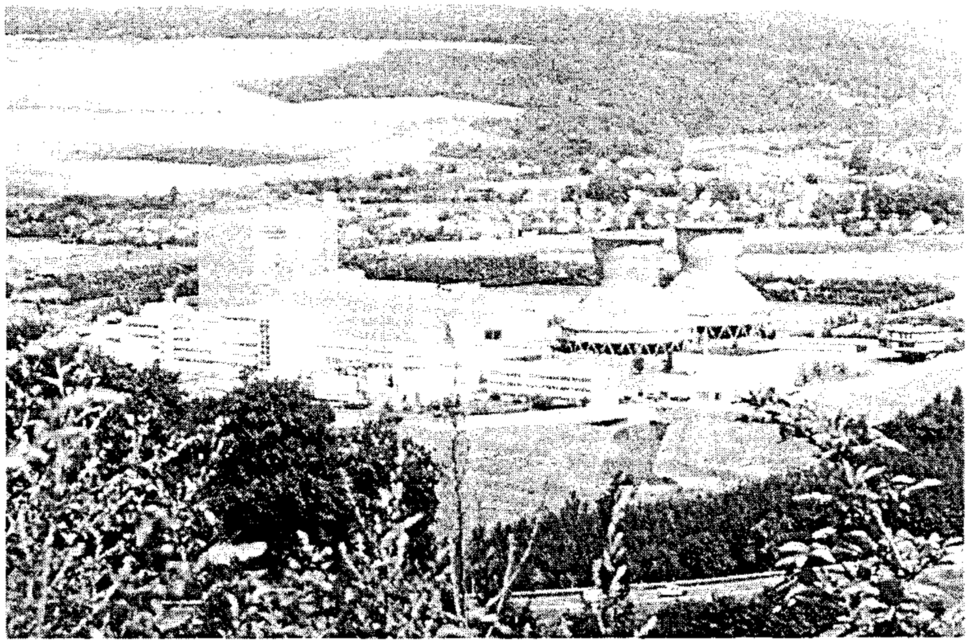




TÜV NORD GRUPPE

Main Projects: Decommissioning





Main Customers

- Ministry for the Environment of Lower Saxony
- Ministry for Trade and Economics of North-Rhine Westphalia
- Hessian Ministry for the Environment
- Ministry for the Environment of Mecklenburg-Vorpommern
- Federal Agency for Radiation Protection
- others, e.g. ESKOM (South Africa)

Legal Background of TÜV Role in Nuclear Technology

• Each nuclear facility	⇨	to be licensed according to Nuclear Energy Act
• §7 of Nuclear Energy Act		licensing prerequisites
• Most important prerequisite		state of science and technology ("state of the art")
• Verification of licensing prerequisites		external experts may assist
• Technical part of verification		TÜV as consultant of the authorities

⇨ TÜV has complete overview of the technical state of the plant and licensing and surveillance procedures

Tasks of TÜV in Nuclear Technology

- Safety assessment preceding erection and operation of nuclear installations, e.g. NPP, Storage Facility
- Surveillance and safety assessment during operation:

⇨Routine tasks:
- On-site inspections, e.g. recurrent periodic inspections
- Evaluation of modifications, e.g. reload patterns
- Surveillance during plant outages

⇨Special tasks:
- Evaluation of incidents
- Evaluation of the PSA

- Decommissioning of nuclear installations, e.g. NPP

Extent of TÜV in Assessment and Surveillance

The following figures are valid for an LWR NPP:

- Safety assessment and inspections preceding erection and operation: total of ca. 250 man years

- Surveillance and safety assessment during operation:

  ⇨Routine tasks: total of ca. 20 man years per year
  ⇨Special tasks: total of ca. 5 man years per year

Steps in NPP Licensing in Germany (Normal Procedure)

Licensing basis:	§ 7 of the Nuclear Energy Act; basic requirements to be met by the application
Application:	To be submitted to the licensing authorities by the applicant
Common practice:	Licensing in consecutive steps ("partial license")

- concept and buildings
- components and systems
- non-nuclear preoperational tests
- permanent operation license

During operation:	Licensee can apply for modifications of the licensed plant

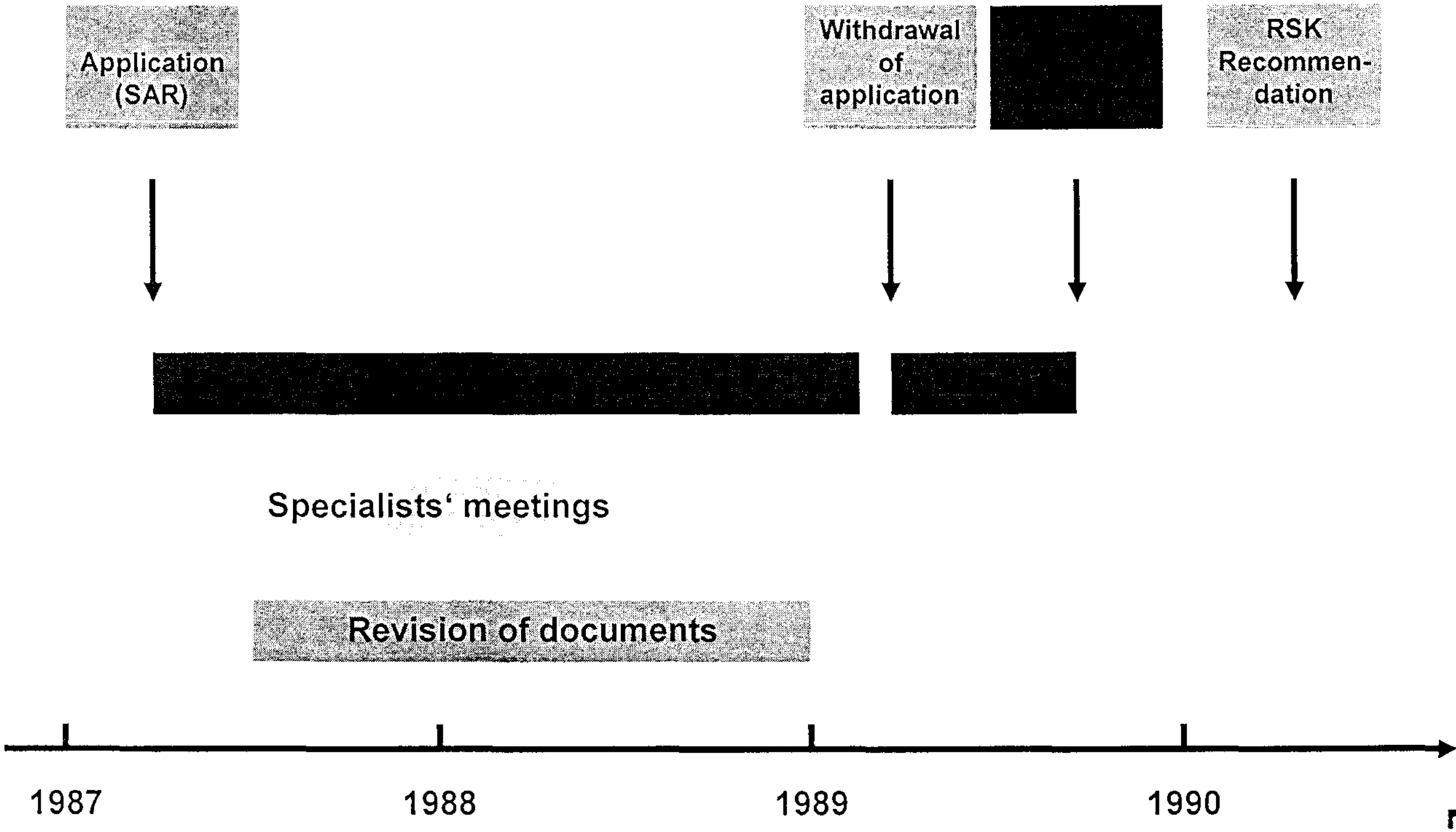
Licensing Procedure for the HTR-2 Modular Reactor

- Licensing basis: § 7a of the Nuclear Energy Act (site-independent license)
- Extent and content of the application (and license) less than the first partial license in a normal licensing procedure, but exceeding that of a conceptual license
- Validity of license limited to a certain time (previsional license)
- Licensee not obliged to make use of the licensing decision

Course of the Licensing Process for the HTR-2 NPP

- **April 1987:** Application of Siemens/Interatom for a site-independent license of the German HTR-2 NPP according to § 7a of the Nuclear Energy Act
- **Licensing authority:** Ministry of the Environment of Lower Saxony
- **Expert organization:** TÜV Hannover/Sachsen-Anhalt e.V.
- **April 1989:** Withdrawal of application for political reasons, termination of the licensing process
- **May 1989:** Continuation of safety assessment by TÜV Hannover-Sachsen-Anhalt e.V. under contract of the Federal Ministry for Research and Technology
- **October 1989:** Delivery of TÜV safety assessment report as input for RSK recommendation (May 1990)

Licensing of the German HTR-2 NPP



Role of TÜV in the HTR-2 NPP Licensing Process

- **Task:** Safety assessment of concept and design of the HTR-2 NPP
- **Applied method:** Interdisciplinary team - matrix structure – iterative procedure:
  - ⇨ Specialists' meetings in 1987 and first half of 1988
  - ⇨ Revision of licensing documents and completed by applicants early in 1989
  - ⇨ Continuation of assessment in 1989
- **Experts'co sts:** Equivalent to 25 man years
- **Result:**
  - ⇨ Complete and consistent review of an advanced HTR concept
  - ⇨ Safety Assessment Report
  - ⇨ Approval of concept by Reactor Safety Commission

HTR-Module-Specific Technical Rules

- **Problem:** No technical rules and guidelines for HTR-2 NPP available for design and safety assessment
- **Solution:**
  - ⇨ Screening of existing technical rules and guidelines for LWR
  - ⇨ "Filtering" of HTR-specific aspects
  - ⇨ Consideration of concept-specific features
  - ⇨ Consideration of concept-specific scientific and technical publications

⇨ Comprehensive and consistent set of design and evaluation criteria applicable to the HTR-2 NPP

- **Procedure:**
  - ⇨ Derivation and proposal by the applicants
  - ⇨ Verification, modification and approval by the TÜV experts

Documentation of the Assessment Results

- **Safety Assessment Report:** ca. 900 pages, in German
- **Summary:** Safety Assessment of the Design of the Modular HTR-2 NPP

  TÜV Hannover, May 1990

  51 pages, in English and German
- **Publication:** H. Helmers and H. Knieper:

  Review of the safety concept of the HTR 2 reactor plant

  Nuclear Engineering and Design 137 (1992) 89-95

Present Involvement in the HTR Field

TÜV Hannover/Sachsen-Anhalt e.V. contracted by ESKOM for different tasks in the licensing process of the South African Pebble Bed Modular Reactor (PBMR):

- Derivation of a safety classification system and the integrated design process

- Review of Safety Analysis Report, Rev. 0b

- To be expected (contract under negociation):
  Review of Safety Analysis Report, Rev. 1,
  and further QA tasks in the licensing process

Know how transfer to ESKOM for the PBMR

Visit of the NRC-Delegation in Germany
July 23-26, 2001

Dr. Josef Schöning

Westinghouse Reaktor GmbH
Mannheim, Germany

Agreements with Eskom / PBMR

MOU from April 4, 1996

- Partner:
  - Eskom

    ✧
  - German Working Group (KT GmbH / Prof. Schulten, FZJ)
  - HTR GmbH
- Objective: Willingness of the German Partners in principle to support Eskom in their PBMR Project and to give access to the German HTR know how

Agreement from August 8, 1996

- Partner:
  - Eskom

    ✧
  - HTR GmbH
- Objective: Supply of the SAR of the HTR-Module and support Eskom's investigation into the feasibility of a HTR

License Agreement from March 12, 1999

- Partner:
  - Eskom

    ✧
  - HTR GmbH
- Objective: Access to the HTR Technology related technical documentation as well for the HTR reactor as for the HTR Fuel as far as available in the archives.

  Technical assistance and consulting services, agreed on a case by case basis.

Know How Transfer to ESKOM for the PBMR

❐ Enabling Agreement for the Provision of Technical Assistance from June 2001

- Partner:
  - PBMR (Pty) Ltd

    ✧
  - Westinghouse Reaktor GmbH

- Objective: Scope of Technical Assistance:

  Layout, design, construction and calculations of reactor components and reactor systems, e.g.
  - reactor pressure vessel and manifold
  - Graphite core internals and metallic components
  - Disassembly equipment
  - fuel element detection and burn-up measurement system
  - Helium fittings
  - Waste handling system and decontamination system
  - Equipment handling system
  - Other auxiliary reactor systems
  - Radiation monitoring system
  - Reactor protection system and post-event instrumentation
  - Distributed control system

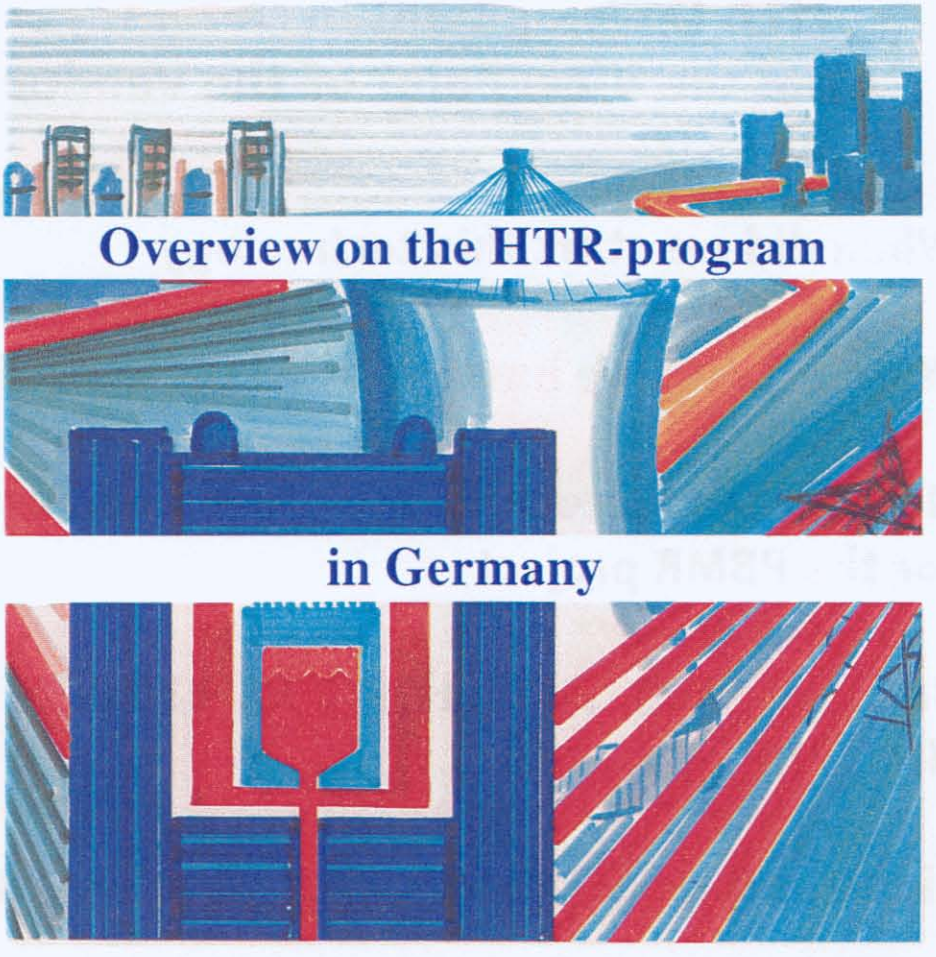
  Investigations of High Temperature Reactor specific Subjects, e.g.
  - Disassembly and maintenance concept
  - Release of fission products and plate-out of solid fission products
  - Impacts caused by graphite dust
  - Helium specific aspects (sealings, bearings, coating etc.)

Know how-transfer

1. Transfer of documents
   - 1.1 definition-ESKOM, WER/HTR
   - 1.2 selection-WER/HTR
   - 1.3 explanation, interpretation,WER/HTR

2. Cooperation
   - 2.1 Consultance-WER/HTR (derivation, correlation, adoption)
   - 2.2 Support (single tasks)-WER/HTR

Overview on the HTR-program in Germany



Visit of the NRC-Delegation in Germany
July 23-26, 2001

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Westinghouse Reaktor GmbH
Mannheim, Germany

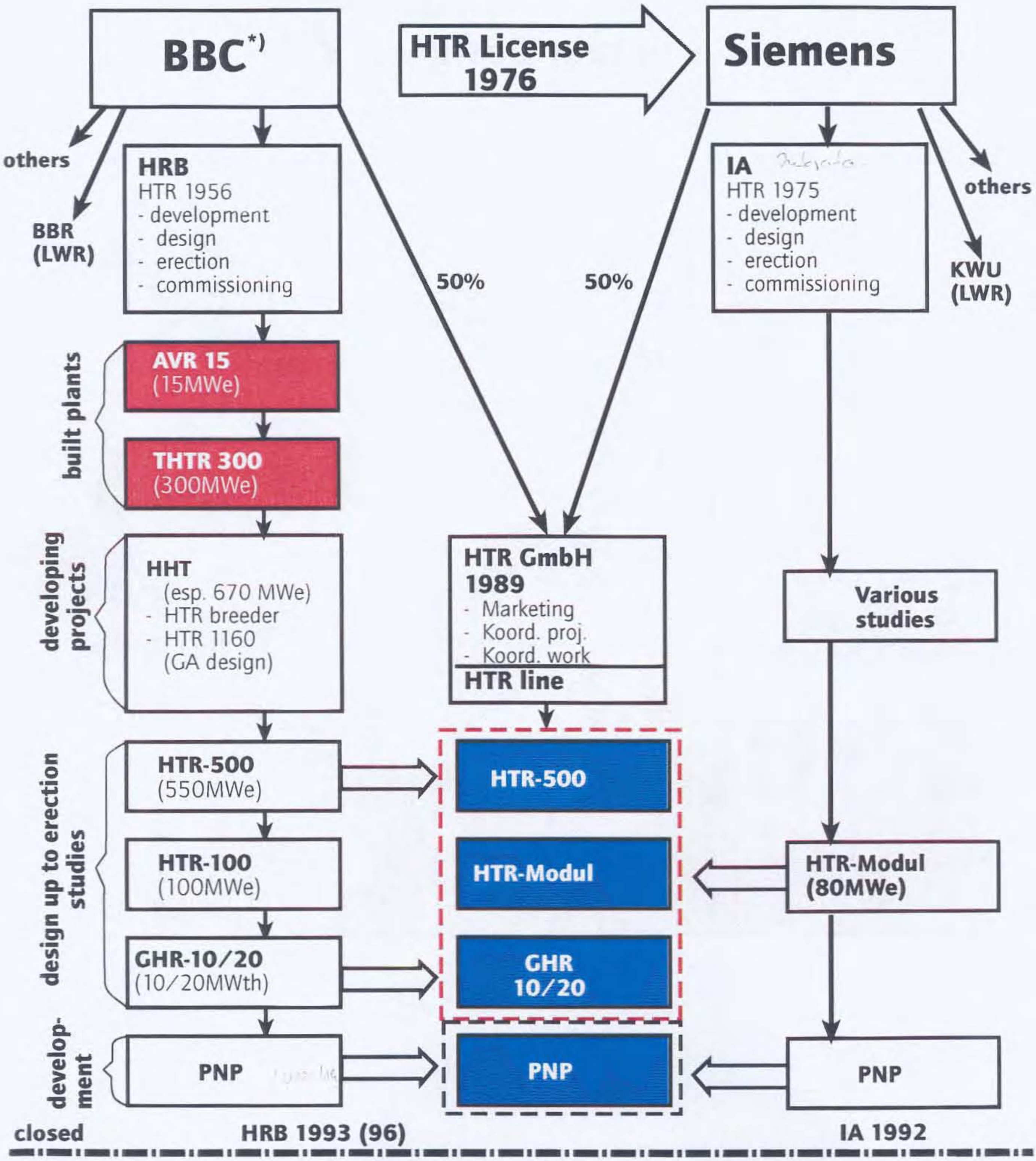
Overview on the HTR-program in Germany

HTR Development in Germany from the View Point of the Vendor

- When did we start this technology
- Which status we have reached up to now
- How far is this technology proven and usable for the PBMR project
- BBC line - AVR, THTR, HTR 500, HTR 100, GHR, HHT, PNP
- Siemens line - HTR Modul, PNP, Various studies
- Concentration of the HTR development in the HTR GmbH because of marketing and cost reasons
- Common reactor line - HTR 500, HTR 100, GHR
- Design and R&D of the main projects Dokument1

Overview on the HTR-program in Germany

HTR Development in Germany



today

HTR GmbH still exists
(new shareholders-WER/FANP)
- **markt observation**
- **providing of licensing and consulting contracts**

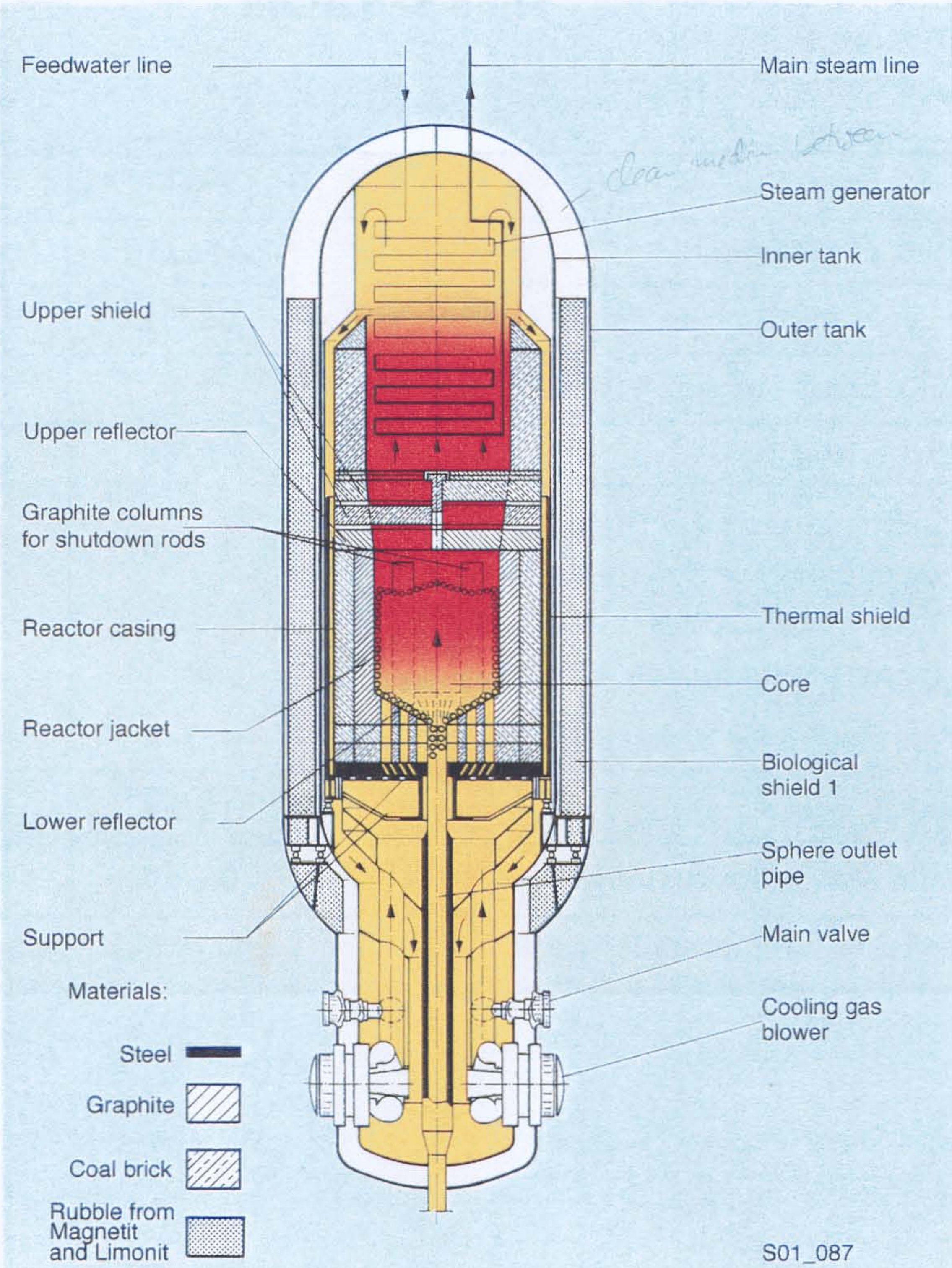
*)BBC → ABB → BNFL/WER

Overview on the HTR-program in Germany

Main Design Data

		AVR	THTR	HTR 500	HTR 100	HTR M	GHR
Thermal Power	MJ/s	46	750	1390	258	200	10/20
Net electrical power	MW	15	296	550	100	80	
Mean power density	MW/m^3	2,6	6	6,56	4,2	3	0,8
Primary gas pressure	bar	10,8	39	55	70	70	30
Hot gas temperature	°C	850/950	787	700	700	700	450
Cold gas temperature	°C	275	262	260	250	250	250
Main steam pressure	bar	72	177	180	190	190	
Main steam temperature	°C	500	530	530	530	530	

AVR
General View on the Reactor

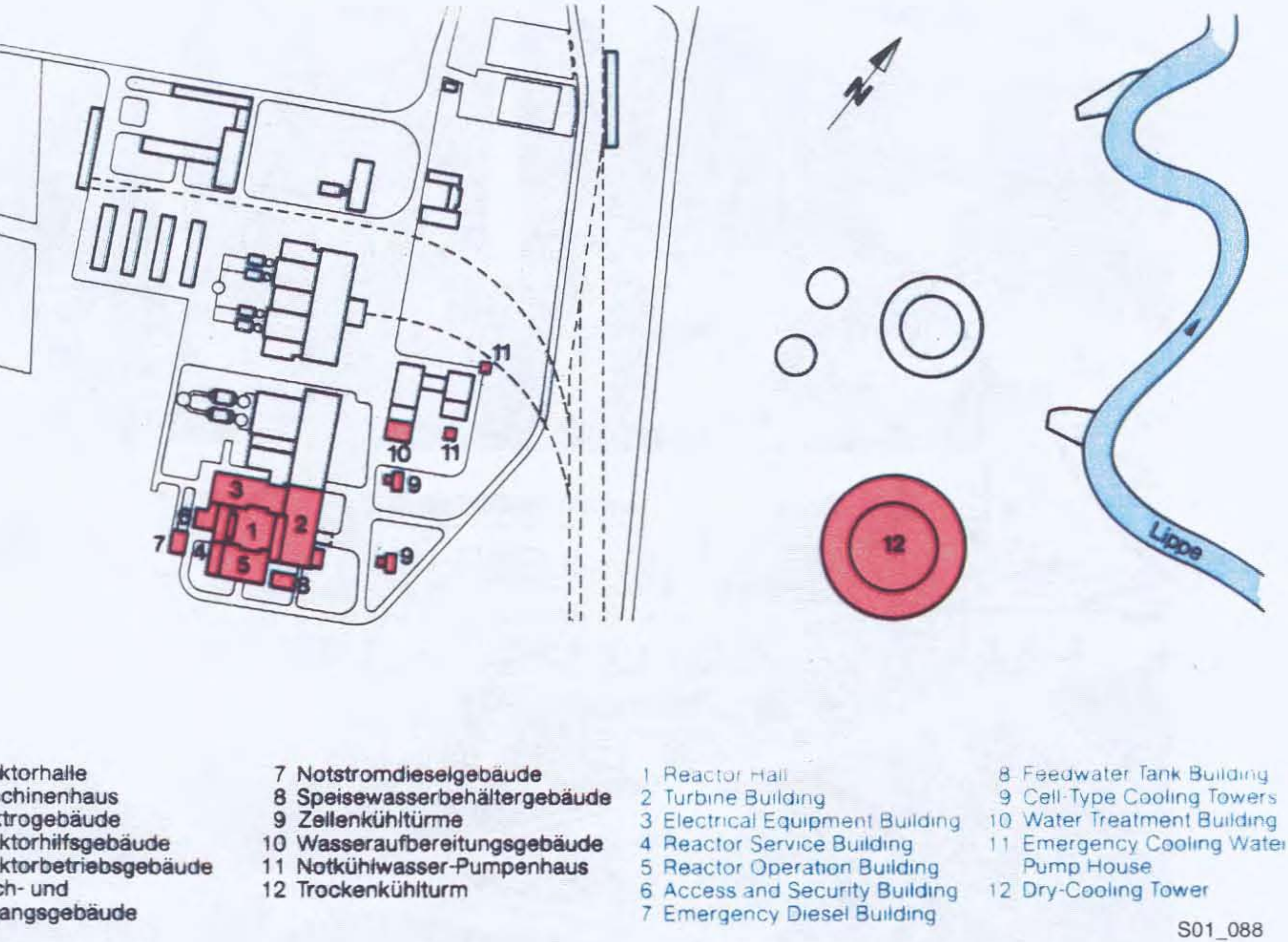


Overview on the HTR-program in Germany

Built Plants
Time Schedule

	AVR	THTR
Start of Planning	middle of '57	middle of '60
Order by utility group and start of construction	08/61	06/71
PCRV proof pressure test		09/82
First criticality	08/66	09/83
Functional tests (THTR) Components and systems 1y Total plant nuclear 0 - 40 % 1y Power production on 40 - 100 % 1y		
First electricity to the grid	12/67	11/85
100 % power output	12/69	09/86
Handover to the customer	05/69	06/87
Decision for decommissioning	End of '89	End of '89

THTR
Site Layout



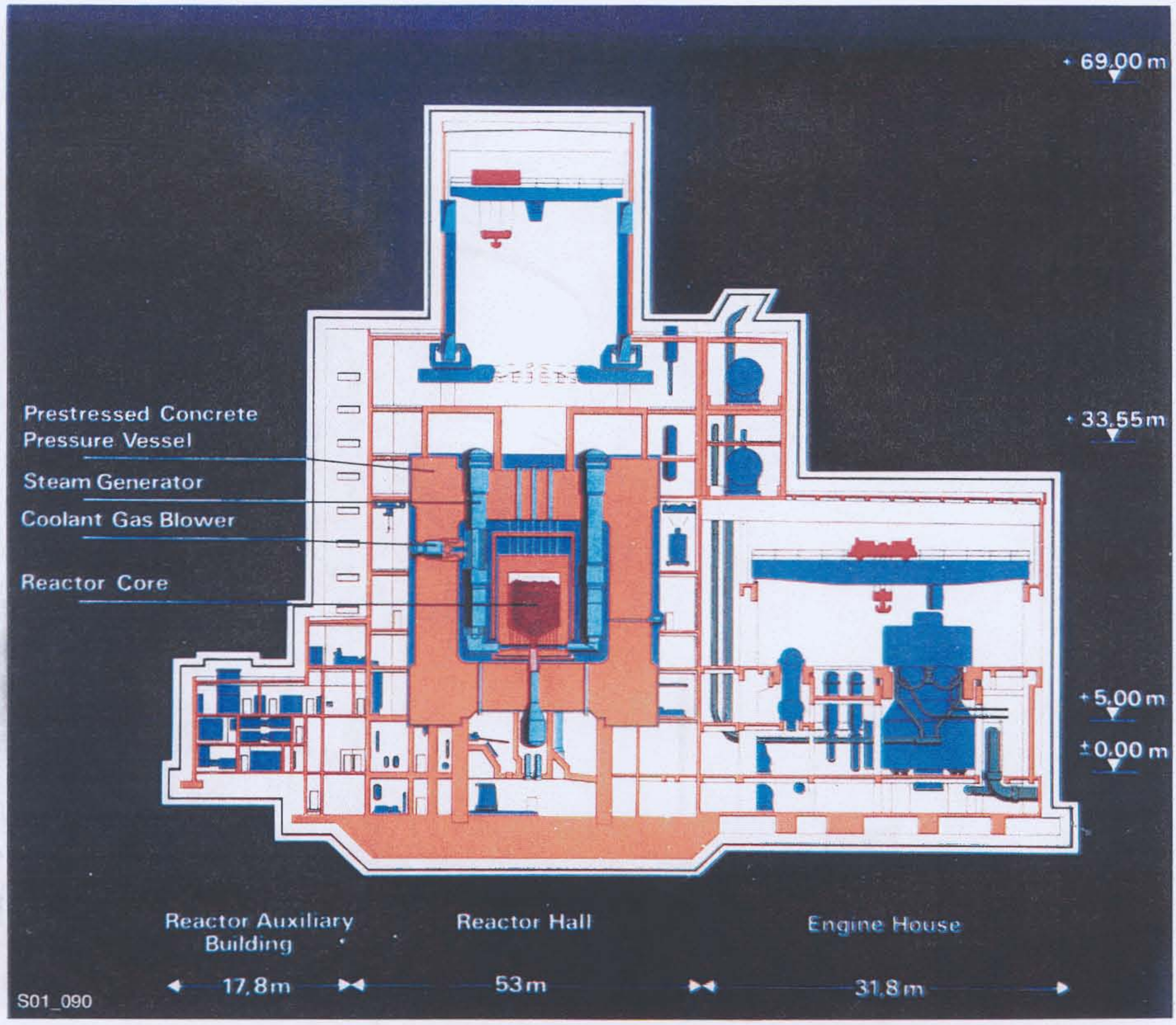
1 Reaktorhalle
2 Maschinenhaus
3 Elektrogebäude
4 Reaktorhilfsgebäude
5 Reaktorbetriebsgebäude
6 Wach- und Zugangsgebäude
7 Notstromdieselgebäude
8 Speisewasserbehältergebäude
9 Zellenkühltürme
10 Wasseraufbereitungsgebäude
11 Notkühlwasser-Pumpenhaus
12 Trockenkühlturm

1 Reactor Hall
2 Turbine Building
3 Electrical Equipment Building
4 Reactor Service Building
5 Reactor Operation Building
6 Access and Security Building
7 Emergency Diesel Building
8 Feedwater Tank Building
9 Cell-Type Cooling Towers
10 Water Treatment Building
11 Emergency Cooling Water Pump House
12 Dry-Cooling Tower

S01_088

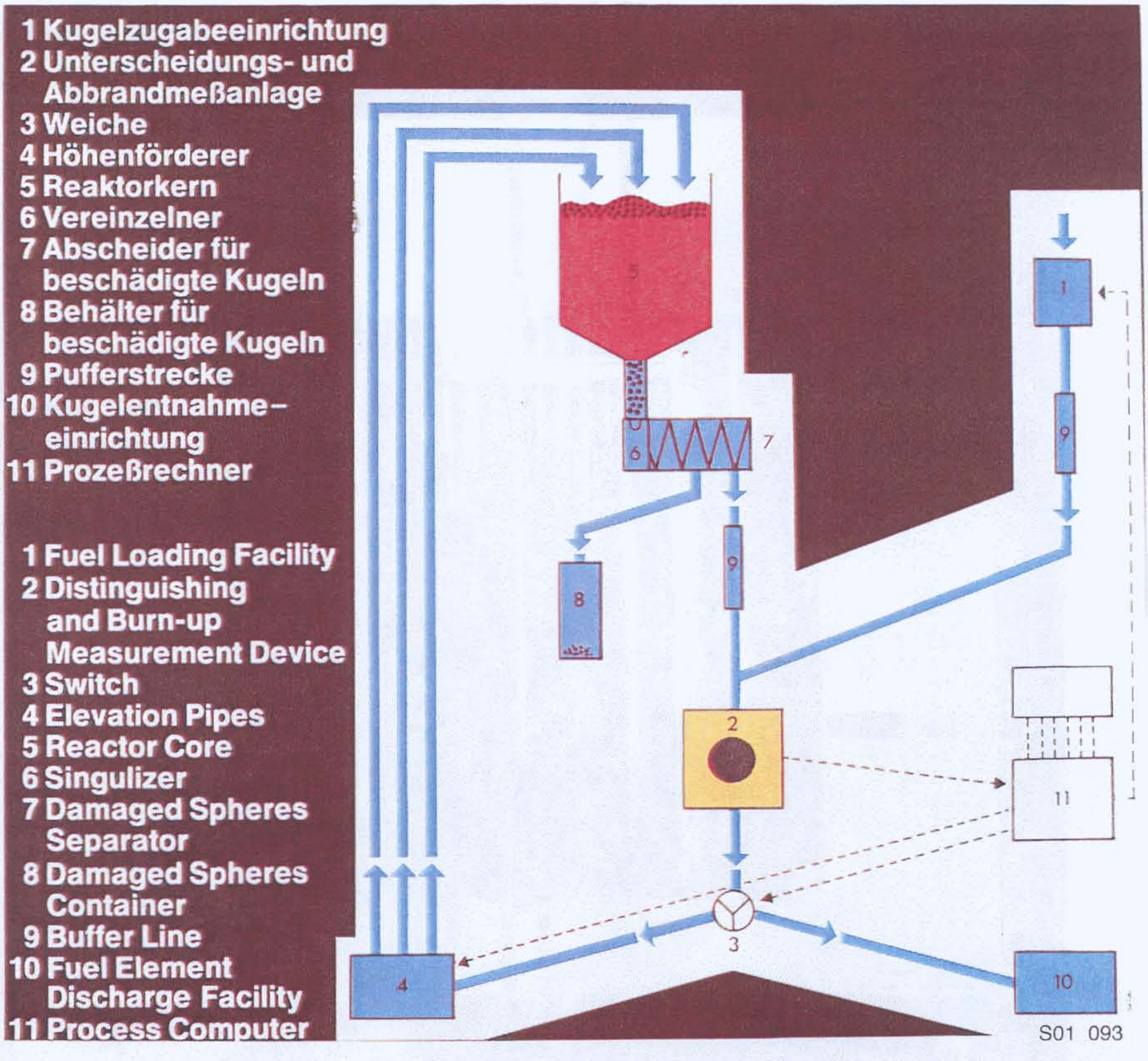
THTR
300 MWe Nuclear Power Plant
Hamm-Uentrop

THTR
Longitudinal Section



THTR
Dry Cooling Tower

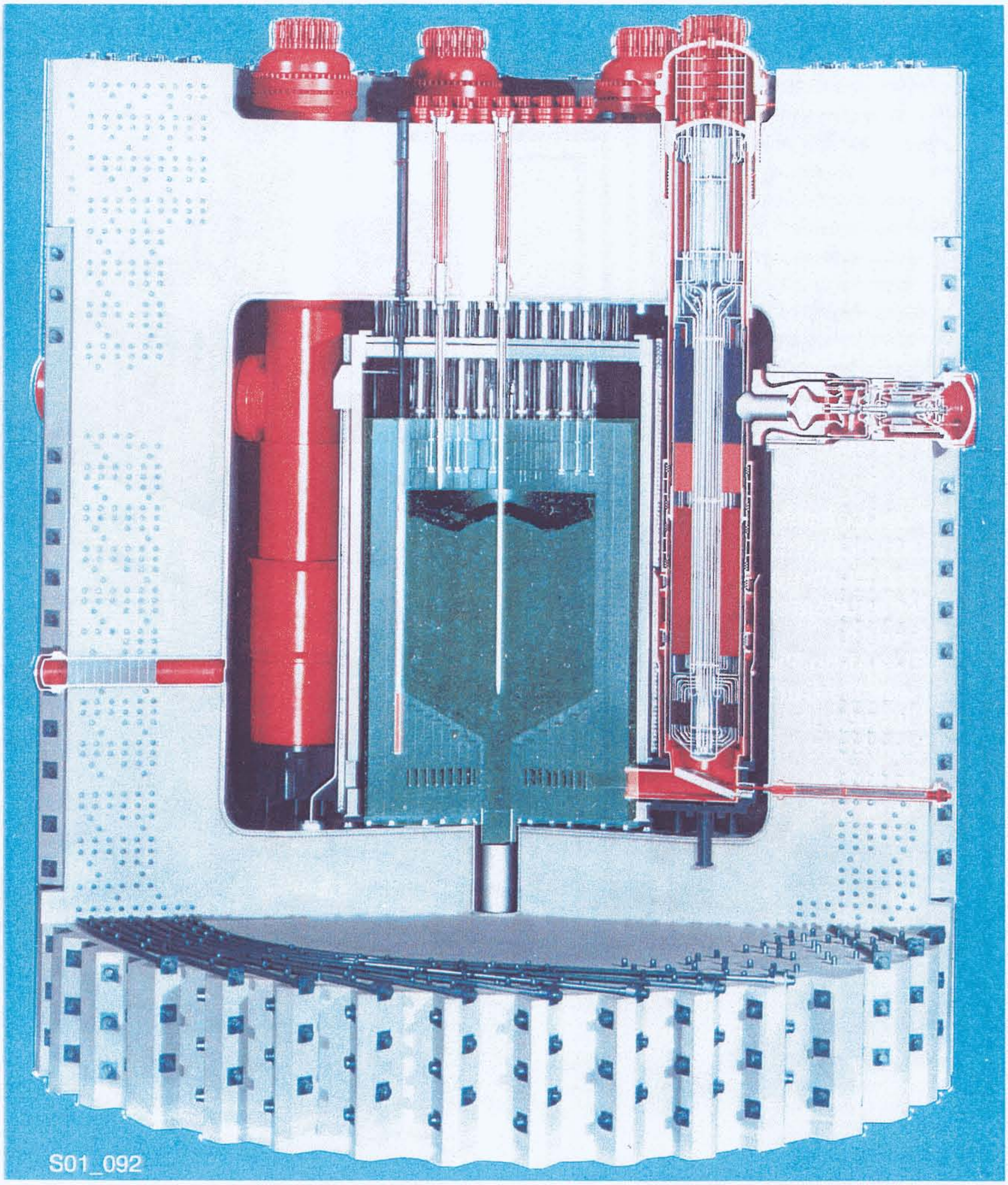
THTR
Fuel Handling System



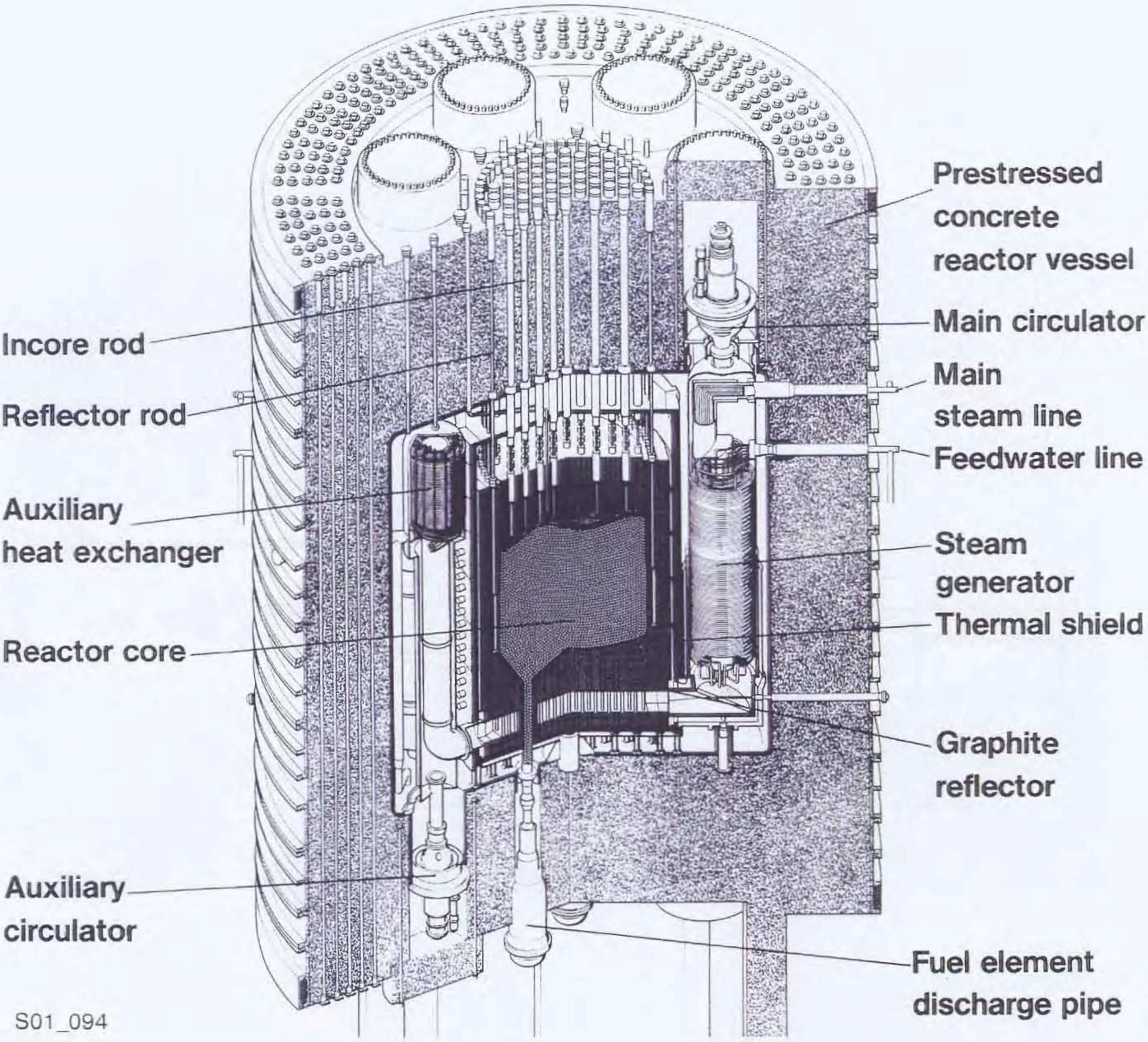
Westinghouse BNFL

Overview on the HTR-program in Germany

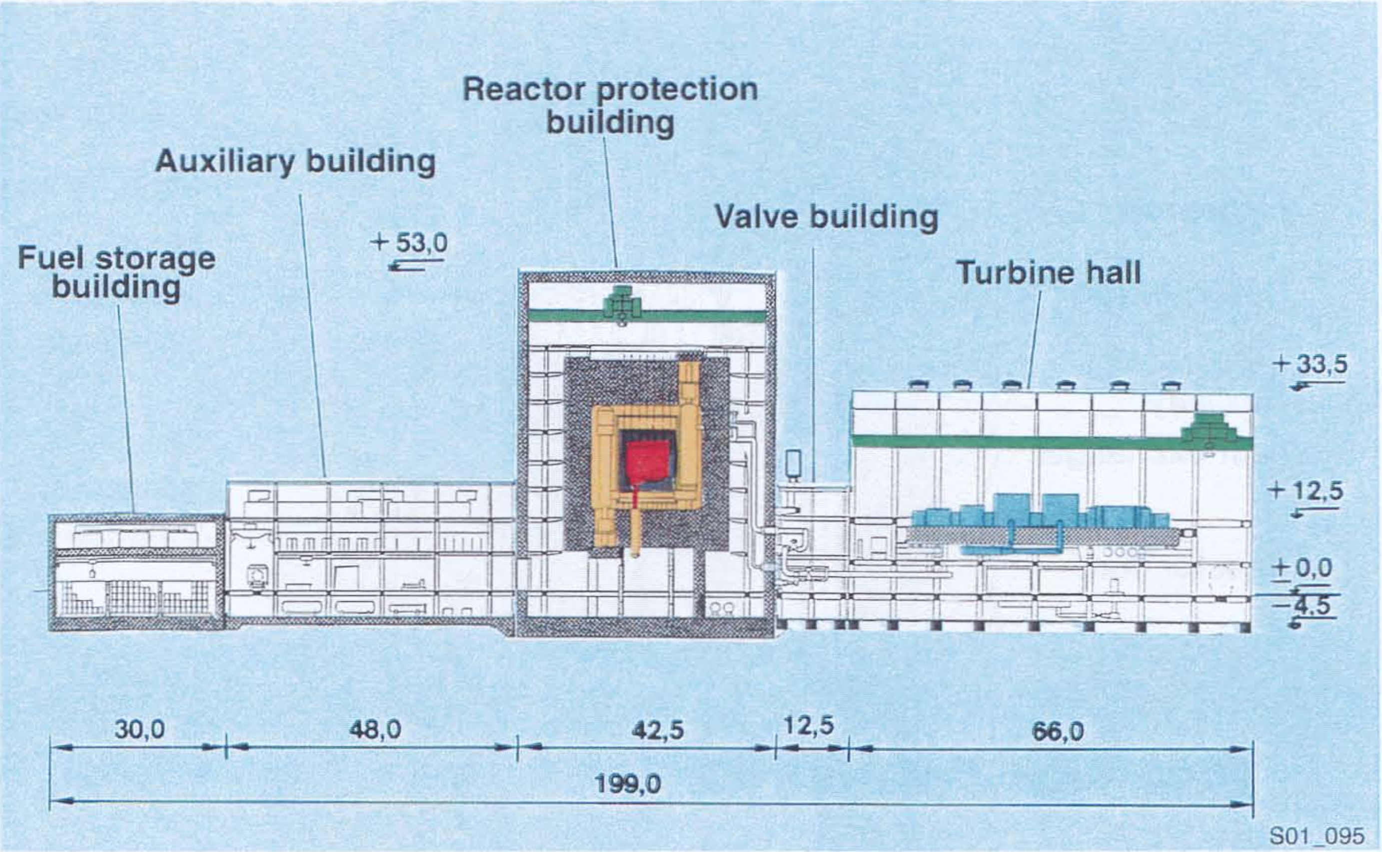
THTR
Prestressed Concrete Reactor Vessel



HTR 500
Reactor Pressure Vessel with Internals

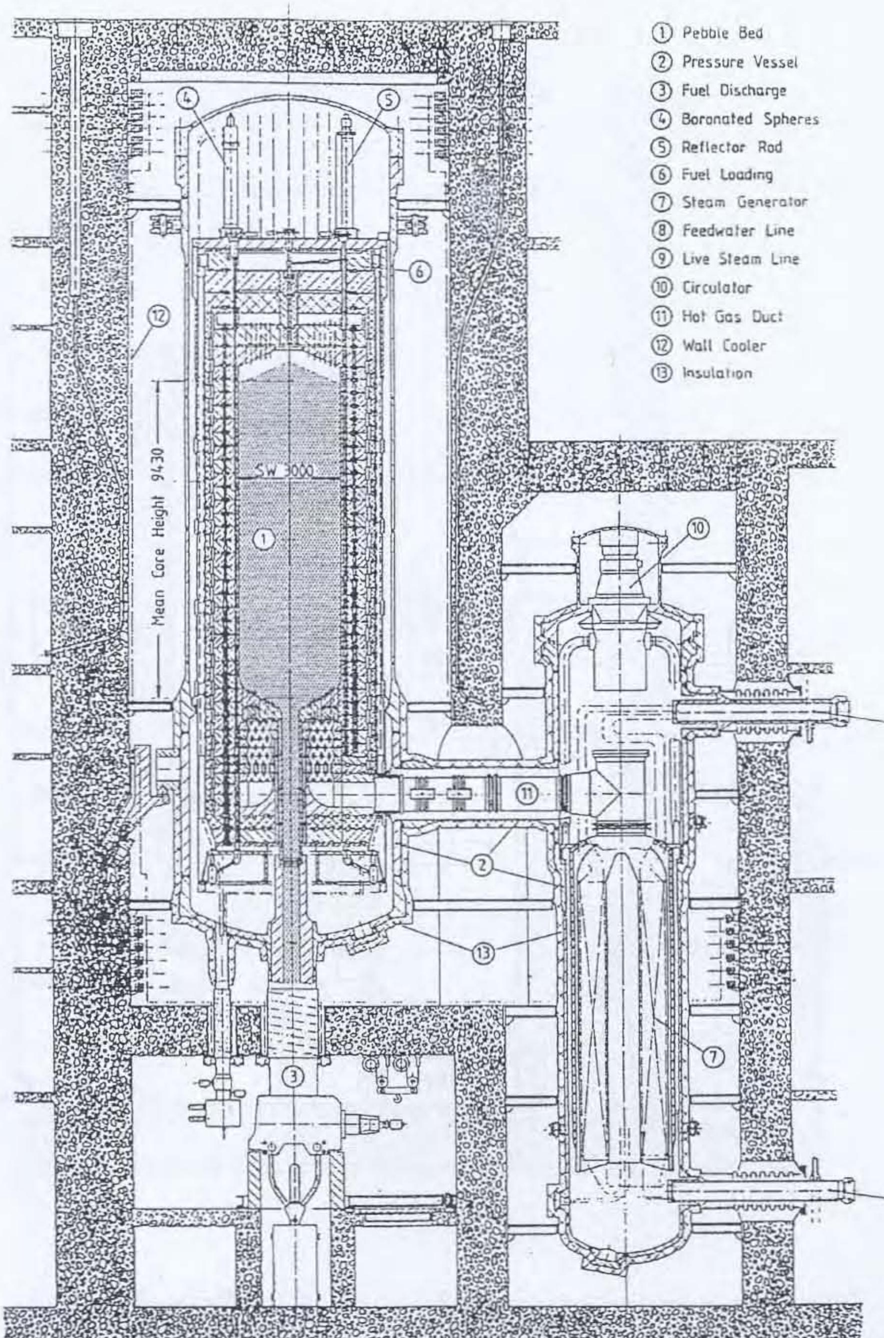


HTR 500
Longitudinal Section



Overview on the HTR-program in Germany

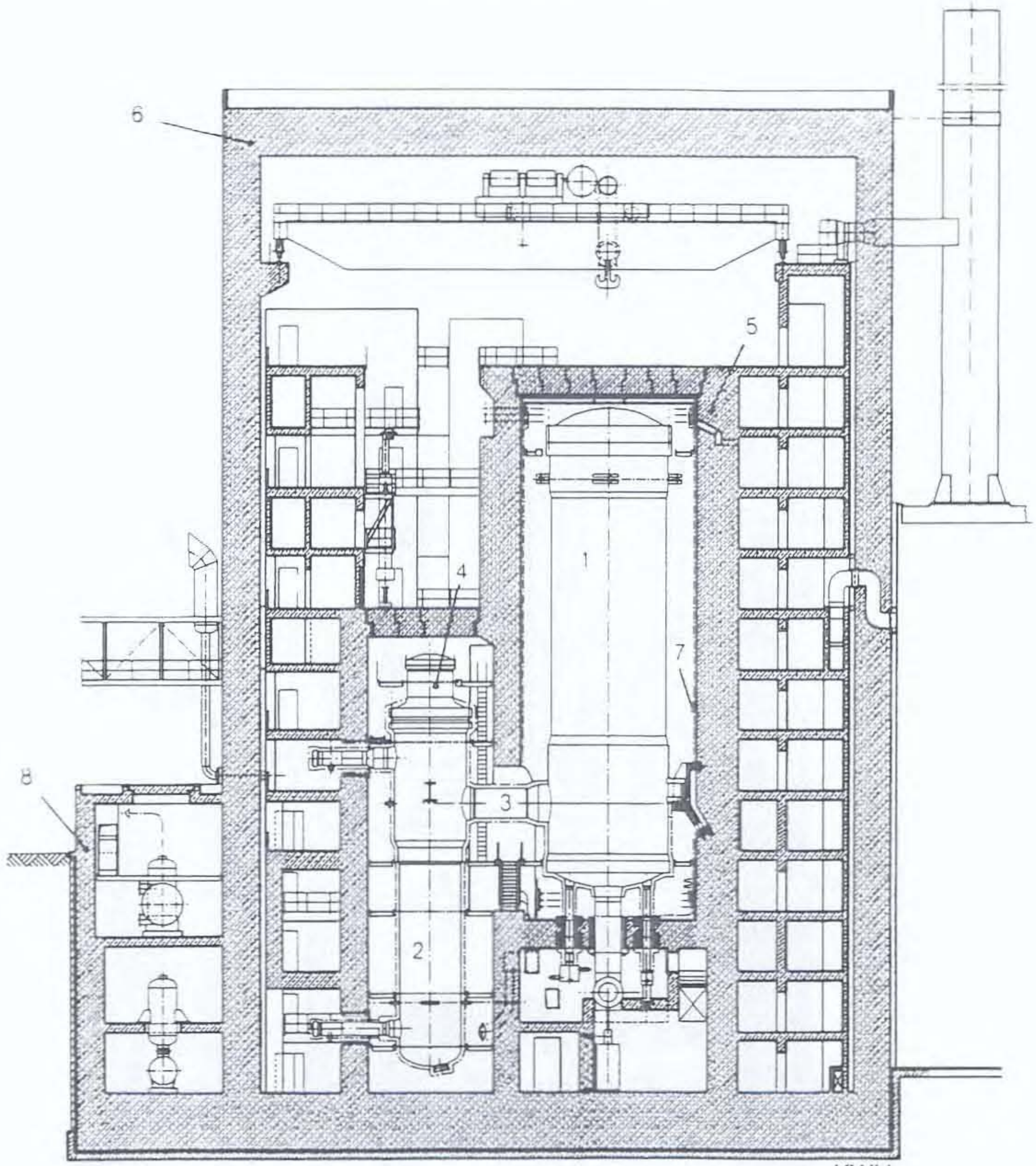
HTR Modul
Cross Section



Westinghouse BNFL

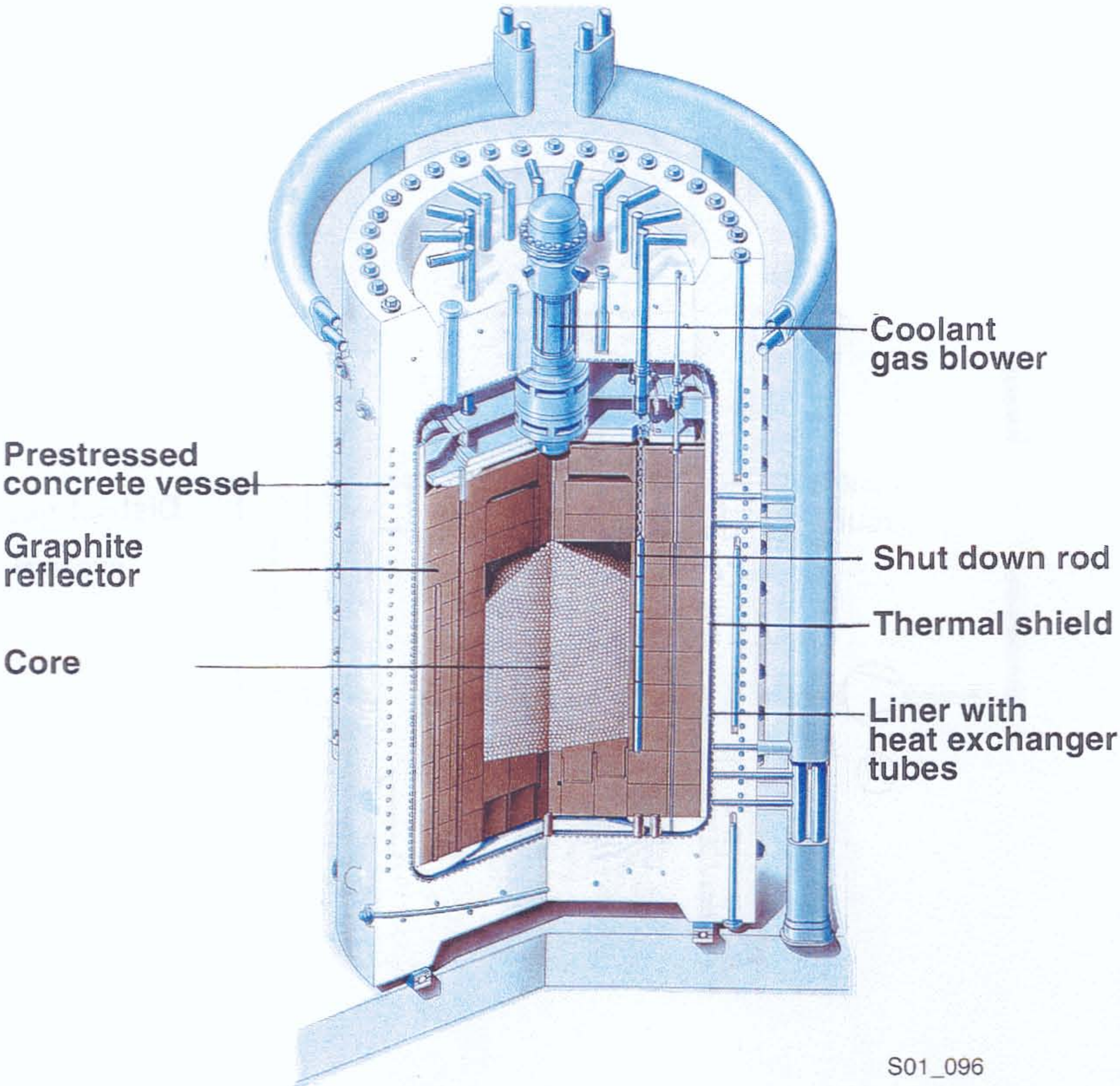
Overview on the HTR-program in Germany

HTR Modul
Cross Section

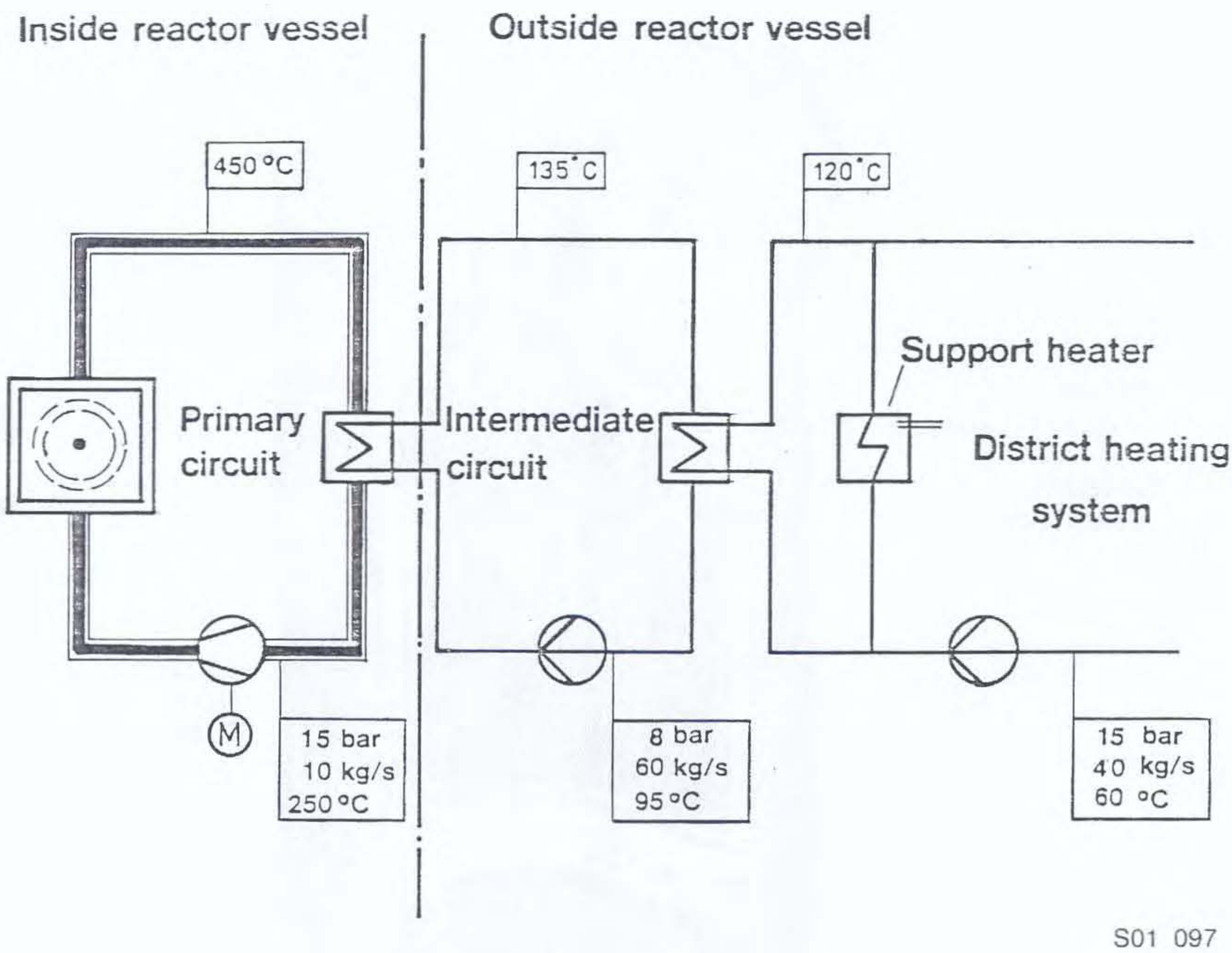


1	Reactor pressure vessel	5	Primary shielding
2	Steam generator vessel	6	Full protection wall
3	Connecting pipe	7	Liner cooler
4	Primary coolant blower	8	Reactor building extension

GHR 10
Gas Cooled Heating Reactor

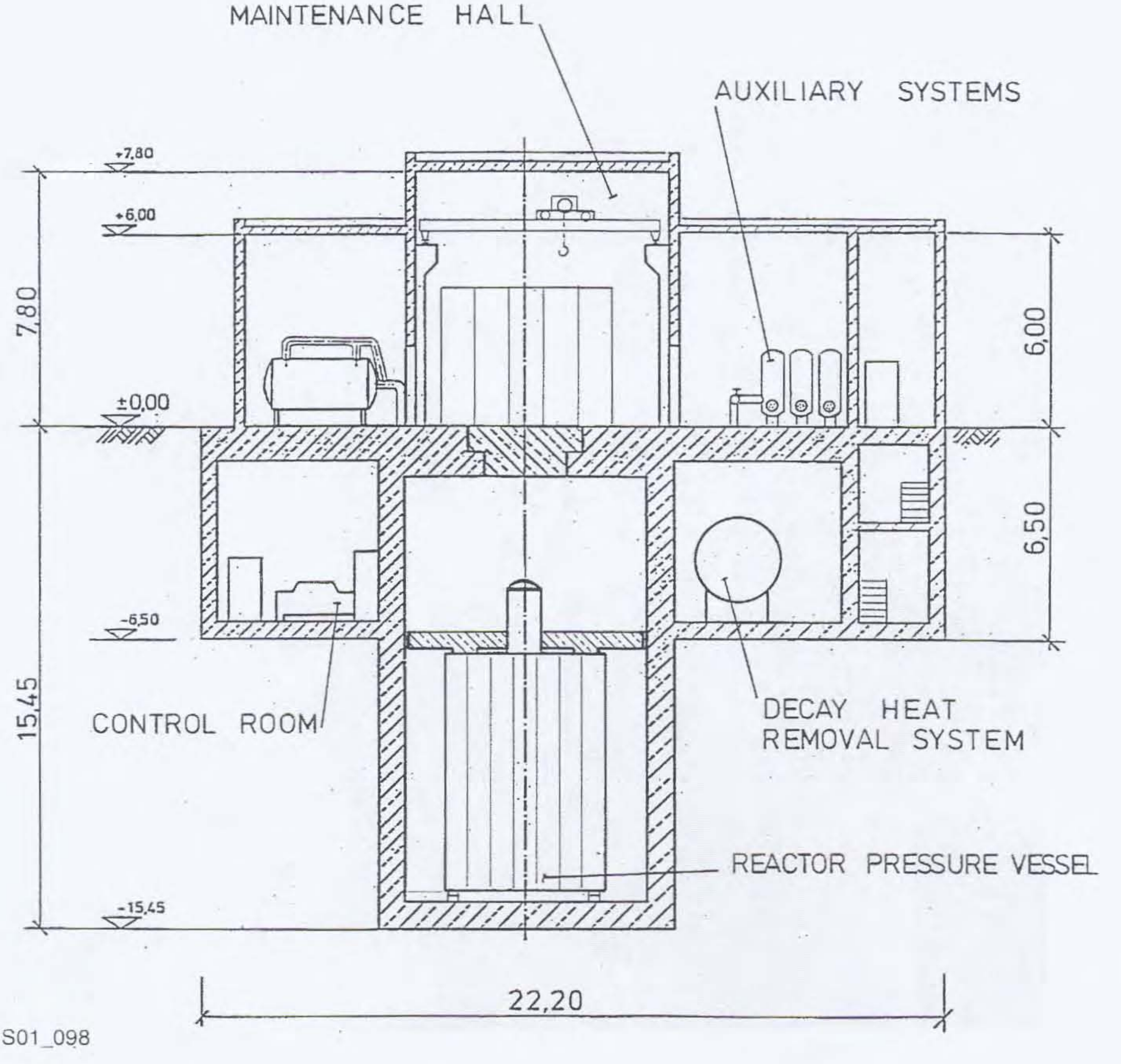


GHR 10
Heat Flow Diagram



Westinghouse BNFL

GHR 10
Longitudinal Section



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Overview on the HTR-program in Germany

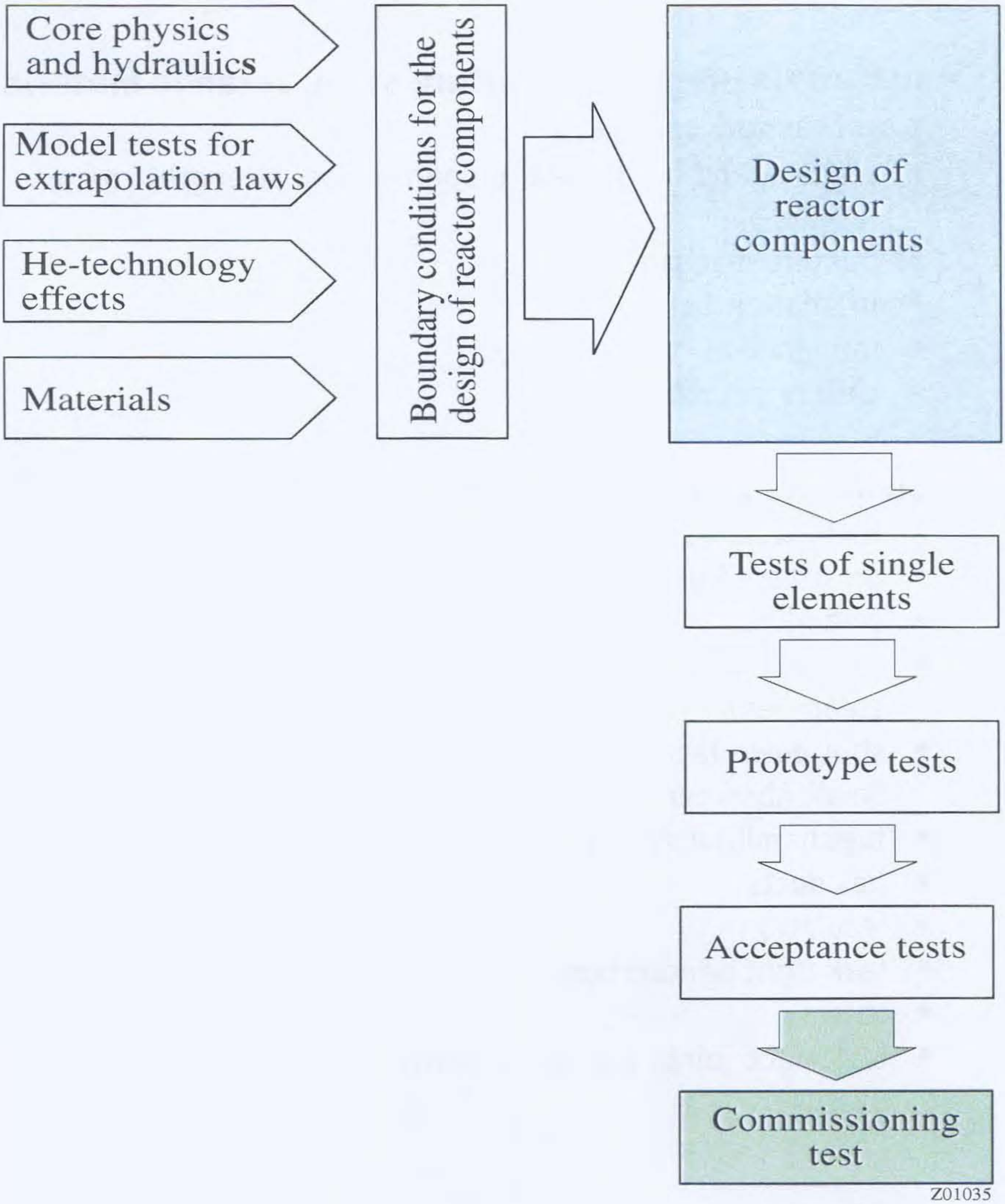
R & D Programs Subsidized by the Government

Project	Partners	Period	Subsidy %	Total Cost* Mio DM
THTR	BBC / HRB Nukem FZJ	64 - 88	95 - 100	~ 3200 (incl. erection)
HHT	BBC / HRB Siemens / IA Nukem FZJ German industry Swiss industry	69 - 82	60 - 90	~ 200
PNP	BBC / HRB Siemens / IA Nukem FZJ German industry	76 - 92	75 - 90	~ 150
Others (HTR500 HTR100 HTR Modul GHR10)	BBC / HRB Siemens / IA FZJ German industry Swiss industry	82 - 88	50 - 90	~ 100

*in the figures the costs of FZJ and foreign partners are not included

Overview on the HTR-program in Germany

Scope and Depth of R & D



Overview on the HTR-program in Germany

The R & D Programs

- **materials - high temperature steel, ceramic material, prestressed concrete**
  - physical and mechanical properties e.g. strength, creep, fatigue
  - fracture mechanics
  - irradiation behaviour
  - influence of corrosion
  - tritium permeation

- **elements and components**
  - fuel elements
  - prestressed pressure vessel
  - ceramic core internals
  - heat exchanger - steam generator, He/He heat exchanger, steam reformer
  - shut down facilities - incore rods, reflector rods, Small Absorber Spheres (SAS)
  - fuel handling system
  - gas ducts
  - insulation
  - leak tight penetrations
  - valves
  - leak tight joints e.g. screw joints, flanges

The R & D Programs

- **He-technology**
  - friction
  - wear
  - bearing
  - welding
  - coating
  - sealing

- **pebble bed mechanics - e.g. flow characteristics, forces, abrasion**

- **thermo hydraulics - e.g. leakages, bypasses, gas mixing**

- **fission product behaviour and decontamination**

- **earth quake behaviour especially of pebble bed and core internals**

- **impacts of graphite dust in the primary circuit**

- **He purification**

Overview on the HTR-program in Germany

Most Significant and Extensive R & D Work

	Investigations				
		experimental			
	analytical	m	c	s	p
Fuel elements	X	X	X		X
PCRV	X	X	X	X	
Ceramic core structure	X	X	X	X	
He-technology aspects	X	X	X		
Core mechanics	X			X	

m = material
c = components
s = model (various scales)
p = prototype

Nuclear Engineering and Design 121 (1990) 293–298
North-Holland

CONCEPT LICENSING PROCEDURE FOR AN HTR-MODULE NUCLEAR POWER PLANT

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In April 1987 the companies Siemens and Interatom applied in the West German state of Lower Saxony for a concept licensing procedure to be initiated for an HTR-Module nuclear power plant. In addition to a safety analysis report, numerous additional papers were submitted to the authorized experts. In April 1989 proceedings were suspended for political and legal reasons. By this time both the fire protection report and the plant security concept report had been completed. The safety concept review was continued by order of the Federal Minister for Research and Technology. The draft safety concept report was completed in July 1989. The final version was completed at the end of 1989.

1. Introduction: Description of the plant subject to the licensing procedure

The HTR-Module power plant is a thermal power plant for the cogeneration of electricity and process steam (fig. 1). The process steam can be used for a wide variety of applications in the chemical industry, for example, or for tertiary oil extraction.

The place of the fossil-fired heat source of conventional power plants is taken in the HTR Module by two

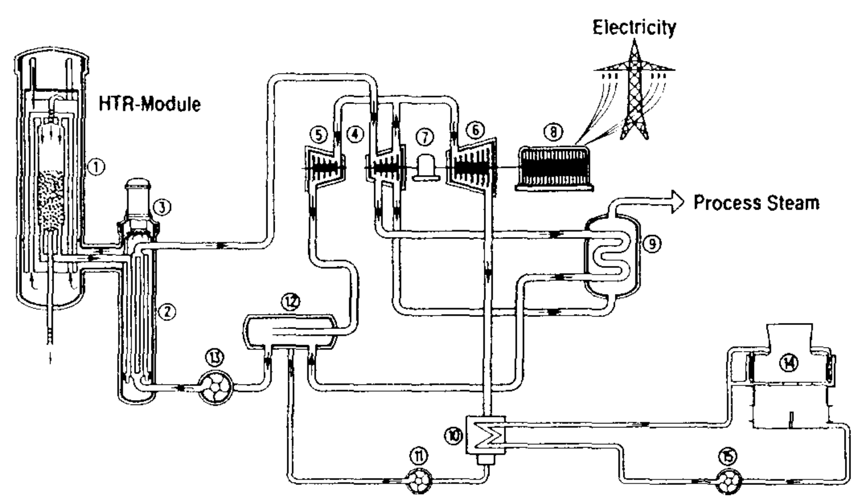

Fig. 1. HTR-Module power plant for cogeneration of electrical power and process steam.

0029-5493/90/$03.50

nuclear steam supply systems (modular units). Each modular unit comprises one high-temperature reactor, one steam generator and one primary gas blower.

The heat liberated by nuclear fission in the HTR is transported from the reactor through the coaxial duct to the steam generator by the primary coolant, helium, which is circulated by the primary gas blower. In the steam generator, the heat is transferred to the water/steam conversion system which is designed and operated as a purely non-nuclear plant.

Auxiliary and supporting systems connected to the primary system are provided for operation of the reactor; furthermore, safety of the reactor is assured by systems which fulfil the task of keeping loadings on components and structures within acceptable limits under accident conditions and which minimize the impact of accidents on the operating personnel and the environment. The reactor auxiliary systems are installed in the reactor building and in the reactor auxiliary building.

The cooling loads in the power plant are served by closed cooling water systems which remove the rejected heat to service water systems.

Operation of the power plant is controlled and monitored from the central control room. Normal operation is largely automated by means of open- and closed-loop controls which correct minor deviations from required setpoints. In the event of major deviations, automatic operational limiting controls restore the plant to the conditions requisite for operation. If trip limits set in the reactor protection system are reached, the necessary safety-related counter-measures are automatically initiated.

2. Licensing procedure / safety concept review

The licensing procedure for the concept of the HTR-Module nuclear power plant was intended to confirm the licensability of the plant and thus to establish a solid legal basis for further planning work.

2.1. Basis of licensing procedure

The construction and operation of nuclear power plants in the Federal Republic of Germany are subject to national legislation and ordinances such as the Atomic Energy Act (AtG), Atomic Licensing Procedure Code (AtVfV), Radiological Protection Ordinance (StrlSchV) and are also governed by engineering rules and regulations.

The licensing procedure for power plants is conducted in stages. At the first stage an application is submitted for a preliminary ruling on the concept pursuant to Article 7 of the AtG which would be preceded by a preliminary positive overall evaluation of construction of the plant and operation thereof pursuant to Article 18 of the AtVfV.

If no actual site has been selected, the application for a preliminary ruling is based on Article 7a (basis of the licensing procedure for the HTR-Module power plant).

As in the case of an application for the construction and operation of a nuclear power plant (Article 7) the requirements of Article 3 of the AtVfV have also to be satisfied. For a preliminary ruling pursuant to Article 7a of the AtG, emphasis is placed on:

- the Safety Analysis Report,
- supplementary plans, drawings and descriptions, and
- information on measures provided for protection of the plant against sabotage.

2.2. Time schedule of the licensing procedure / safety concept review

In April 1987 an application for a licence on the conceptual design was docketed by the State of Lower Saxony (FRG). In May 1987 the licensing authority nominated experts competent in the various fields of engineering. The main expert was the "Technischer Überwachungsverein Norddeutschland (TÜV Hanover)", who subcontracted parts of the nuclear steam supply system and radiological issues to the "Technischer Überwachungsverein Rheinland (TÜV Cologne)". The work for the licensing procedure was then initiated in May 1987 on the basis of the Safety Analysis Report. By the end of 1987 about 100 supplementary reports, including detailed drawings, stress analysis reports, reports on seismic design and fuel technology, had to be submitted. Further additional reports and other documents (approx. 150) were supplied in 1988. The submitted Safety Analysis Report was reviewed and reconsidered in June–August 1988 in the light of the standpoint of the licensing authority and the experts and to incorporate changes in system design. The corresponding changes to the Safety Analysis Report to take into account the results achieved in the licensing procedure were agreed, the Safety Analysis Report was revised and submitted in September 1988. The draft review report was completed in mid-1989 (fig. 2).

However, the application for a preliminary ruling on the concept was withdrawn and the licensing procedure discontinued by the Lower Saxon Ministry for the En-

April 87	Application for initiation of concept licensing procedure pursuant to Art. 7 a of the Atomic Energy Act docketed with Lower Saxon Ministry for the Environment (licensing authority) on the basis of safety analysis report submitted by Siemens/Interatom
Mai 87	Lower Saxon Ministry for the Environment retains TÜV Hanover to conduct safety review of HTR Module concept
Sept. - Dec. 87	Technical consultations with experts and licensing authority; approximately 100 technical documents generated for this purpose
Febr. 88	Experts call for more supplementary technical documents
Sept. 88	Revision of safety analysis report completed; submission to licensing authority and expert
Dec. 88	Start of RSK consultations (not yet completed)
Febr. 89	Report on fire protection concept completed
March 89	Report on plant security concept completed
April 89	Application for concept licensing procedure withdrawn by applicant and proceedings suspended by Lower Saxon Ministry for the Environment
May 89	Review continued by TÜV Hanover on behalf of BMFT
July 89	Draft review report submitted by TÜV Hanover
Sept. 89	Final meeting of RSK Subcommittee for HTRs
Oct. 89	Final meeting of RSK Subcommittee for Electrical Engineering
Dec. 89	Completion of final review report

Fig. 2. Time schedule of the licensing procedure/safety concept reviews.

March 90 Recommendation on the HTR Safety Concept by RSK

vironment in April 1989 as the legal case for a non-site-specific licensing procedure in Lower Saxony was questionable.

The order placed with TÜV Hanover for the safety review was therefore cancelled by the Lower Saxon Ministry for the Environment. By this time the review had reached an advanced state of progress and great interest was expressed in the work being completed. Consequently, TÜV Hanover was retained in May 1989 to complete the safety concept review without reference to a specific licensing procedure as part of a research and development contract on behalf of the Federal Minister for Research and Technology (BMFT) and to document the results. The draft safety review report then followed in July 1989. The final version was completed in late 1989.

Excepted from this review were the fire protection concept and aspects of protection against sabotage. The Lower Saxon Ministry for the Environment placed orders for separate reviews for these topics.

A review of the fire protection concept was performed by the Civil Engineering Institute (IBMB) of the University of Brunswick, and a review of the plant security concept by the Reactor Safety Association (GRS). These reports were completed within the scope of the licensing procedure in February and March 1989.

At the request of the Lower Saxon Ministry for the Environment, the Reactor Safety Commission (RSK) started reviewing the concept of the HTR-Module in late 1988. The concept was presented at numerous sessions of the RSK subcommittees for HTRs and electrical engineering. These subcommittees completed their consultations in September and October 1989. Pending the completion of discussions by the civil engineering subcommittee it is expected that the RSK will announce its recommendation on the HTR-Module in spring 1990.

3. Safety analysis report

For light-water reactors the contents of Safety Analysis Reports are governed by a Bulletin of the Federal Ministry for the Interior (BMI) with the title "Contents Checklist and Format for a Standard Safety Analysis Report for Pressurized Water Reactor and Boiling Water Reactor Nuclear Power Plants". For the HTR-Module this "Contents Checklist" was modified to reflect:

- the lack of a site,
- the application for a preliminary ruling, and
- the new technology.

Fig. 3 shows the contents of the HTR Module Safety Analysis Report.

Since no decision on an actual site has yet been made, Section 1 of the Safety Analysis Report contains

I	Introduction
II	Table of Contents
III	List of Tables
IV	List of Figures
V	Abbreviations
VI	Codes from Identification System for Power Plants (KKS)
VII	Graphical Symbols used for Mechanical, Electrical and Instrumentation and Control Equipment
1	Site
2	General Design Features of the HTR Module Power Plant
3	Power Plant
4	Radioactive Materials and Radiological Protection
5	Power Plant Operation
6	Accident Analysis
7	Quality Assurance
8	Decommissioning
9	Waste Management Provisions
10	Guidelines and Technical Rules

Fig. 3. Contents of the Safety Analysis Report.

2 General Design Features of the HTR Module Power Plant

2.1 Introductory Remarks

2.2 Characteristic Safety Features

- Barriers against Release of Radioactivity
- Inherent Safety

2.3 Technical Design Features

- Reactor (fuel elements, reactor core, control and shutdown systems)
- Nuclear Steam Supply System (pressure vessel unit, primary system and steam generator isolation)
- Confinement Envelope
- Residual Heat Removal
- Emergency Power Supply
- Reactor Protection System
- Remote Shutdown Station

2.4 Nuclear Classification and Quality Requirements

2.5 Summary of Design Basis Events

2.6 Postulates and Measures for In-Plant Events

2.7 Postulates and Measures for External Events

Fig. 4. Contents of Section 2 (SAR).

postulated site conditions to the extent necessary for evaluation of the concept and for arriving at a positive preliminary overall evaluation. The postulates made for this purpose are representative of numerous potential sites in FRG. Section 2 of the Report contains a collation of general design features of the HTR-Module power plant on which the application for a preliminary ruling on the concept pursuant to Article 7a of the AtG is based. Fig. 4 shows the contents of Section 2 of the Safety Analysis Report.

Sections 3 to 9 of the Safety Analysis Report contain descriptions of the power plant and operation thereof in accordance with the requirements stated in the above-mentioned "Contents Checklist" of the BMI, with modifications specific to the HTR-Module. For purposes such as providing a basis for a preliminary overall evaluation, these sections give greater details on the concept described in Section 2 for postulated, exemplary conditions of service.

Since the HTR-Module has been designed as a universally usable energy source, detailing of the concept for actual conditions of service would in essence only have a bearing on:
- overall layout and non-nuclear structures,
- steam power conversion system, and
- electrical systems (grid connection).

In contrast, actual conditions of service and site conditions would hardly affect construction of the reactor systems such as
- reactor core,
- nuclear steam supply system, and
- reactor auxiliary systems,

and, by definition, have no influence on the general design features described in Section 2.

Section 10, titled "Guidelines and Technical Rules", gives a survey of the application or adaptation of the Bulletins of the BMI and of the KTA Safety Standards to the HTR-Module.

4. Results of reviews

4.1. Fire protection review

In addition to the fire protection concept described in Section 2.6 of the Safety Analysis Report, a wide range of, in part, very detailed additional documents were submitted to the expert, in this case the Civil Engineering Institute (IBMB) of the University of Brunswick. The report, completed in February 1989, approves the fire protection concept. The few requirements and recommendations can be fulfilled without problems and have no effect on plant technology and safety.

4.2. Plant security concept Review

The plant security concept—the entirety of provisions for protection of the plant and operation thereof against sabotage—was reviewed by the Reactor Safety Association (GRS). The report, completed in March 1989, approves the concept and concludes that the inherent features of the HTR-Module warrant less stringent security precautions than those taken for LWRs.

4.3. Safety concept review

The main document for the safety concept review by TÜV Hanover was the Safety Analysis Report, above all Section 2 which describes the general features of the HTR-Module central to the concept. The opinion of the expert as given in the Safety Concept Review and also special features and design principles are given below following the same breakdown as Section 2 (see fig. 4).

Section 2.2 (SAR): Characteristic Safety Features
- The engineering configuration and nuclear design of the HTR Module is such that, even in the event of postulated failure of all active shutdown and residual heat removal systems, the fuel temperature stabilizes

at 1620°C. No appreciable release of radioactivity from the fuel elements occurs below this temperature.
- Active residual heat removal systems which limit the loadings on components and structures surrounding the core can fail for several hours without the allowable limits being exceeded.

Assessment in report: approved.

Section 2.3 (SAR): Technical Design Features

- Fuel element
  - Coatings (TRISO).
  - Enrichment ($8 \pm 0.5\%$).
  - 1620°C max. temperature, minimal release through SiC layer.
  - Particle failure curve (manufacturing defects: $\leqq 6 \times 10^{-5}$, irradiation-induced: $\leqq 2 \times 10^{-4}$; accident-induced: $\leqq 5 \times 10^{-4}$).

  Assessment in report: approved.
- Reactor core
  - By virtue of core design, fuel temperature stays below 1620°C under all accident conditions even on loss of active residual heat removal.
  - Due to uranium content of 7 g per fuel element the reactivity excursion on water inleakage is less than on inadvertent withdrawal of all reflector rods.
  - Design for unrestricted load cycling between 50 and 100%.

  Assessment in report: approved; restriction on part-load operation below 50% and during the running-in phase (because no analyses submitted for this case): limits on absorber ball level in storage vessels.
- Shutdown systems
  - Shutdown by absorbers in reflector holes.
  - Shutdown by 6 rods and 18 absorber ball units.
  - Location of rod drive mechanisms in RPV.
  - Location of all absorber ball unit components needed for shutdown in RPV.

  Assessment in report: design and configuration approved. Reactivity balances for equilibrium core approved but those for running-in phase up to several months have relatively small margins; consequences: reactor power might be below of 200 MW at first.
- Pressure vessel unit
  - Consists of reactor pressure vessel, gas duct pressure vessel and steam generator pressure vessel inclusive of valve banks on RPV, nozzles of steam generator pressure vessel.
  - Offset configuration, thus limiting natural circulation in the primary system.
  - Leak before break, assured safety for entire pressure vessel unit.

  Assessment in report: approved after discussion of dissimilar-metal weld and change of material for main steam nozzle. Requirement: preservice pressure test to include RPV nozzles.
- Primary and secondary system isolation
  - Primary system by two valves in each line of which only one operated by reactor protection system (failsafe).
  - Secondary system by two valves in each line (failsafe) both actuated by reactor protection system whenever reactor is shut down. Consequently, rest of secondary system outside reactor building has no functions important to safety.
  - Primary system overpressurization protection: two safety valves; secondary system: one safety valve backed up by steam generator relief system.

  Assessment in report: approved.
- Confinement envelope
  - Consisting of reactor building and other features (secured subatmospheric pressure system, building pressure relief system, HVAC systems isolation).
  - Normal operation: no filtering.
  - At overhauls: filtering by exhaust air filtering system (aerosols).
    During major depressurization accident (non-isolable DN 65 line): unfiltered venting through two dampers to vent stack.
  - Other depressurization accidents: possibility of filtering by subatmospheric pressure system (iodine filter).
  - Environmental impact of all accidents far below limits prescribed in Art. 28.3 of the Radiological Protection Ordinance even without active measures taken or filtering; consequently no containment necessary.

  Assessment in report: approved. Requirement: higher grade exhaust air filtering system.
- Residual heat removal
  - Provided by secondary system, cavity coolers, helium purification system.
  - On loss of active cooling, residual heat removed from core to cavity coolers solely by thermal conduction, radiation and natural convection.
  - Secured component cooling system, two-train.
  - With cavity coolers intact and loss of core cooling, core can run hot for lengthy period of time (15 h) without design limits for RPV and concrete of reactor cavity being violated.
  - External supply can be connected to cavity coolers in the event of severe accident conditions.

  Assessment in report: approved (see emergency power supply below for restriction).

- Emergency power supply
  - Two trains served by two diesel generator sets, started by operational sequencing controls or by hand.
  - DC buses (e.g. reactor protection system) battery-buffered for two hours.
  - Reactor system can sustain loss of power for at least fifteen hours (loss of auxiliary power supply, failure of diesel generator sets) without design limits being violated.

  Assessment in report: approved. Restriction: quality assurance for diesels must be so strict that the diesel generators can certainly be started within the fifteen-hour period.
- Reactor protection system
  - Few process variables.
  - Three protective actions always actuated on shutdown (reflector rod drop, blower trip, steam generator isolation); additionally steam generator pressure relief on tube failure and primary system isolation during depressurization accidents.
  - All actions failsafe.
  - Station blackout longer than two hours can be sustained since all protective actions are initiated, plant is transferred to safe condition, reactor protection system has no further tasks to fulfil.

  Assessment in report: approved. Source-range neutron flux instrumentation to be of reactor protection grade.
- Remote shutdown station
  - Located in reactor building (designed for aircraft crash, blast wave).
  - Power supply by diesels in switchgear building.
  - On station blackout, single train battery power supply for fifteen hours, possibility of connecting up external power supply after that.
  - Monitoring functions only, except for absorber ball shutdown system initiation by hand.

  Assessment in report: approved

Section 2.4 (SAR): Nuclear Classification and Quality Requirements
- Definition of classification criteria and establishment of classes for
  - pressure-retaining and activity-carrying systems,
  - HVAC systems,
  - hoists and cranes,
  - structural steelwork,
- Assignment of systems to classes.
- Identification of quality requirements for classes.

Assessment in report: assignment criteria correctly selected; assignment of systems as correct as possible at the present status. Final assessment of assignment of systems and identification of quality requirements cannot be performed until construction licensing procedure.

Section 2.5 (SAR): Summary of Design Basis Events
- Listing of representative accidents by analogy with "Accident Guidelines for Pressurized Water Reactors".

Assessment in report: approved. Listing of all design basis events is complete, delimitation from hypothetical realm correct.

Section 2.6 (SAR): Postulates and Measures for In-Plant Events
- Break postulates:
  Primary system: one DN 65 connecting line (2A)
  Secondary system: main steam or feedwater line (2A)
  Steam generator tubes: one tube (2A)
- Concurrent main steam line and steam generator tube rupture not postulated

Assessment in report: approved. Requirement: ISI of steam generator tubes

Section 2.7 (SAR): Postulates and Measures for External Events
- Building design for earthquake:
  - Reactor building
  - Reactor building annex
  - Switchgear building
  - Reactor auxiliary building; only sealed concrete pit and its main load-bearing structures
- Building design for aircraft crash, blast wave:
  - Reactor building
- System design for earthquake, aircraft crash,
  - Pressure vessel unit
  - Steam generator tubes
  - Reactor coolant piping as far as isolation valves
  - Secondary system inside reactor building
  - Remote shutdown station
  - Components of reactor protection system inside reactor building
  - Shutdown systems inside reactor pressure vessel
  - Cavity coolers
- System design for earthquake:
  - Secured closed cooling system
  - Secured service water system
  - Reactor protection system
  - Emergency power systems

Assessment in report: approved.

Forschungszentrum Jülich GmbH
Research Centre Jülich

Appendix

Know-how on the Pebble Bed HTR owned by FZJ being of Relevance for the PBMR-Project of ESKOM

Contributions of the Research Centre Jülich to the PBMR-Project of ESKOM

PBMR = Pebble Bed Modular Reactor
ESKOM = a Utility of Republic of South Africa

Compilation: Heiko Barnert

3. February 2000

Remark:
This Appendix is an appendix to the document with the title:
"General Working Programme FZJ/PBMR", dated 2. February 2000, consisting of 22 pages.

Know-how on the Pebble Bed HTR owned by the FZJ

Introduction

The Forschungszentrum Jülich GmbH, FZJ, (Research Center Jülich) is in its Research Programme for „Safety Research and Reactor Technology" performing fundamental research work on safety aspects of innovative future reactor systems. The objective of the work is to contribute to solutions for the realization of a catastrophe-free nuclear energy technology. The Amended Atomic Energy Act of Germany requires that, even for events which can be practically excluded, the consequences remain confined to the plant so that, for example, evacuation is not necessary. Advanced reactor concepts therefore should satisfy three basic requirements for all cases of accidents:

- Self-acting limitation of nuclear power and fuel element temperatures,
- self-acting removal of afterheat from the reactor system, and
- self-acting maintenance of fission product barriers, mainly in the fuel element.

The background for this research work is that the Research Centre, Jülich, has been the research leader in the German Development Programme of the Pebble Bed High Temperature Reactor, HTR, for more than three decades. This Research and Development Work on the Pebble Bed HTR was done in co-operation with various German industrial companies. It is in this area of research and development that the Research Centre Jülich owns know-how in a number of fields of the technology of the HTR.

The R & D work was performed in the following institutes of the Research Centre Jülich:

Institute for Reactor Development,

Institute for Reactor Components,

Institute for Reactor Materials, including Hot Cells, and

Institute for Nuclear Chemical Technology, as well as

Institute for Nuclear Safety Research

with assistance of Central Institutes of Applied Mathematics, Electronics and Technology. Over the many years the R & D work was concentrated on the following projects:

Project: "High Temperature Reactor Fuel Cycle" (HTR- Brennstoffkreislauf, HBK), with the Fuel Pebble Mass Test Programme for THTR fuel (BISO fuel) and TRISO fuel at AVR and with the „JUPITER" (= Jülich Pilot Plant for the Thorium Extraction Reprocessing) and its operation demonstration at Research Centre Jülich.

Project: "AVR" (AVR = Arbeitsgemeinschaft Versuchsreaktor, Joint Working Group Experimental Reactor) financed via the Research Centre Jülich
with the Mass Test of fuel spheres and the Experimental Program at AVR on a

large number of topics, e.g. self-acting safety characteristics and operational behaviour of fuel and of the plant.

Project: "THTR-300" (THTR-300 = Thorium High Temperature Reactor 300 Mega Watt$_e$) with contributions to the technology and the licensing, in particular via fuel qualification by post irradiation examination work.

Project: "High Temperature Helium Turbine, HHT", with High Temperature Experimental Turbine Pilot Plant, HHV, 50 MWe input, and its operation demonstration.

Project: "Prototype Plant Nuclear Process Heat, PNP" (= Prototypanlage Nukleare Prozeßwärme) including the processes Hydrogen Gasification of Lignite and Steam Gasification of Hard Coal for the production of Substitute Natural Gas and Synthesis Gas, including HTR, design work for high temperature heat production with temperatures up to 1000 °C and the High Temperature Metallic Materials Programme.

Project: "Nuclear Long Distance Energy" for the medium distance transportation of high temperature heat from the HTR via the chemically reacting system $CH_4 + H_20 = 3H_2 + C0$, including the 10 MW scale Pilot Plant EVA/ADAM II:

Project: "High Temperature Reactor MODULE", with fundamental work to the qualification of TRISO coated particle fuel and to the licensing of the plant.

Project: "High Temperature Reactor Plants (HTR-Anlagen, HTA)", with R & D contributions reactor components, primary loop components, safety, including fission product behaviour, materials development, fuel elements, graphite, and final disposal.

Project: "AVR II" for the demonstration of HTR process heat application at temperatures of up to 950 °C with the Methane-Steam-Reforming process with a small HTR-Module (not realized).

Project: "AVR-Reconstruction" for the demonstration of HTR process heat application at temperature of up to 950 °C with the Methan-Steam-Reforming process using 50 % of the hot helium (950 °C) of the operating AVR beside ist electricity production (not realized).

Project: "Pre-Stressed Cast Steel Vessel Experimental Plant" 2.5 m outer diameter, 3 m outer height, 60 bar helium pressure for the demonstration of the pre-stressed vessel technology at Institute of Safety Research and Reactor Technology (former Institute for Reactor Development) of the Research Centre Jülich.

The above summarizes the R & D work on the Pebble Bed HTR done in the Research Centre Jülich in a broad overview. Therefore this summary is not complete and special topics may be addressed later and than be added.

The co-operation between the Research Centre Jülich GmbH and the industrial partners was ensured via Working Circles with representatives from participating parties.

List on Topics of Know-how on Pebble Bed HTR Technology owned by the Research Centre Jülich

1) Installations

The Research Centre Jülich operates a material test reactor, hot cells, intermediate spent fuel storage facilities, and low active waste storage facilities.

2) Safety and Licensing

The Research Centre owns know-how in the field of safety of nuclear plants due to the participation of heads of institutes in the German Reactor Safety Commission and due to review work, dialogue and work for license authorities and other governmental bodies as well as the public in general via parties and other similar institutions and by public information courses. Special know-how has been developed by contributions to the licensing procedures of the THTR-300, of the HTR-Modul and other HTR projects. Special experiences have been gained in the project "AVR Reconstruction for Process Heat Applications Demonstration" with heat in the form of hot helium of 950 °C with the result of a positive recommendation for licensing, given by an Advisory Council, consisting of member of the Reactor Safety Commission, established by the Federal Minister of the Interior of the Federal Republic of Germany.

3) HTR Plant Concept Realization

The Research Centre has developed and owns know-how on the development of concepts for the HTR under the headlines of electricity production via steam cycle, breeding, using the thorium fuel cycle, high temperature heat production for electricity production using a direct gas turbine cycle and for process heat applications. A good proof for that was the successful increasement of the mean outlet temperature of the AVR to 950 °C in 1974 and the successful demonstration of the high efficient product retention capability of TRISO fuel for gas turbine and process heat applications.

4) HTR fuel development and Operational Behaviour

The Research Centre Jülich was involved in the development of pebble type fuel elements for the HTR concept, together with German Industrial Companies, and owns know-how in particular in the "hot qualification" by post-irradiation examination experimental research and development work, as well as in the processes of kernel fabrication, and coating of coated particles. The objective of the respective research and development work was to increase the retention capability of the coated particles and to be able to operate the fuel fabrication under remote conditions. The base for the high quality of the fuel pebbles were the fuel mass test program on BISO fuel for THTR and on TRISO fuel for advanced applications in the AVR. Additional know-how was collected by the R & D work for and the operation of the JUPITER plant (= Jülich Pilot Plant for the Thorium extraction reprocessing), intended for reprocessing of THTR fuel, but not long-term applied (political reason).

5) AVR operational and experimental results

The Research Centre owns the results and the respective know-how of the 21 years of operation and of the experimental programme performed all that time. Of particular importance is the know-how from the 21 experimental programs performed in the last 3 years of operation to finalize the experiments at AVR with international participation. The topics in that program were:

- Stationary and dynamical reactor physics,
- hot temperature coefficient of the reactivity,
- production of plutonium in selected fuel pebbles,
- loss of coolant accident,
- release of fission products by depressurization (not finished to end of 1988),
- neutron and gamma fields,
- maximum outlet temperature of core,
- combined thermoelement and noise thermometer,
- instruments back fitting, hot gas sampling VAMPYR I, and
- plate-out-loop VAMPYR II; dust production and remobilization,
- selective filtering for tritium,
- tritium measurements,
- gas cleaning by gettering,
- input specific impurities, as well as
- mass tests of fuel elements for THTR for highly enriched TRISO coated $(ThU)O_2$-particle fuel pebbles, and for low enriched TRISO coated UO_2-particle fuel pebbles.

6) Reactor Ceramic Materials

The Research Centre Jülich has performed Research and Development Work in the field of ceramic high temperature materials for the production of high temperature heat in the form of hot helium with temperatures up to 1 000 °C as there is the oxide fuel with uranium as well as thorium, the coatings of the coated particles, that is the pyrolytic carbon layers, and the silicon carbide layer, as well as buffle layers. In addition there is the core structural high temperature ceramic material, namely graphite in the various qualities. The base for the developed know-how on that materials was the fact that all post-irradiation examination and hot qualification research and development work was done in the Research Centre Jülich.

7) Reactor High Temperature Metallic Materials

The Research Centre Jülich has performed Research and Development Work on high temperature metallic materials for the application of high temperature heat in the form of high temperature helium with temperatures up to 950 °C. Topics were to meet conditions of HTR-helium, and for a number of metallic materials meeting at the same time additionally process gas conditions on the other side of a wall structure. Topics were: helium corrosion, process gas corrosion, reduction of permeation of hydrogen and tritium, a large number of experimental proof tests on structural characteristics, as e.g. low cycle fatigue. Examples of materials are:

X10NiCrAlTi3220	(Alloy 800),
NiCr22Mo	(Nimonic86),
NiCr22Fe18Mo	(Hastelloy X), and
NiCr22Co12Mo	(Inconel 617), as well as

AC66 for the steam coal gasification process helium heat exchanger.

8) Reactor Components

The Research Centre Jülich owns know-how from R & D work on reactor components. Examples are:

Pebble bed configuration: Pebble flow mechanics, hydrogenamics, natural convection of gases in pebble beds,

Small absorber sphere systems: Flow behaviour, also inside the pebble bed, with respect to different limiters.

Heat transfer conditions: As for the pebble bed, and also for different heat exchangers, produced by the operation of a close to 1:1 scale gas loop inclusive circulators,

High temperature insulation material: Fibres and ceramics, including natural convection, depressurisation effects on insulations with demonstrated limiting pressure transients, e.g. 5 bar /s for PNP and 10 bar/s for HHT,

Pebble bed and fuel sphere accident simulation: Water ingress and air ingress causing corrosion

Concrete vessel heat-up accident simulation: Behaviour of the concrete and the included metallic material as well as the insulation, including cooling water refill procedures,

and so on.

9) Proof Test Experiments

The Research Centre has performed and is still performing proof test experiments in close to scale 1:1 facilities, to examine the feasibility of a future catastrophe-free nuclear energy technology, to demonstrate fundamental self-acting safety characteristics of the HTR concept, to use the experimental results for the validation of computer code systems and to facilitate future licensing procedures with the experimental proof test results. The proof test facilities are :

SARA : Self- Acting Removal of Afterheat,
NACOC : Natural Convection in the Core with Corrosion
SEAD : Self-Acting Separation of Droplets of Water, and
GRANIC : Granulate Injection into Core.

10) Nuclear Waste Management

The Research Centre has performed and is still performing Research Work on Nuclear Waste Management. The work focuses on

characterizing radioactive wastes,

treatment and storage of radioactive substances, scale 1:1 experiments,

radiochemical studies on waste partitioning for isotope transmutation, and

development of methods for quality assurance of the wastes to be disposed.

In addition to that know-how has been gained by the licensing of an experimental final disposal of HTR spent fuel spheres in a saltmine.

Concept of Inherent Safe Modular HTR

Kurt Kugeler
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Forschungszentrum Jülich
Institute für Safety Research and Reactor Technology (ISR)
Research Centre Jülich

24
~~28~~. July 2001

Requirements on the Safety of Future Reactors

Modified German Atomic Act

Amendent of the Atomic Energy Act as a further precaution against risks to the general public (7th Amendment to the Atomic Energy Act of 27.07.1994): The amended German Atomic Energy Act (1) (Art. 7, para. 2a) for future plants stipulates "that even such events whose occurence is practically excluded by the precautions to be taken against damage would not necessitate decisive measures for protection against the damaging effects of ionizing radiation outside the plant fencing,...". In the substantiation of the bill for parliamentary discussion, terms from the text of paragraph (2a) are defined in more detail. It is thus postulated that "accidents with core melt" are controlled and "evacuations" are not necessary.

Consequences:

- no evacuations necessary
- no relocations necessary

Requirements:

- Future nuclear energy uses (including waste disposal) must avoid serious radiological impacts outside the plant.
- Evidence (as integral as possible) must be provided for the safety behaviour of reactors.

Requirements on the Safety of Future Reactors

New reactors require a high degree of safety
$\cong$ catastrophe-free nuclear technology

Definition:

Catastrophe-free nuclear technology is realized, if the radioactive fission products in all possible accidents are retained nearly totally inside the reactor plant.

Consequences for the environment:

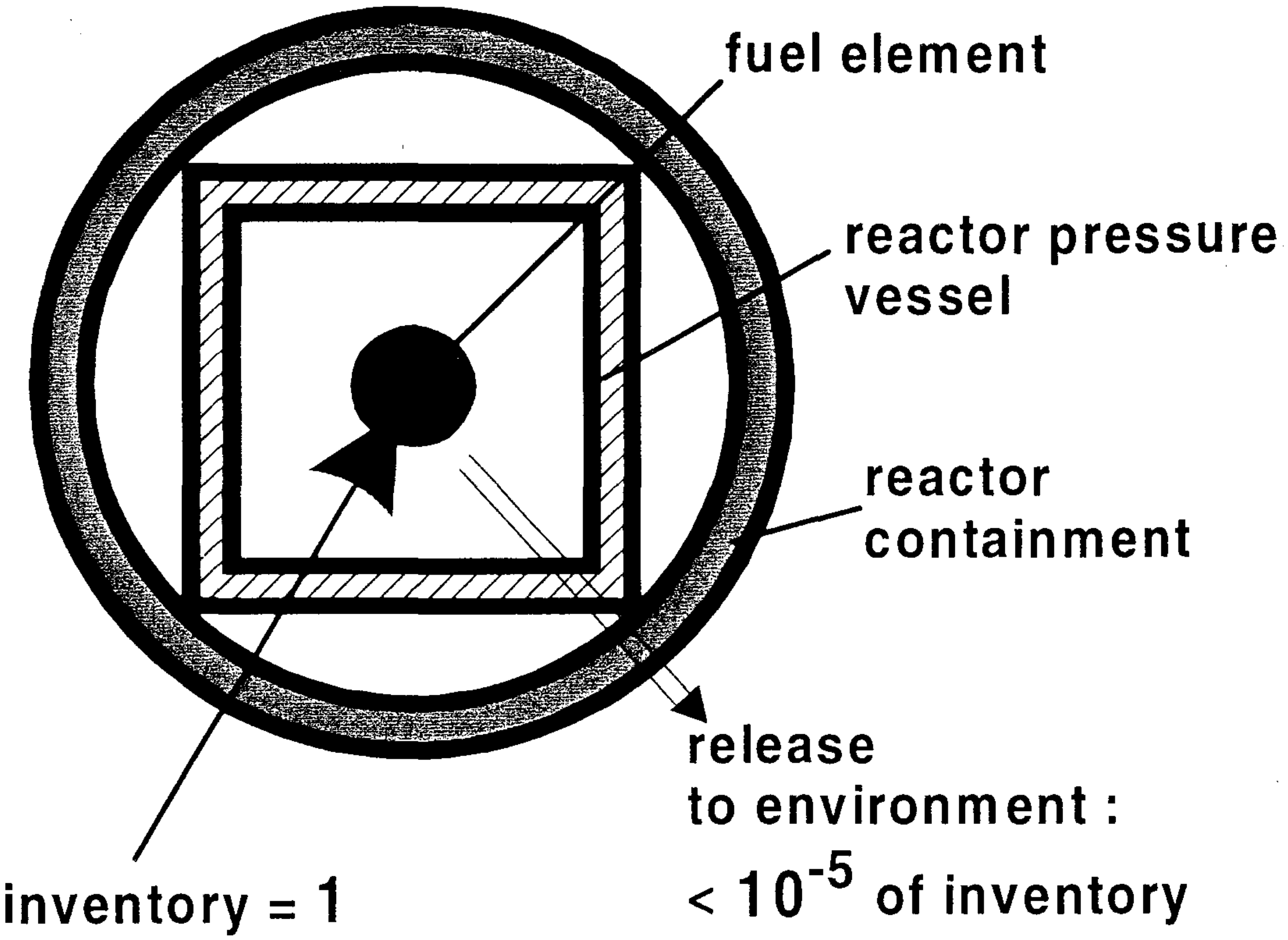
- **no immediated fatalities**
- **no evacuation**
- **no resettlement**

Area of Definition:

1. **for all accidents due to internal reasons: (failure of shut down systems, decay heat removal, loss of coolant, failure of components)**
2. **for all accidents due to external reasons: (earthquake, gas cloud explosion, air crash)**
3. **external accidents beyond class 2. require special consideration (sabotage, war, meteorite, extreme earthquake)**

Safety requirements of modular HTR

(following the modified German Atomic law from 1994)



Retention of fissile material and fission products inside the nuclear plant:

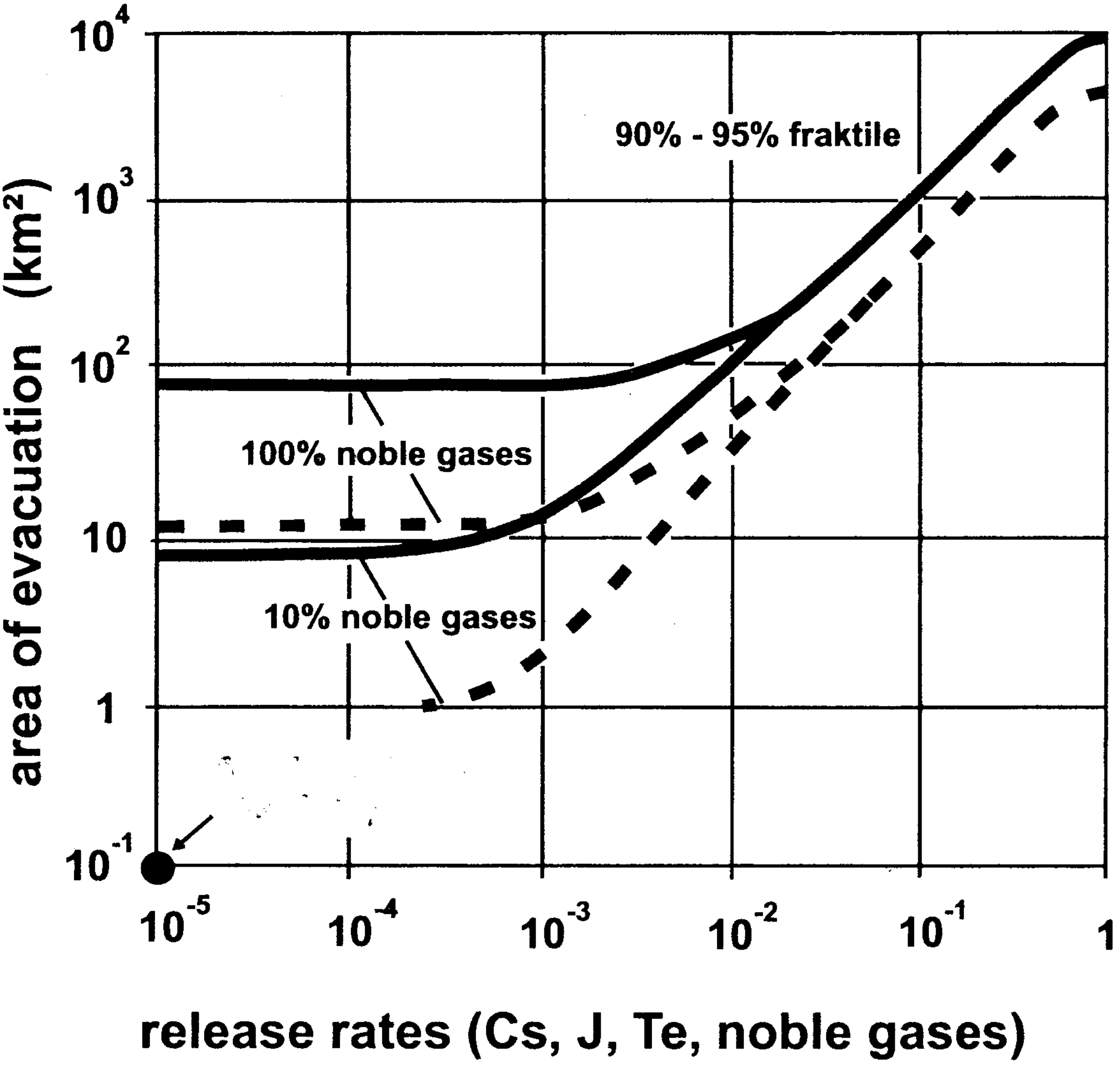
- no evacuation
- no resettlement
- no early fatalities

↓

„catastrophe-free nuclear technology“

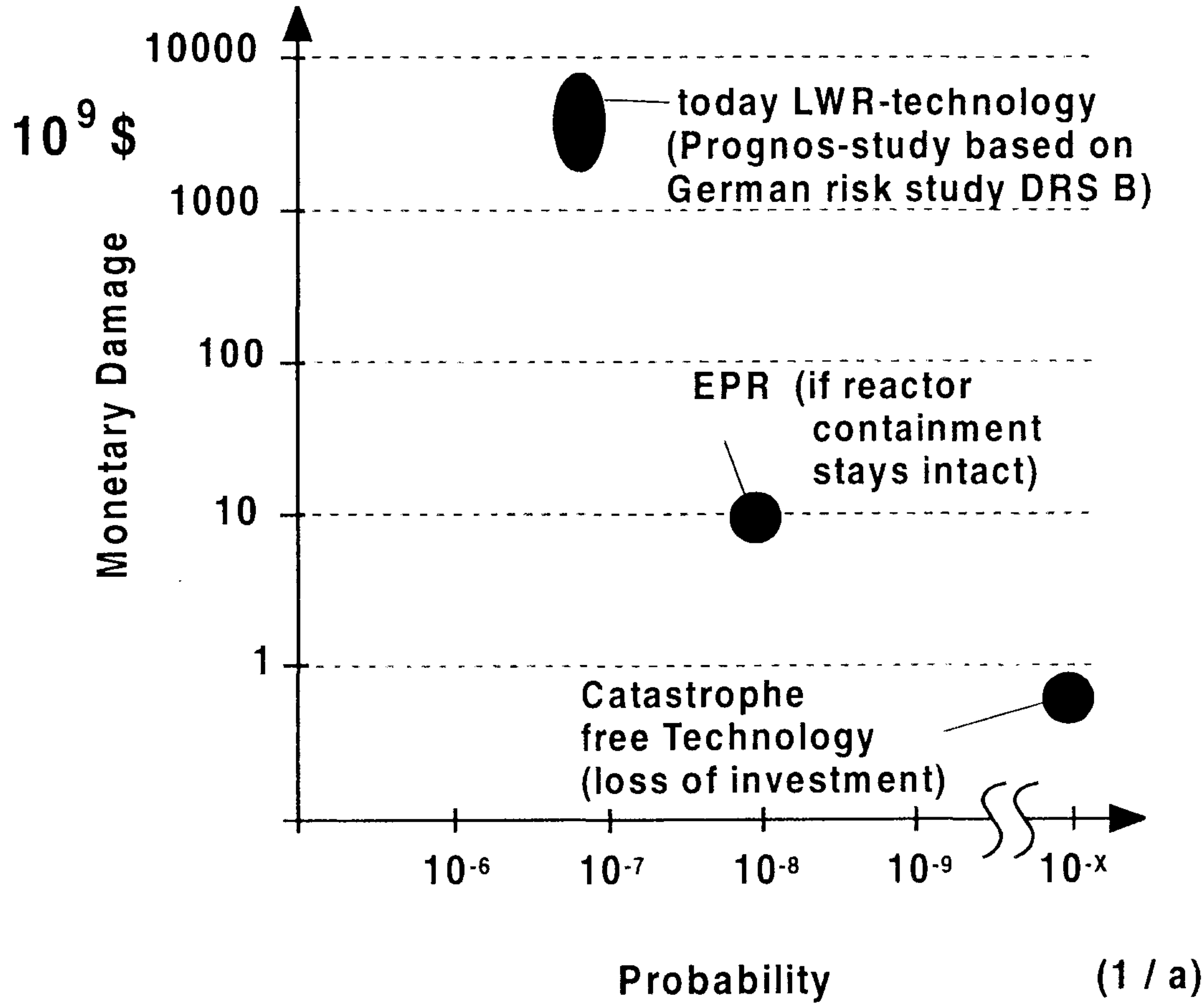
Requirements on the Safety of Future Reactors

Definition of catastrophe-free nuclear technology related to the release of fission products (cesium, iodine, tellurium) from the plant to the environment (question of resettlement):



Economic aspects of modular HTR

aspect: Monetary damage caused by severe accidents



- X is a very large number
- scientific work: how large is X?
  Where is a limit of consideration?

④ Safety Aspects of PBMR

Assumptions on accidents, limits of consideration

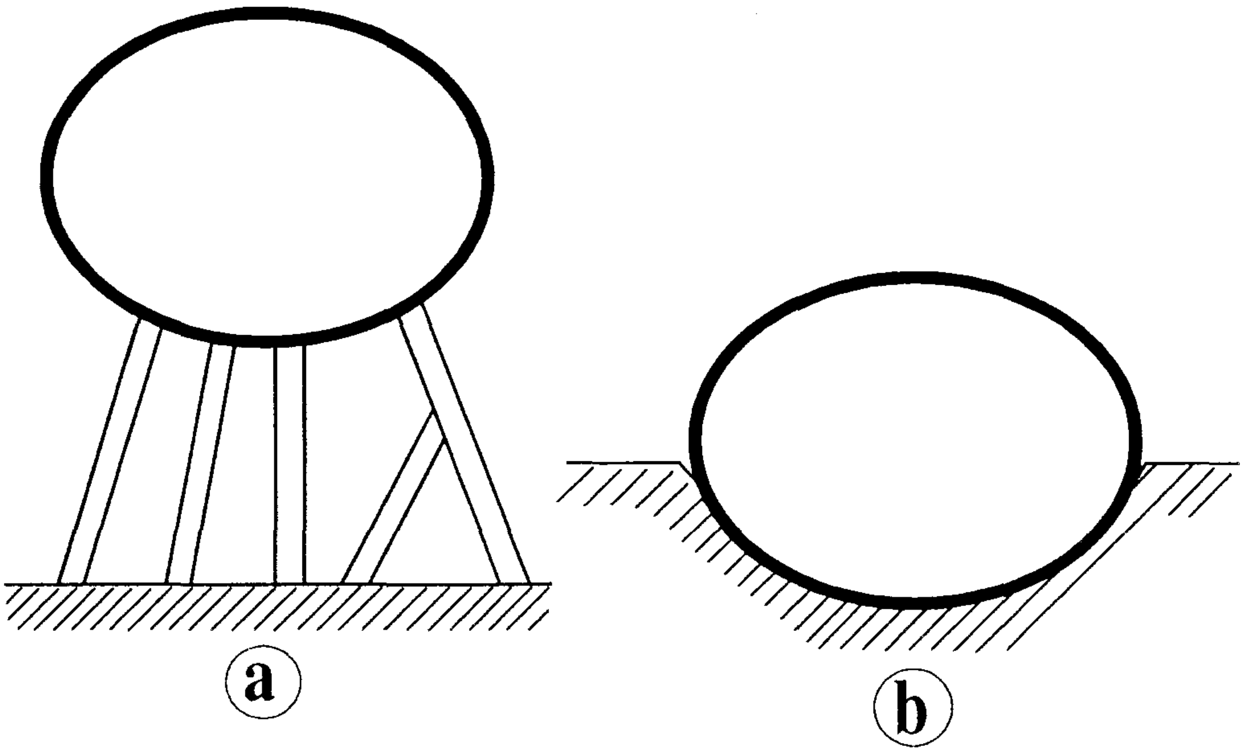
1. internal reasons
   - total loss of coolant
   - total failure of active decay heat removal
   - massive ingress of water to primary system
   - massive ingress of air to primary system
   - extreme reactivity accidents
   - massive failures of reactor components

2. external reasons
   - aircraft crash
   - gas cloud explosion
   - earthquakes
   - fire
   - tornados, hurricans
   - flood

---

3. external reasons (beyond conside-ration)
   - sabotage
   - war impacts
   - extreme earthquakes
   - meteorites

Inherent safety principle explanation:



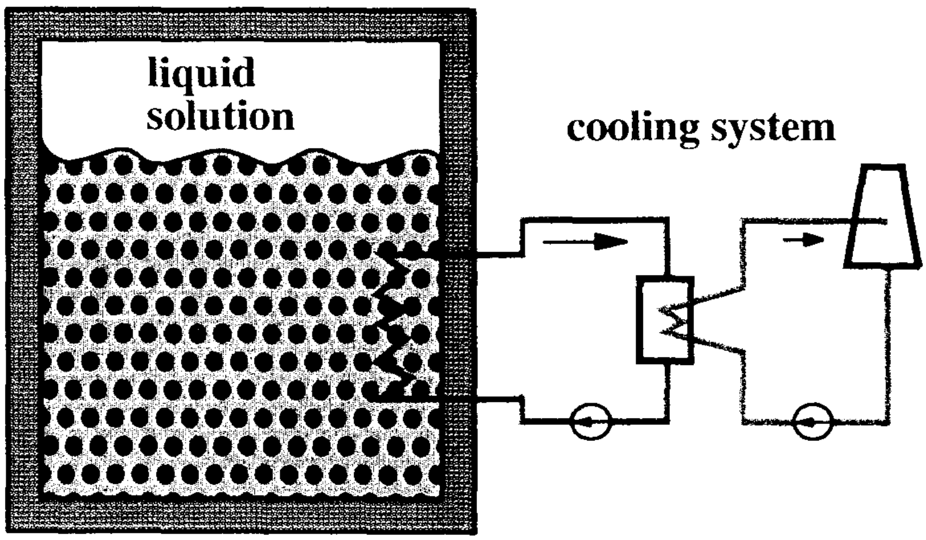
- system a is safe by engineering; probability of failure is larger than zero (but very small)

- system b is inherently safe; probability of failure is zero

- possible limitations for solution b : extreme earth quakes, war impacts

Requirements on the safety of future reactors

Intermediate storage of high level radioactive waste from reprocessing as an example of catastrophe free nuclear technology



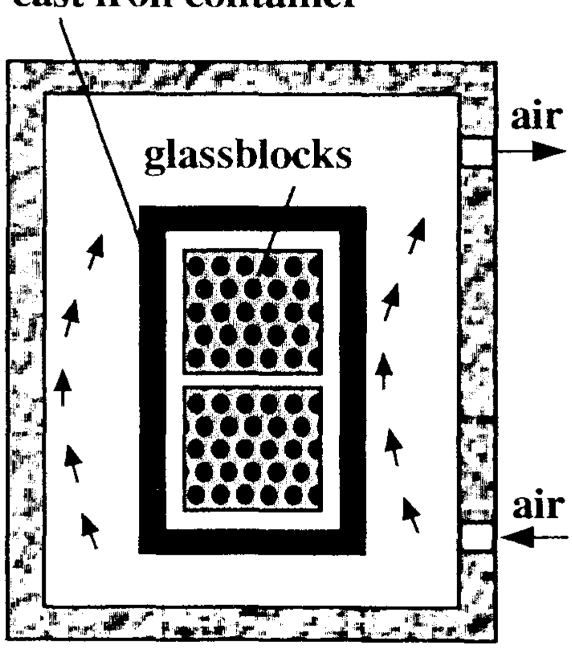
liquid solution in tanks

↓

active cooling (A)

↓

probabilistic safety concept



fission products in glassblocks and cast iron container

↓

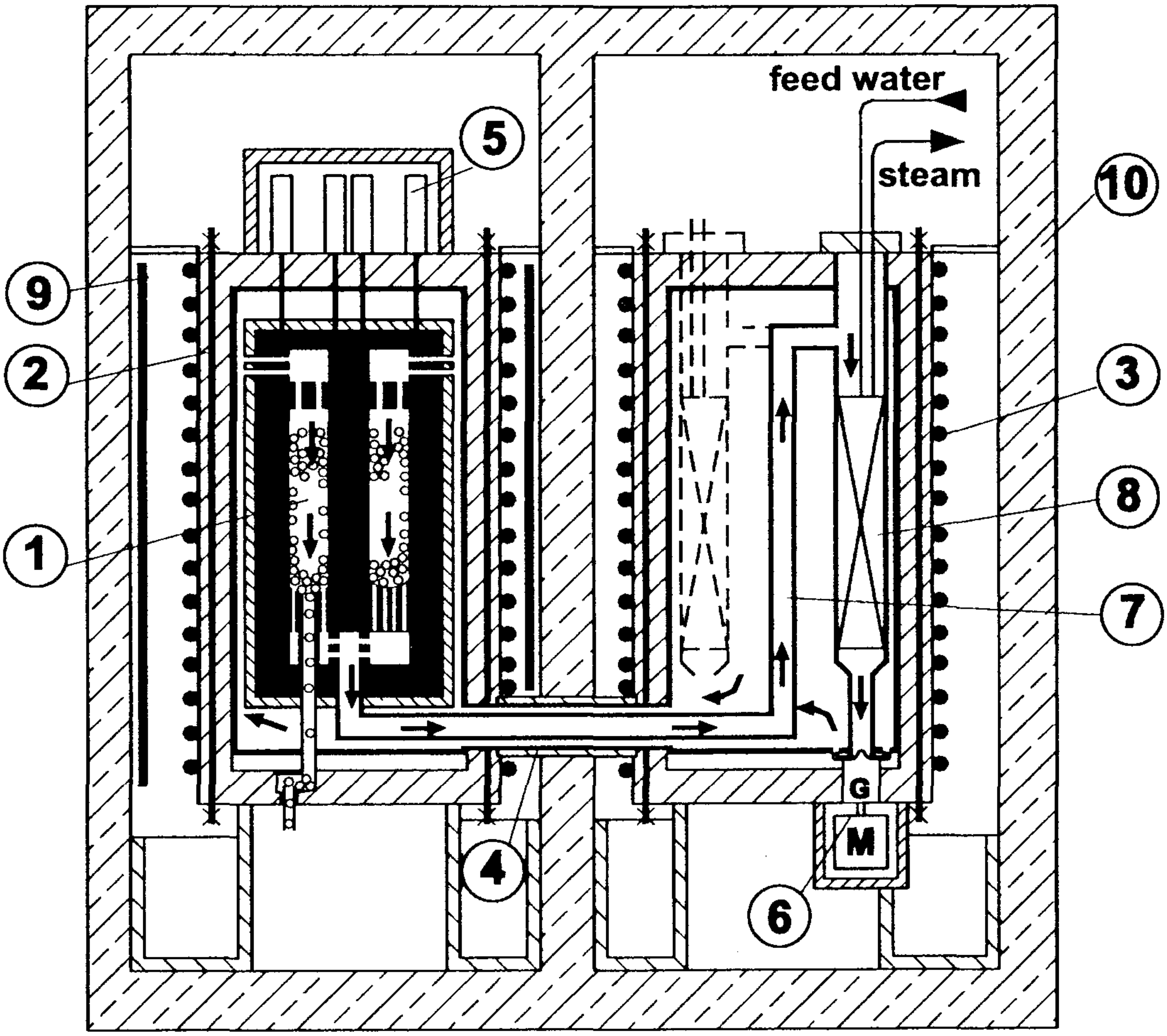
selfacting cooling (B)

↓

catastrophe free technology

→ no internal or external reasons for large fission product release in case (B)

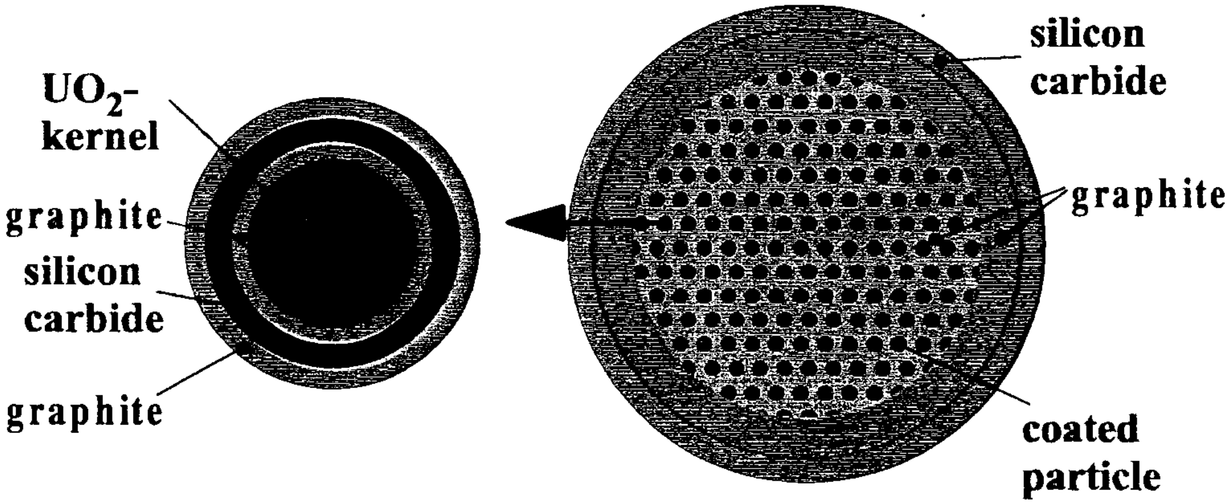
Concept of a "catastrophe free" nuclear reactor

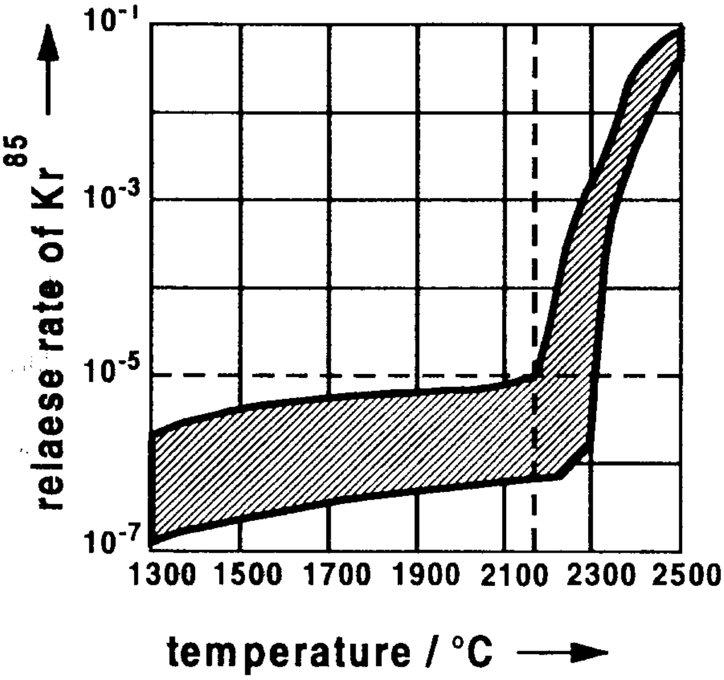


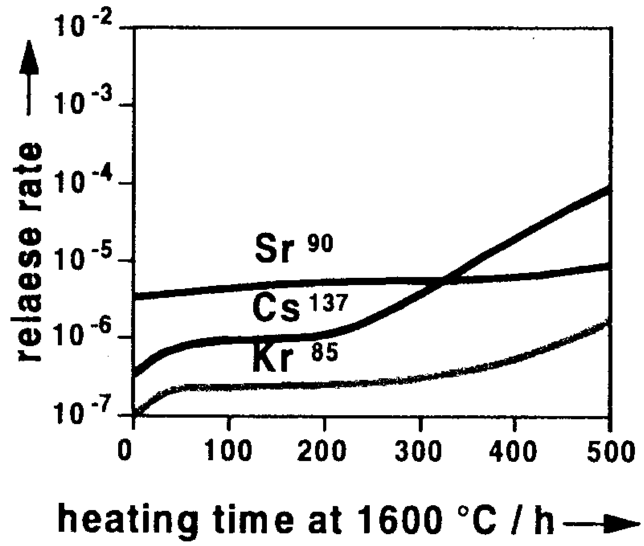
- data: 300 MW_{th}, 250 $\rightarrow$ 700 °C, annular core, TRISO - fuel, T_{fuel}^{max} < 1600 °C

- applications: steam cycle, gasturbine, combined cycle, cogeneration process

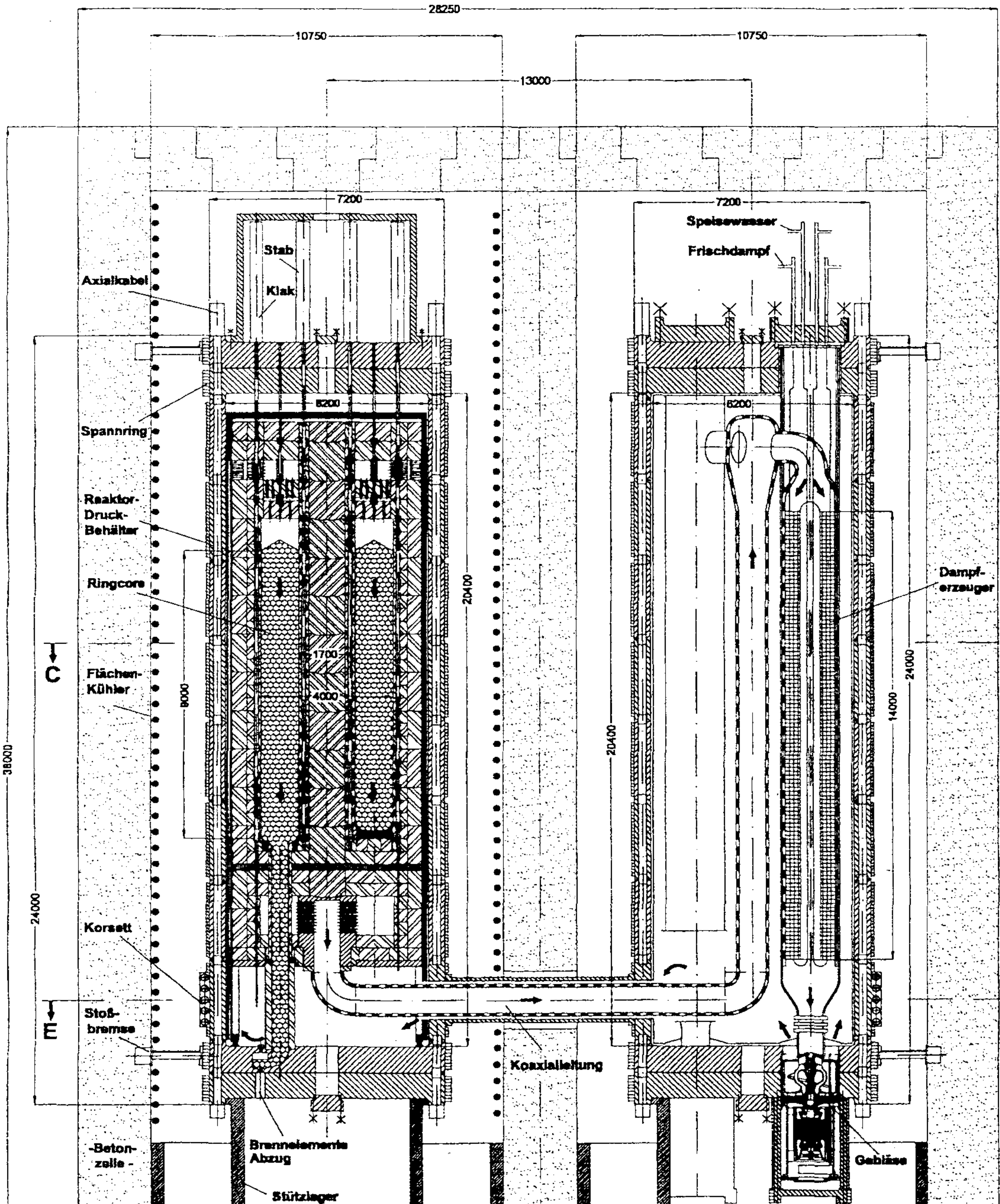
Pebble bed fuel and safety characteristics







- **Retention of fission products till 1600 °C proven**



Forschungszentrum Jülich

Institut für Sicherheitsforschung und Reaktortechnik

Jülich, 26.03.2001, M. Kugeler, H. Hohn

Entwurf

0 2 m

Kugelhaufen-Reaktor

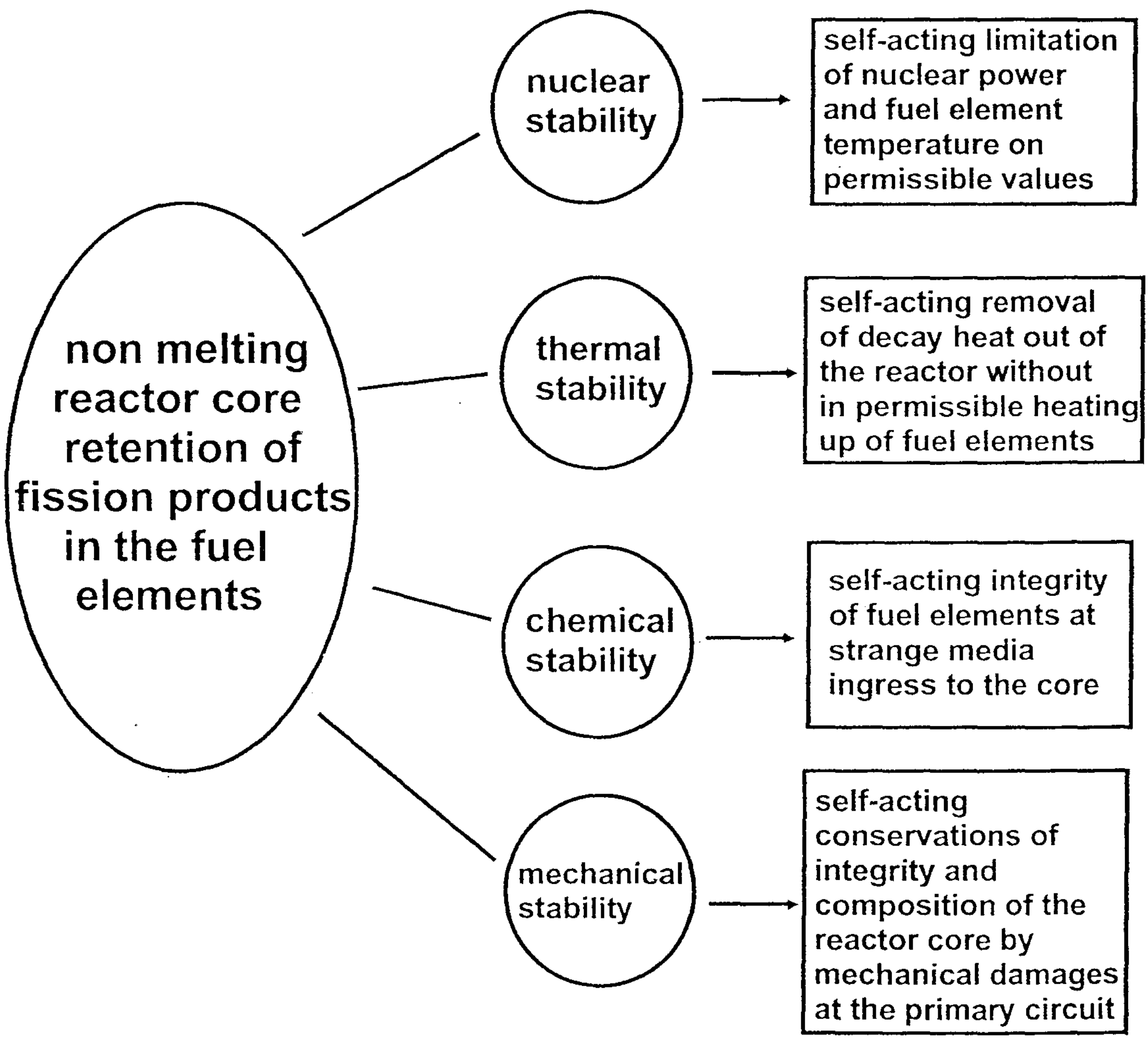
$N = 300$ MW$_{th}$ Heliumkühlung $p = 60$ bar, $T_{austritt} = 350°C$

Zweikreisanlage, 3 DE, MEDUL-Beschickung

Ringcore: $D = 4$ m, $d = 1{,}7$ m, $H = 9$ m, $L = 3{,}24$ MW/m^3

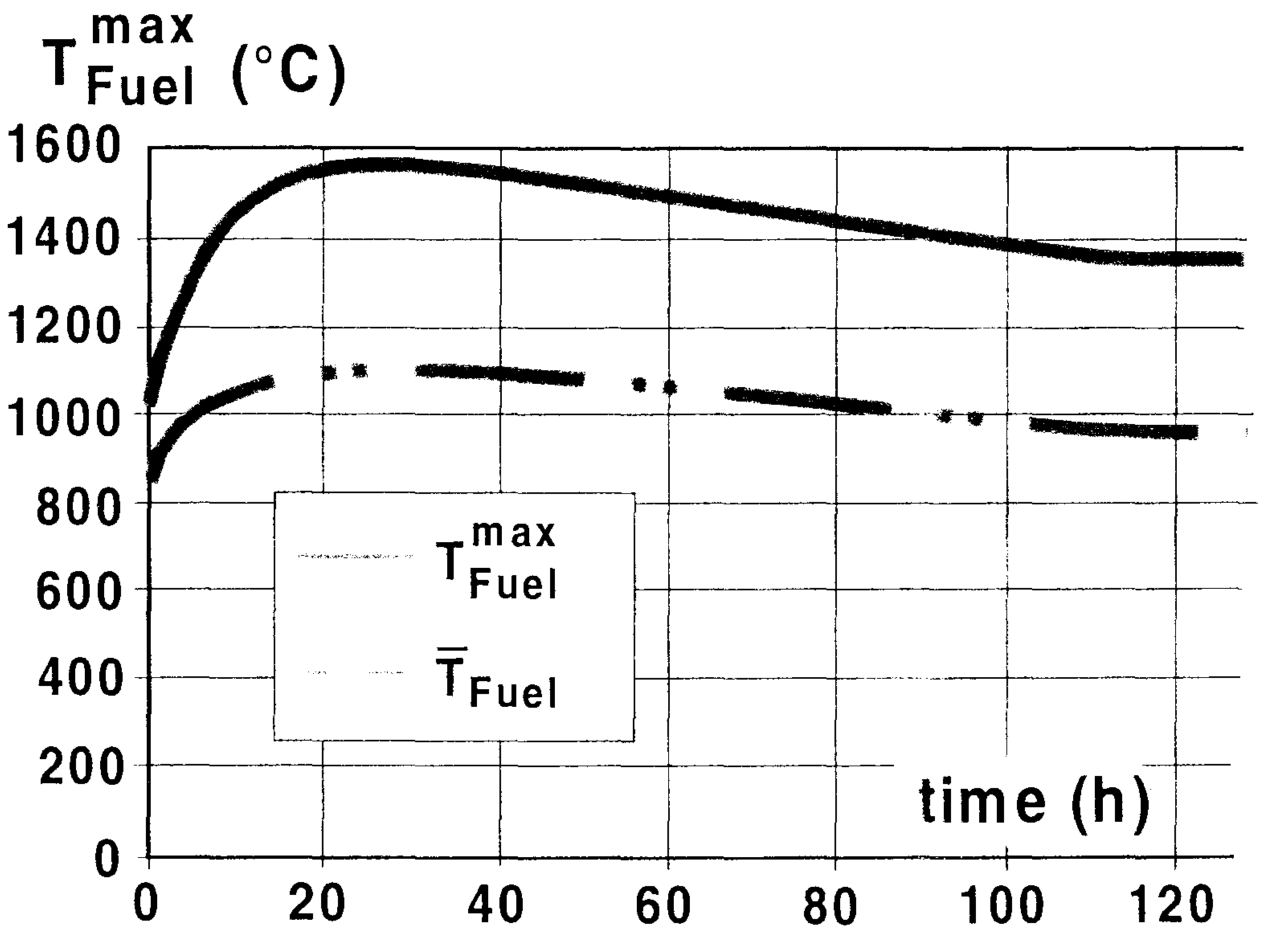
Reactors without core melt

Principles of stability as the conditions for reactors which never can melt or be impermissible overheated



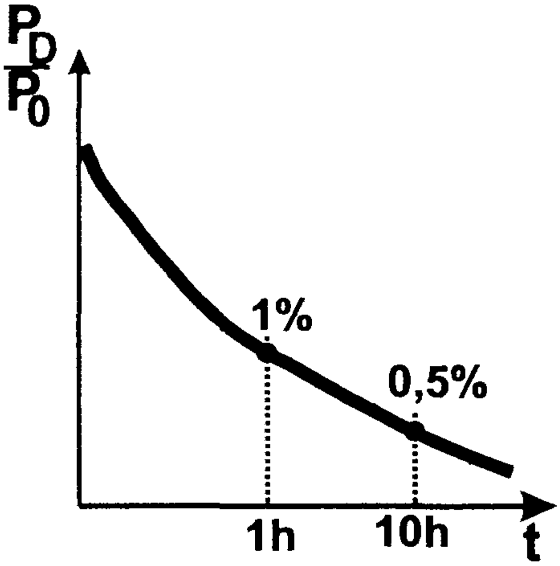
Selfacting decay heat removal in modular HTR – concept of PBMR

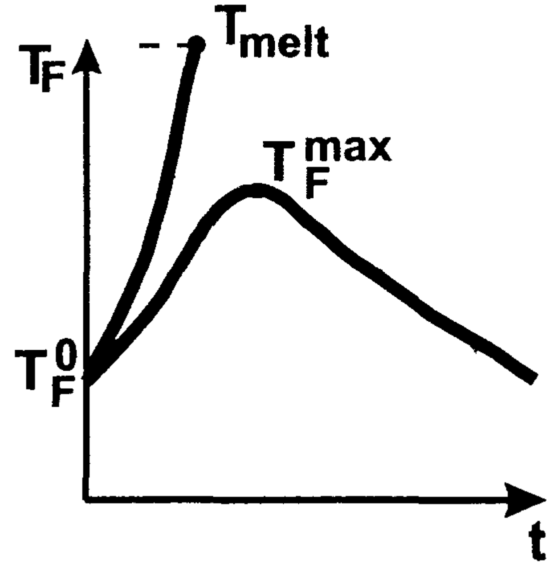
assumption: total loss of coolant + total loss of active decay heat removal + total loss of first shut down system



- core never can melt
- maximal fuel temperature stays below 1600 °C
- most fuel elements stay at much lower temperatures
- fission product release limited to < 10^{-7} inventory

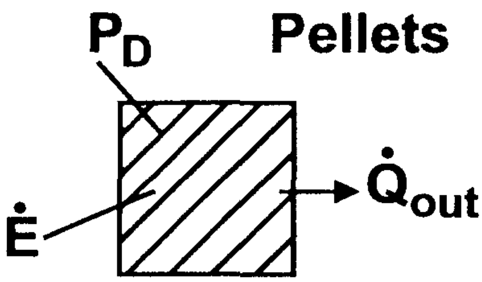
Behaviour of reactors at cooling accidents

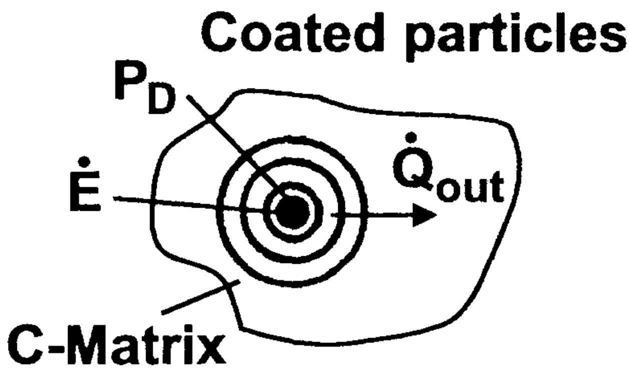




① • $\frac{d}{dt}\int_V \rho \cdot c \cdot T \cdot dV = P_D(t) - \dot{Q}_{out}(t)$

② • $T_F^{max} < T_F^{all}$





$P_D >> \dot{Q}_{out}$	$P_D \approx \dot{Q}_{out}$
$T_F^{max} = 2850°C$	$T_F^{max} < 1600°C$
in 1h --->	after 30h --->
a)	b)

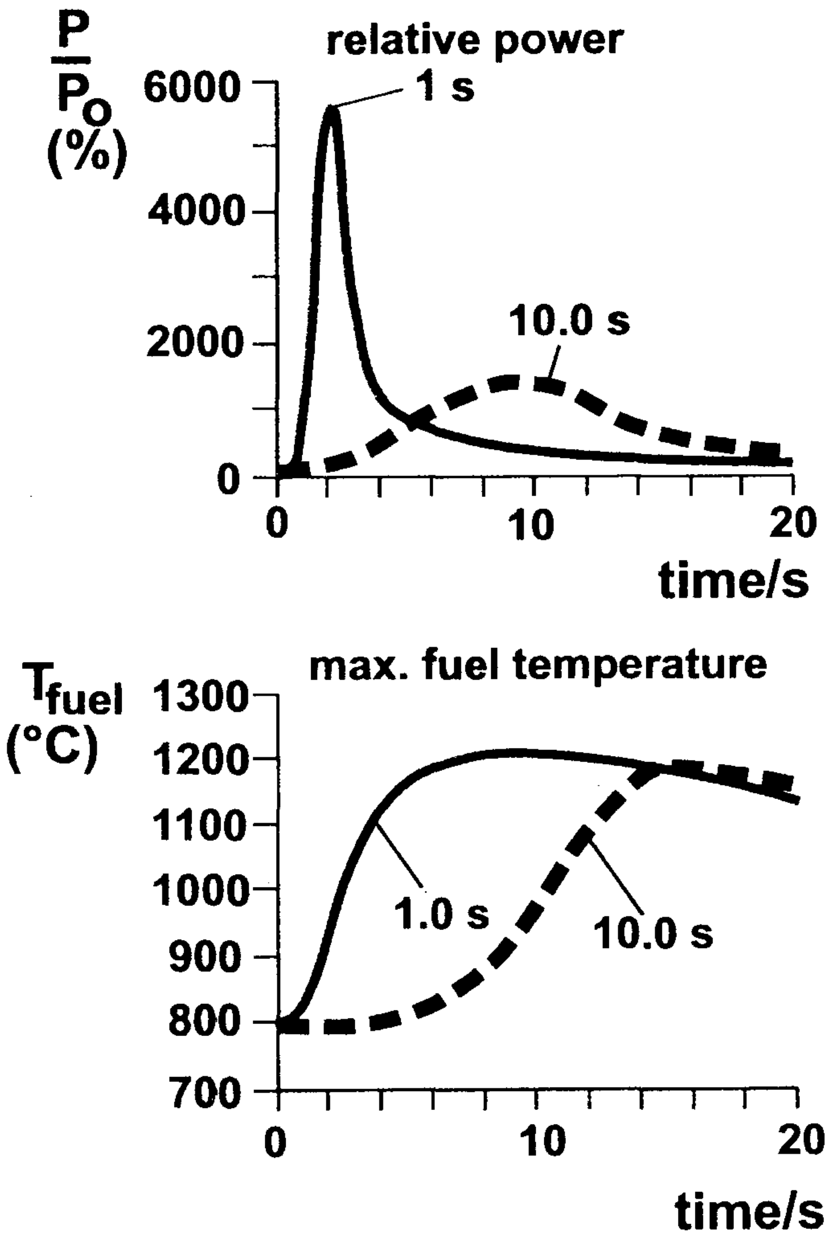
Differences between reactors which need active cooling (a) and reactors which use the principle of self-acting decay heat removal (b)

Behaviour of reactor at extreme nuclear transient

assumptions:

loss of all control rods ($\Delta k = 1{,}2\%$)

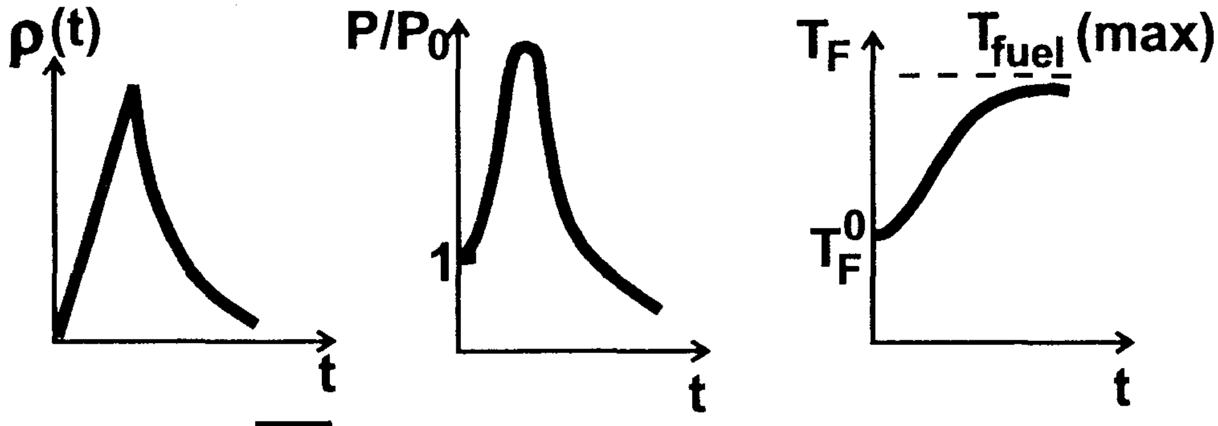
in very short time (1s, 10s)



consequences:

fuel temperatures stay always below 1600°C

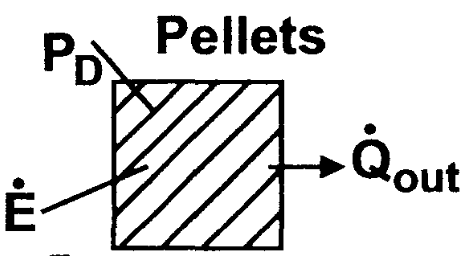
Behaviour of reactor systems at extreme reactivity transients

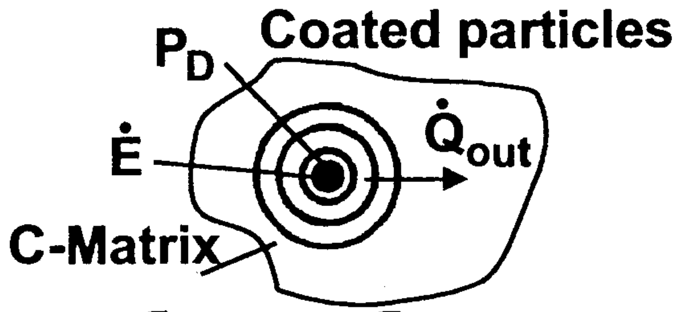


$$① \quad \rho = \rho_0 + \sum \Gamma_i \cdot \Delta x_i \qquad \Gamma_i \text{ always negativ}$$

$$② \quad \int_0^\tau \int_V \rho \cdot c \cdot T \cdot dV \cdot dt = \int_0^\tau P(t) \cdot dt - \int_0^\tau \dot{Q}_{out}(t) \cdot dt$$

$$③ \quad T_F^{max} < T_F^{all}$$





$$\int_0^\tau P(t) \cdot dt >> \int_0^\tau \dot{Q}_{out}(t) \cdot dt$$

$T_F^{max} = 2850°C$

in $\tau \sim 1s$

a)

$$\int_0^\tau P(t) \cdot dt \approx \int_0^\tau \dot{Q}_{out}(t) \cdot dt$$

$T_F^{max} < 1600°C$

after 10s

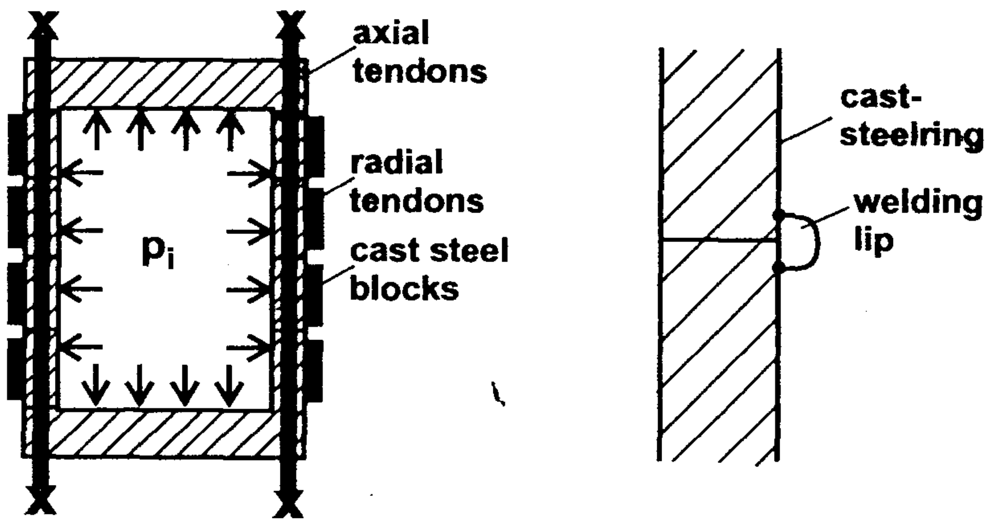
b)

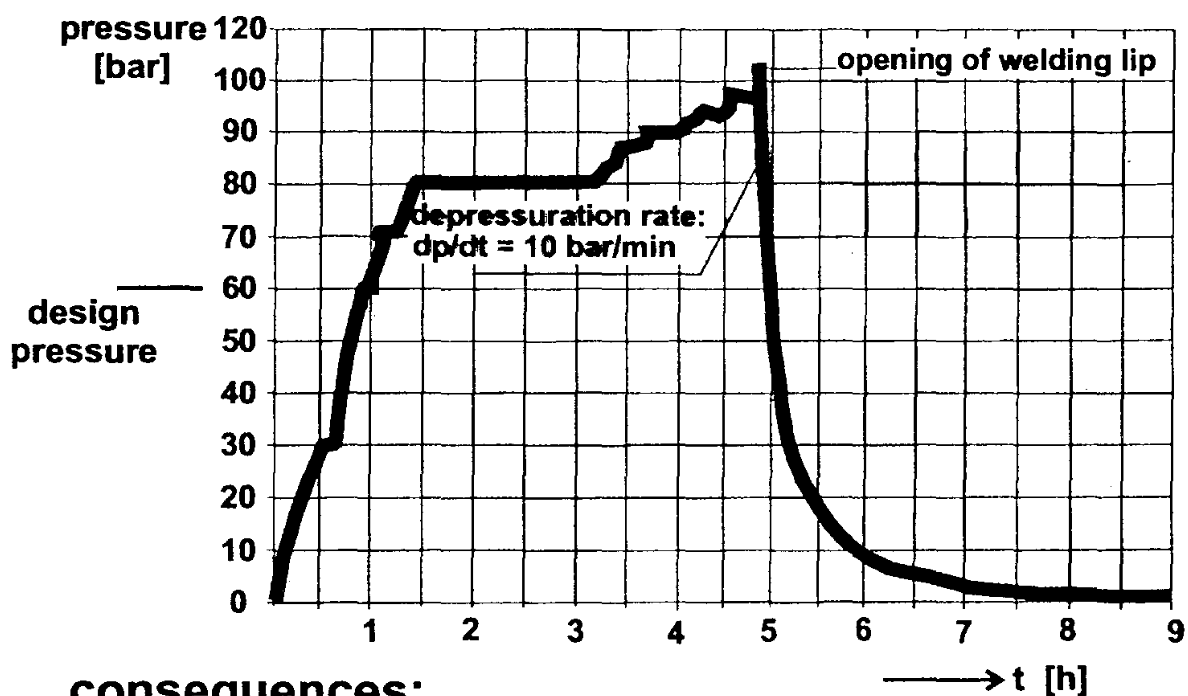
Comparison between the conditions of the fuel of LWR (pellets) and of pebble-bed fuel HTR (TRISO-coated particles) in the case of extreme nuclear accidents

Burst protected reactor pressure vessel

assumptions:

- reactor pressure vessel is totally pre-stressed by axial and radial tendons
- overpressure is governed by opening of welding lips



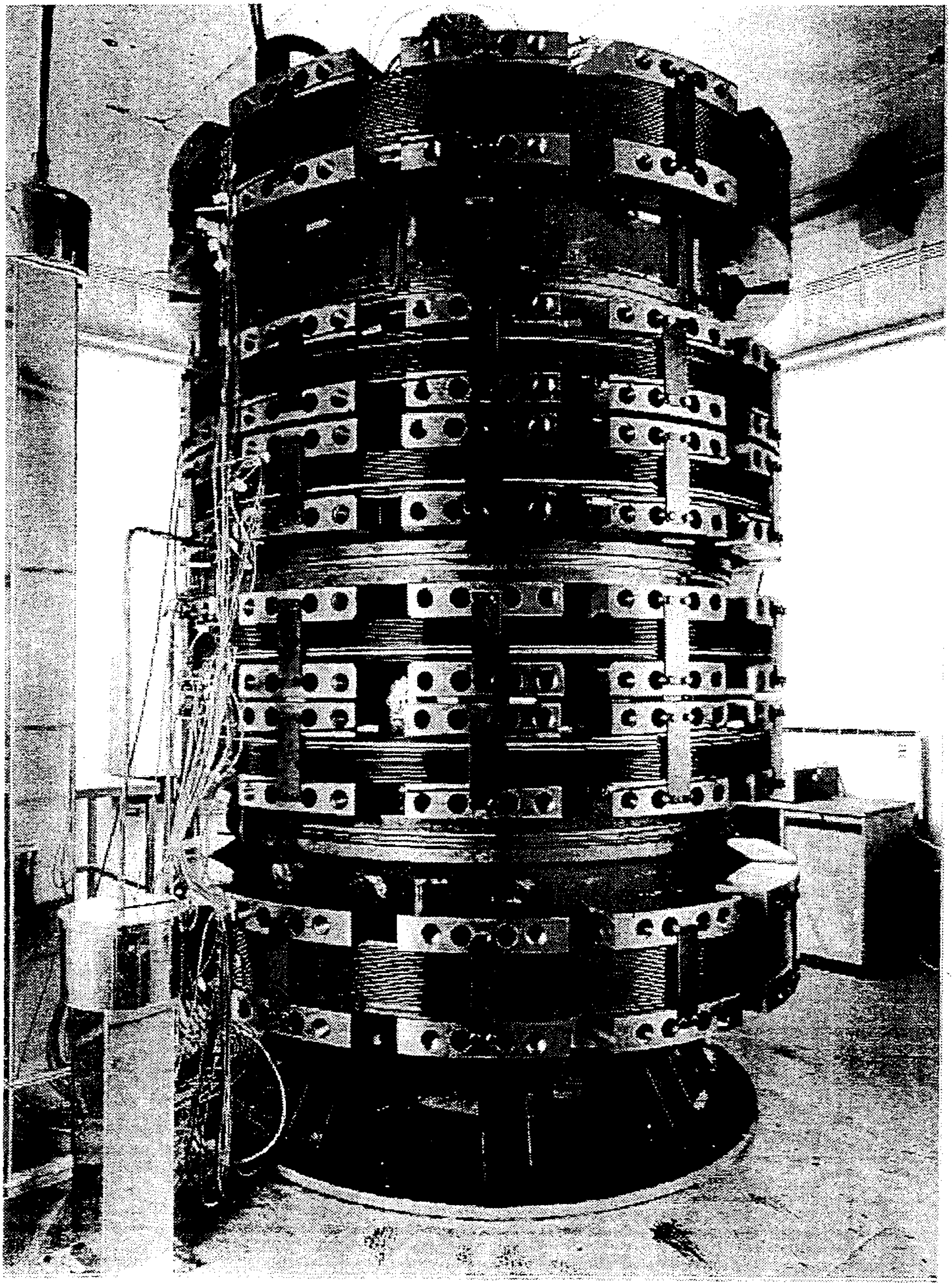


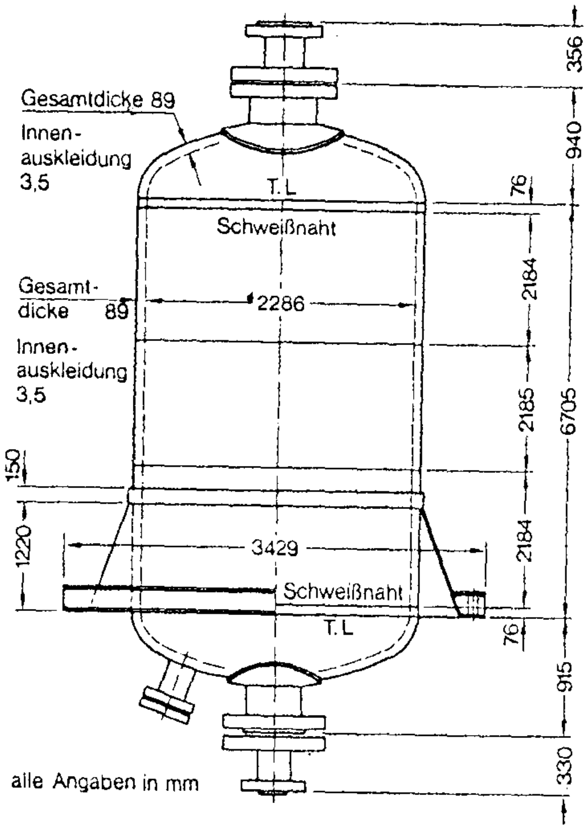
consequences:

- vessel cannot burst
- no large openings possible
- depressurisation rate very low

Mechanical Stability in HTR

Experimental vessel:(inner diameter: 1,5 m; inner height: 2 m; wall temp.: 300 °C; pressure: 60 bar helium; operation time: 30,000 h).
over pressure test: 120 bar: slow depressurization through leak





Schnitt durch den Reaktor der Tokoyuma-Anlage.

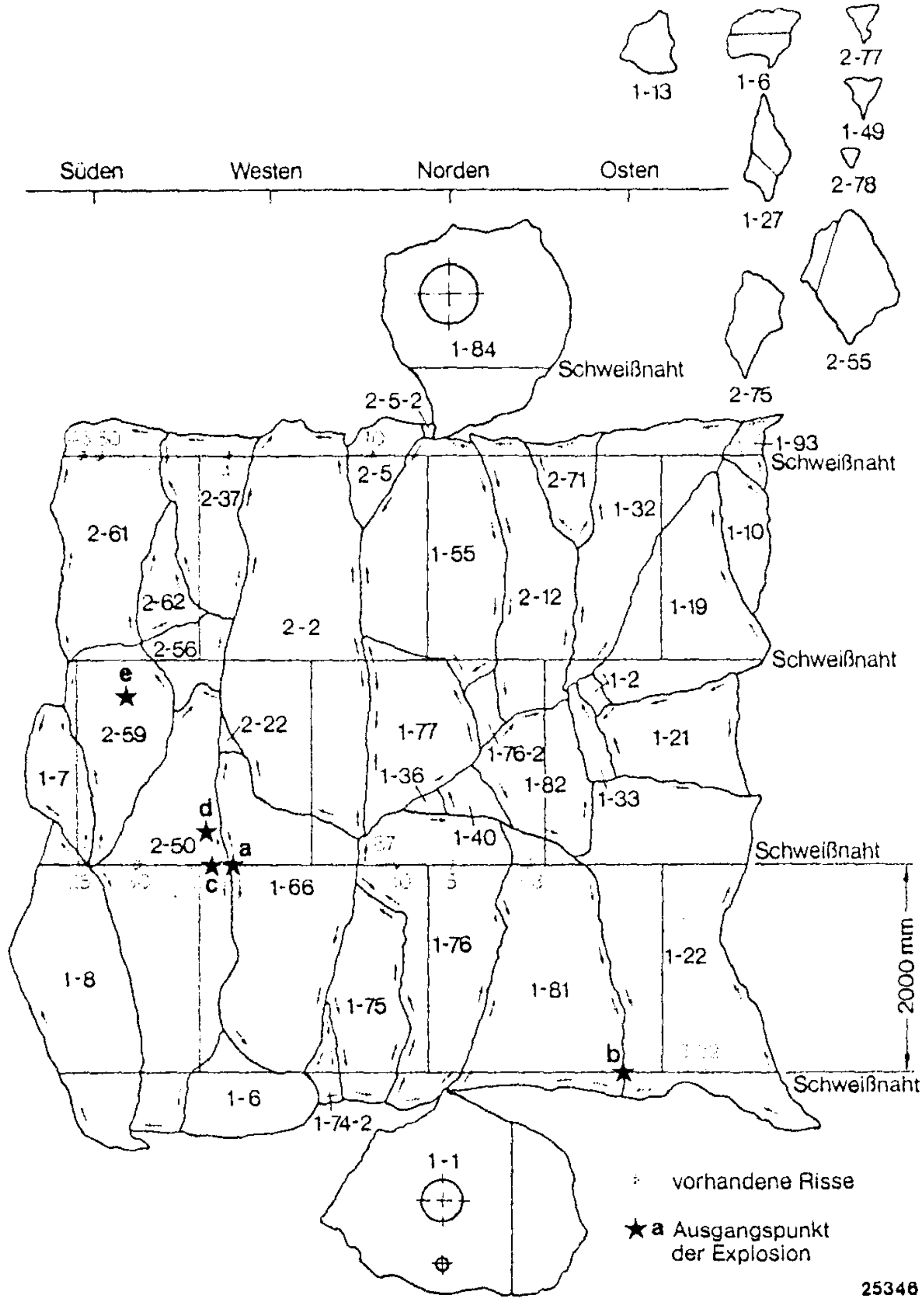


Bild 3. Rekonstruierte Anordnung der geborstenen Teile mit den vorhandenen Rissen (gelbe Markierungen) und dem Ausgangspunkt der Explosion (Markierung a). Die Pfeile deuten den Rißverlauf während der Explosion an.

$p = 65$ bar

$T = 450\,°C$

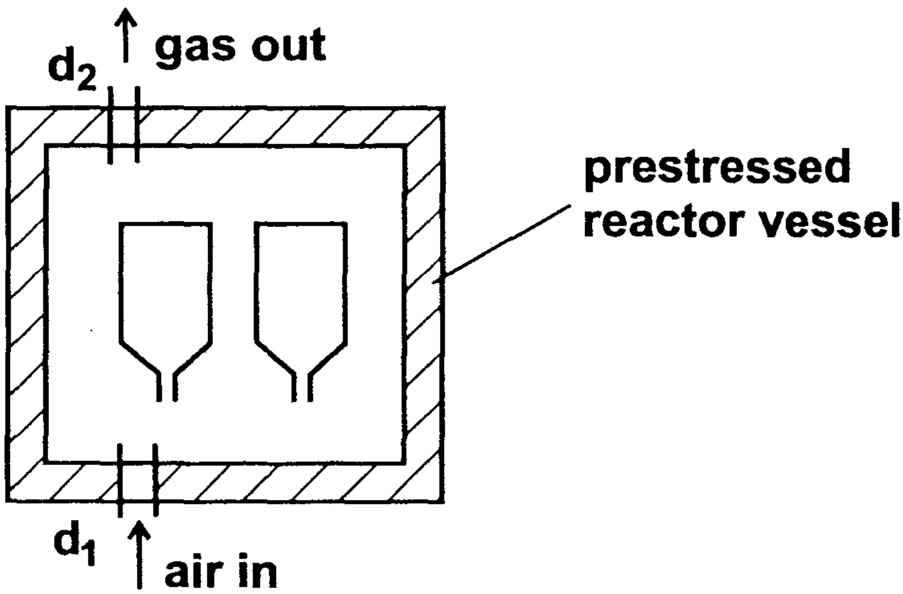
$s = 89$ mm

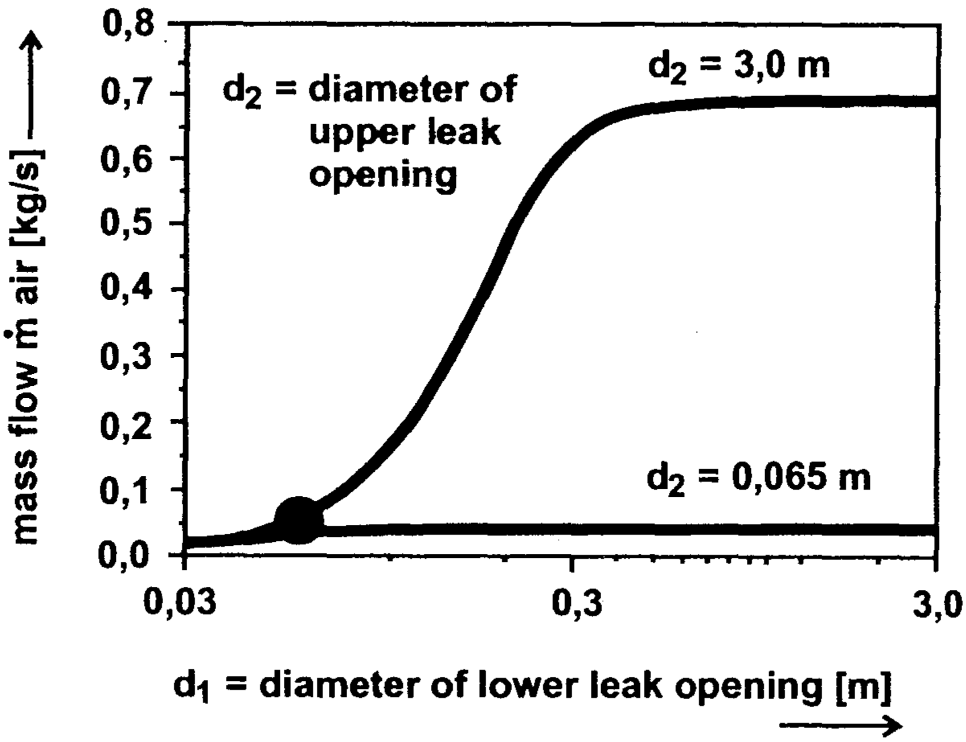
$p_{Prüf} = 55$ bar

Air ingress into the primary circuit

assumptions:

- primary system is pre-stressed
- two openings in primary system occur



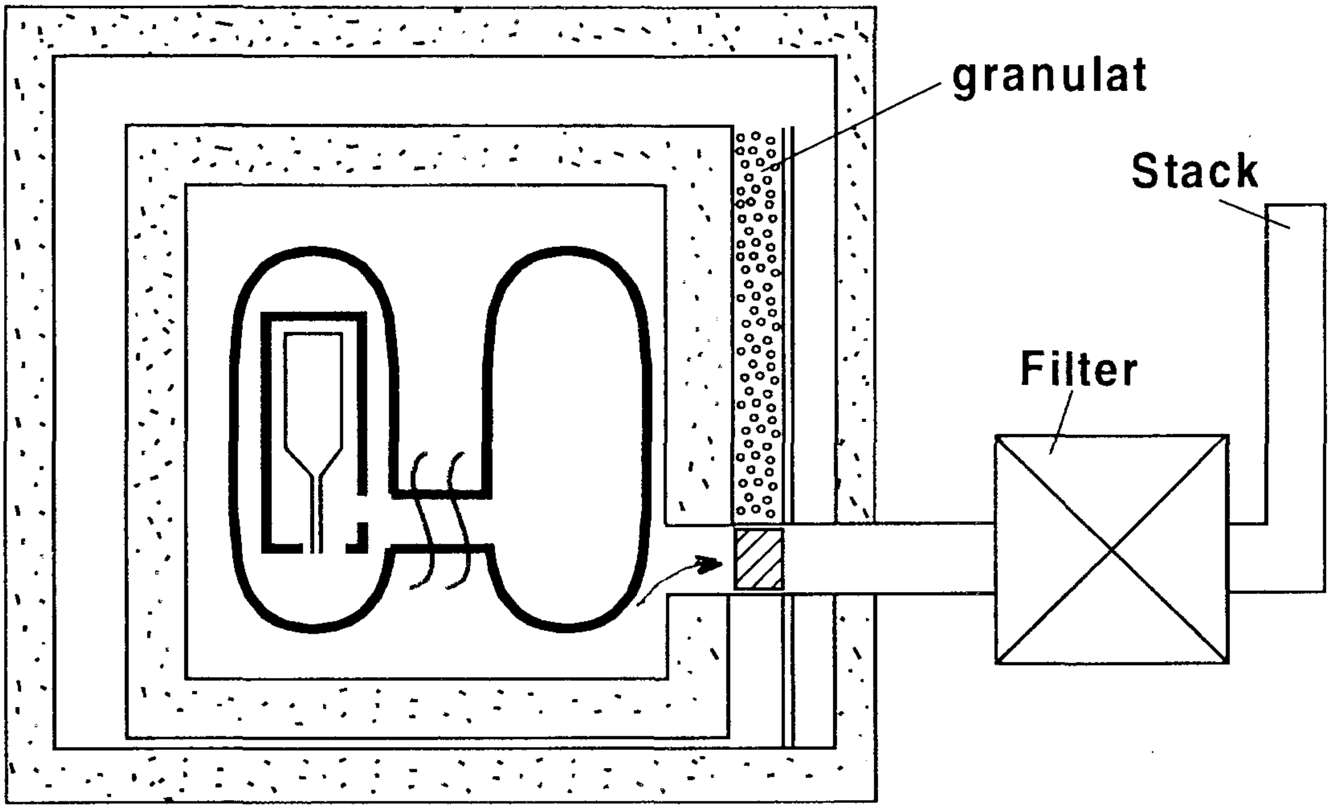


consequences:

- for the case d_1= 6,5 cm and d_2= 6,5 cm air ingress rate is very small (≙ 4 gC/s)
- prestressed reactor vessel allows small flow rates only

Control of air ingress

assumption: break of connecting vessel



- air content of inner concrete cell is limited ($< 5000\ m^3$): limitation of air by selfacting closing by sand, granulat

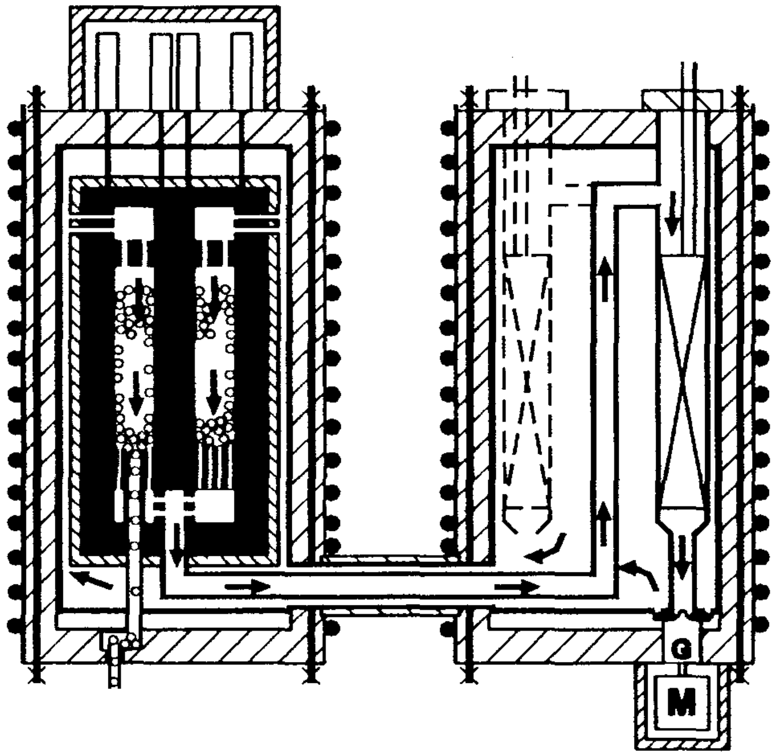
- simple intervention: openings on primary circuit can be closed by foam, sand, because reactor building is accessable even after a long time (protection of investment)

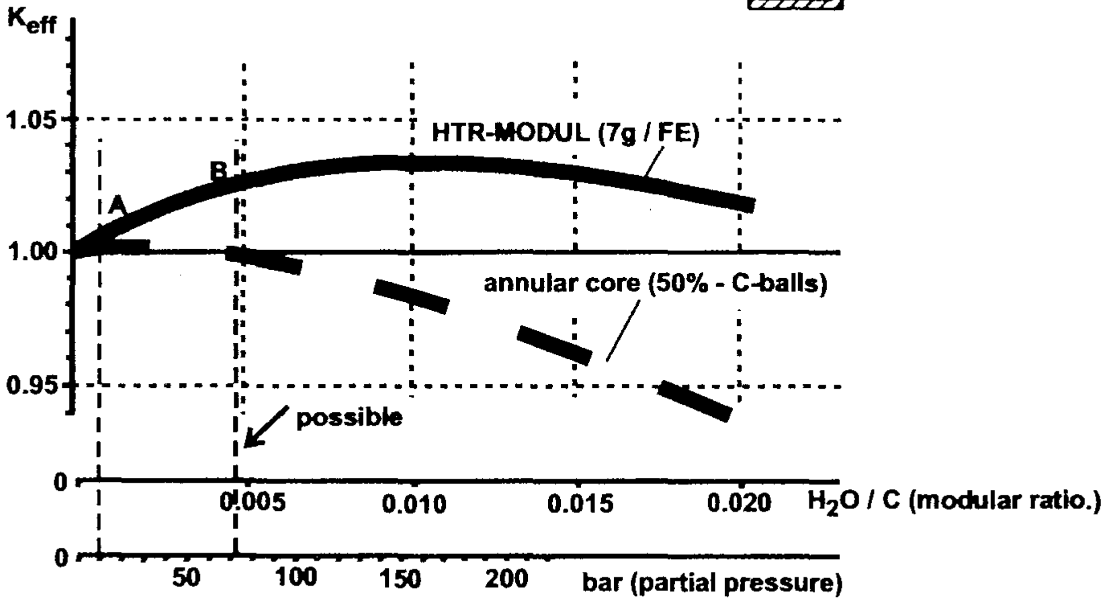
- burn up of graphit limited to less than 100 kgC

Water ingress into the primary circuit

assumptions:

- total inventory of secondary circuit enters the helium circuit

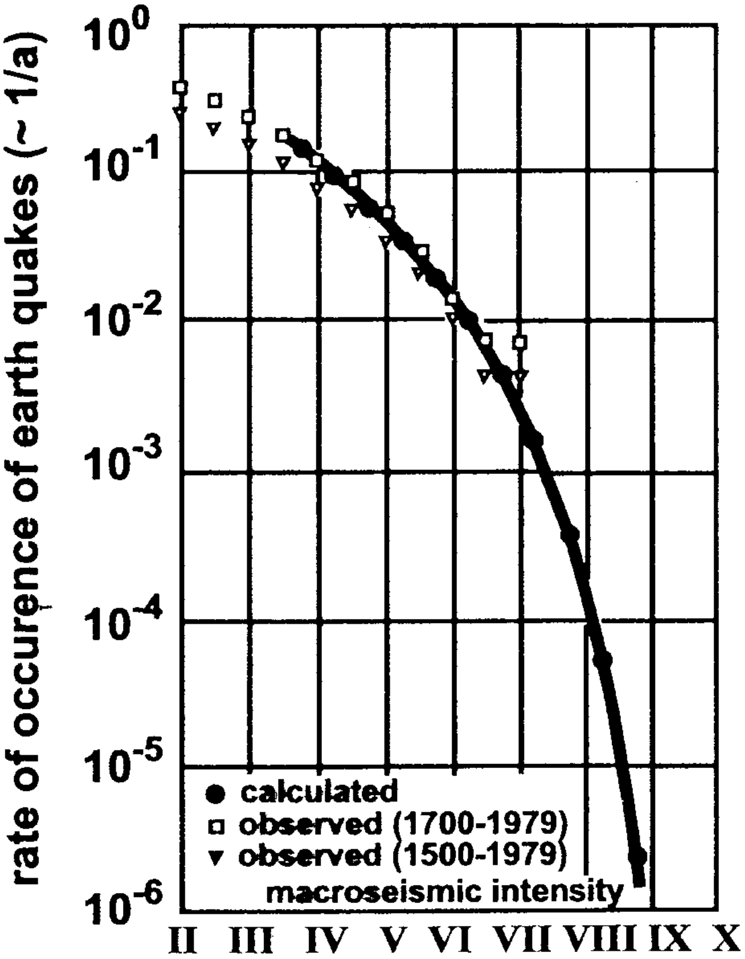




consequences:

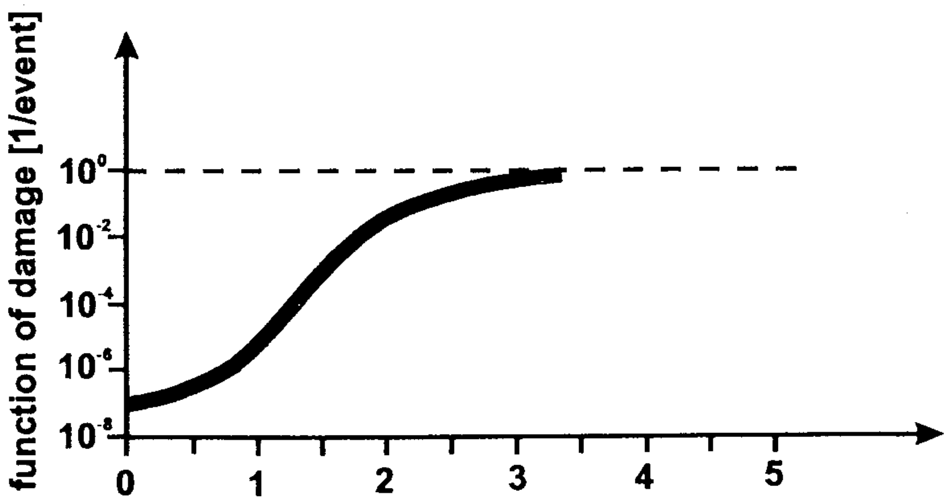
- ΔK_{eff} tolerable
- corrosion limited
- finally welding lips reduce pressure in RPV

Behaviour of nuclear power plants in case of extreme earth quakes



site
KFA Jülich

Intensity of earth quake (MSK)
(IX corresp. ~ 0,3 g acceleration)

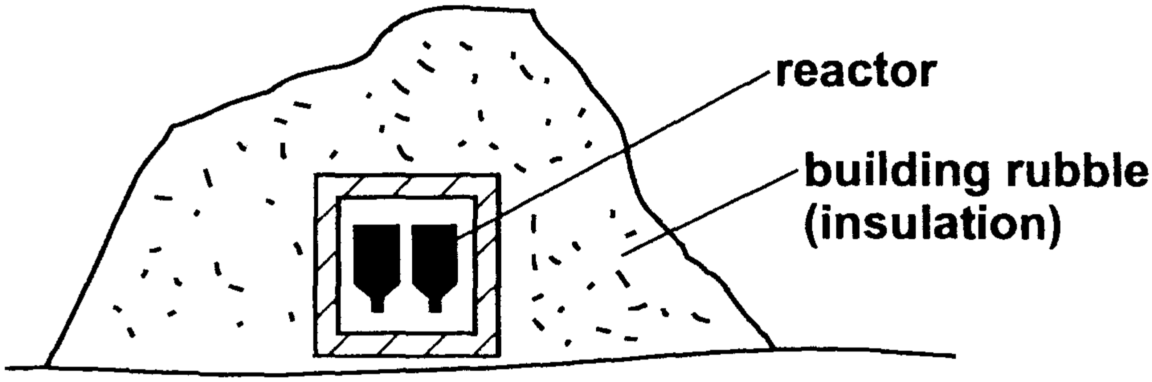


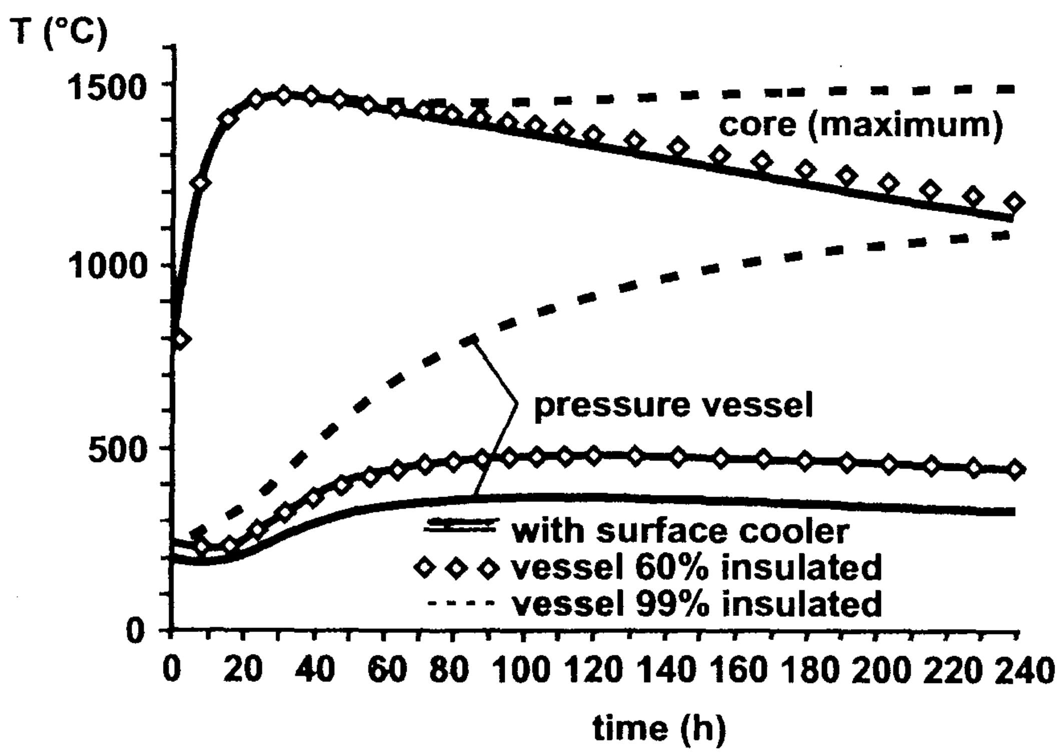
acceleration [m/sec²] ⟶
(0,3 g ≙ 3 m/s²)

Behaviour of reactor in very extreme accidents

assumptions:

- extreme earthquake destroys the reactor building
- reactor is covered completely with building rubble



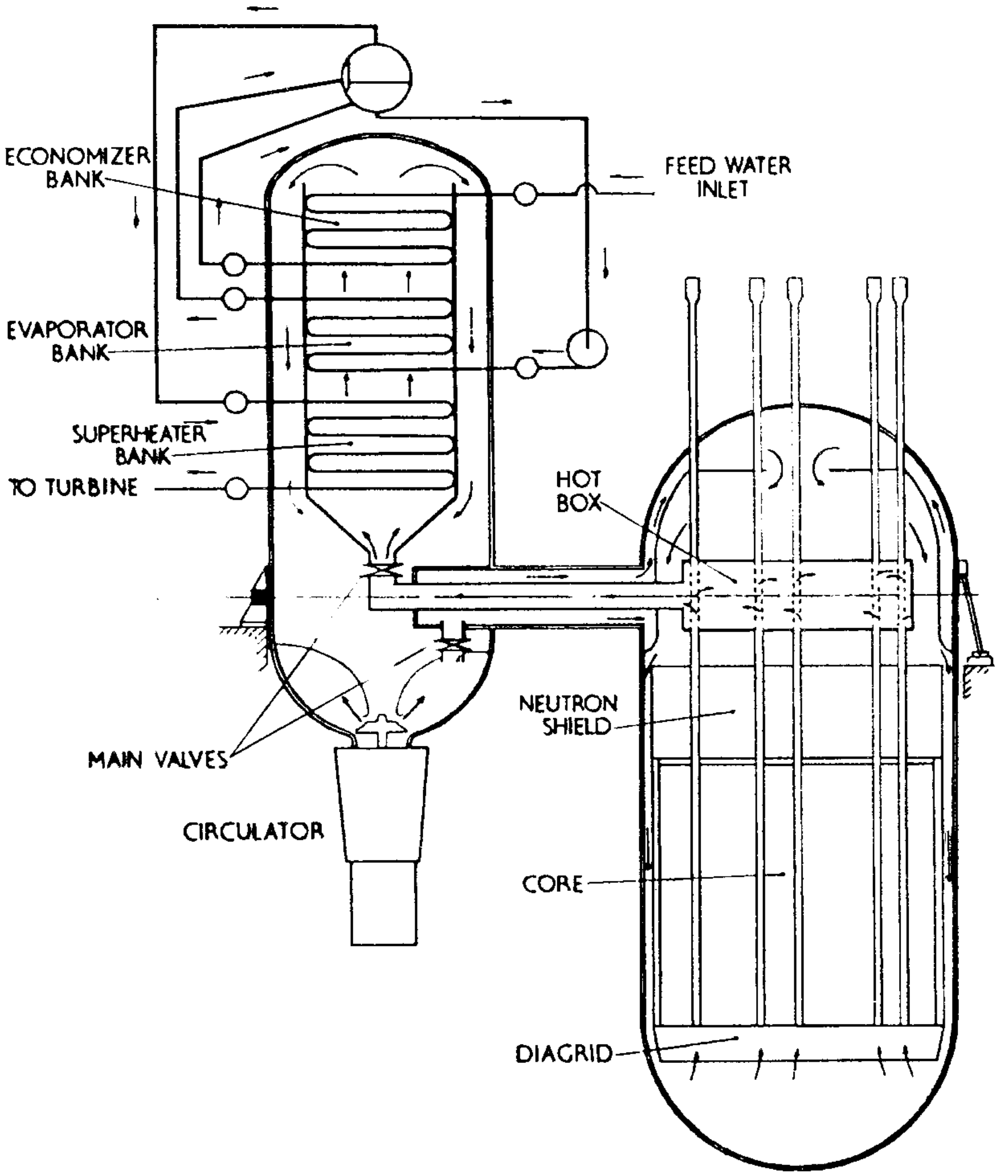


consequences:

- maximal fuel temperatures stay below 1600 °C
- total fission product release from fuel elements < 10^{-5}

Technical realisation of modular HTR-concepts

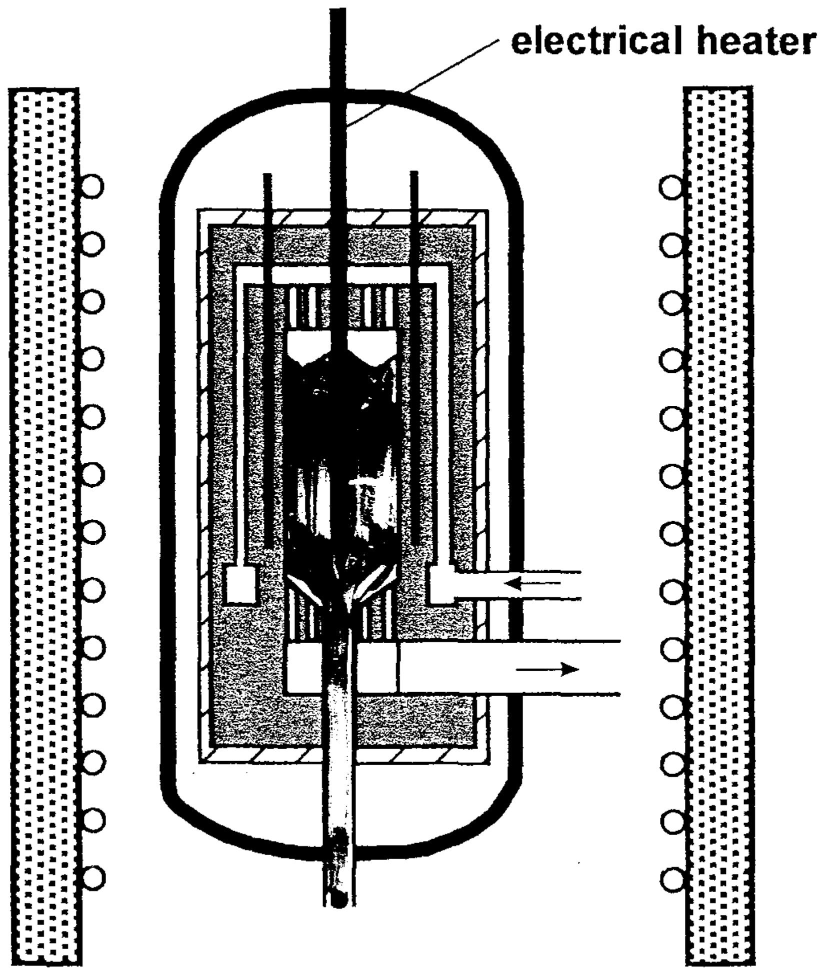
example: WAGR (Windscale advanced gas cooled reactor) in Great Britain



- **primary circuit characteristic for modular HTR (gas flow, coaxial duct, arrangement of steam generator and circulator, primary enclosure)**
- **concept of arangement has been tested successful**

Concept of a "catastrophe-free" nuclear reactor

Experiment to prove the inherent safety in large scale (1/1 or 1/2)



- demonstration of self-acting decay heat removal ($p = 1$ bar, $p = 60$ bar)
- air ingress
- movement of control rods in extreme accidents
- behaviour of reactor pressure vessel in case of overpressure;

Concept of a "catastrophe free" nuclear reactor

Conclusions regarding safety:

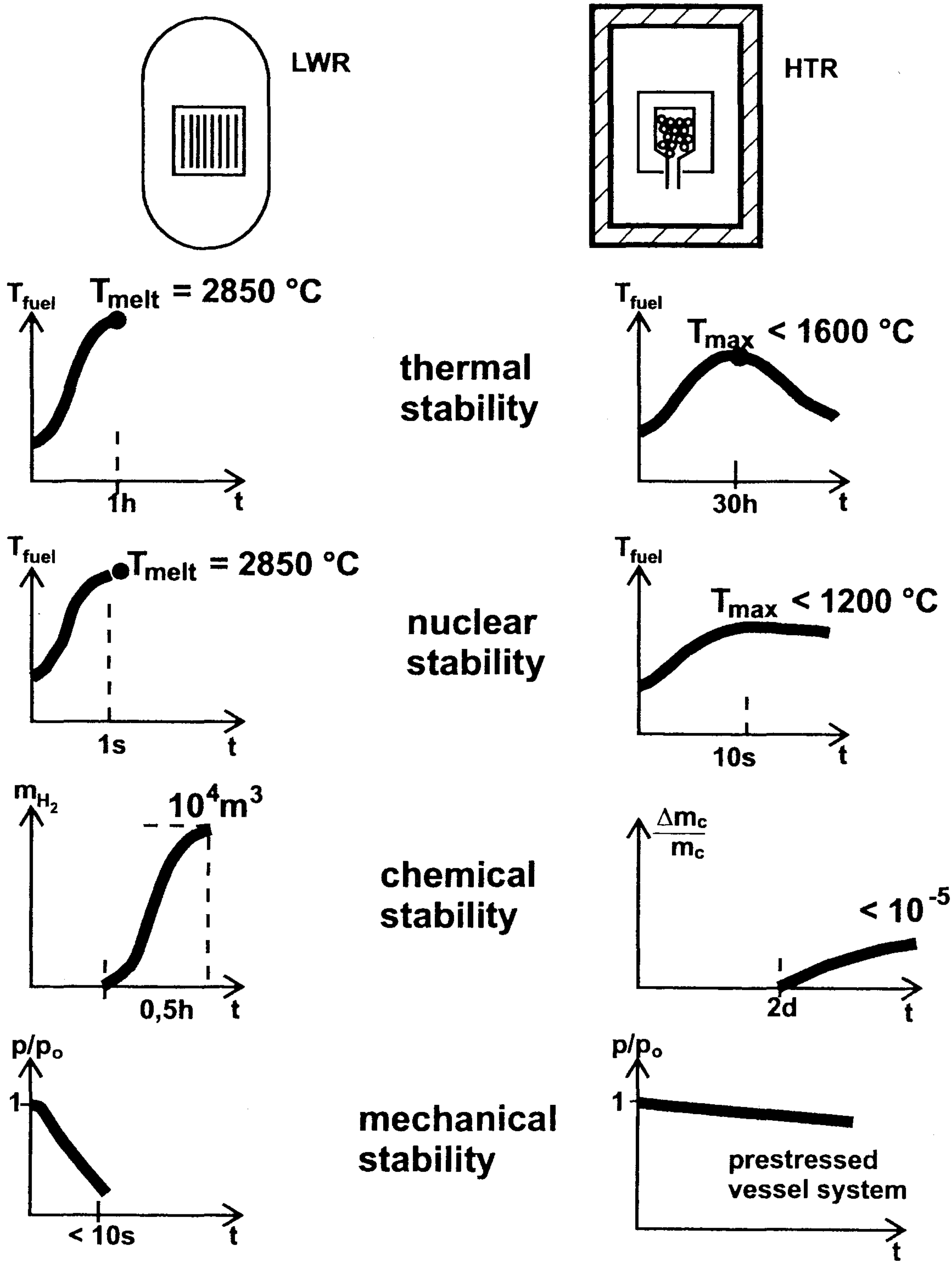
- the fuel never will melt; no super-heating above 1600 °C possible
- the self-acting decay heat removal cannot fail
- the reactor would withstand even large reactivity transients; temperatures stay below 1600 °C
- the reactor vessel cannot burst: no large air ingress possible; no deformation of core or change of composition possible
- large water ingress into the primary circuit causes no problems
- the reactor building withstands standard outer impacts
- even against extreme impacts from outside there are large safety margins

Consequences:

No non-allowable large fission product release is possible in case of accidents; requirements of "catastrophe free" nuclear technology are fulfilled

Safety Aspects of LWR and HTR

Progress in Safety



Concept of a „catastrophe-free“ nuclear reactor

Conclusions

- a nuclear reactor can be designed without non-allowed high fission product release even in extreme accidents

- technologies and the components for this reactor are well known and avaible

- the safety behaviour of this reactor can be tested in large scale

- the concept of „catastrophe-free“ nuclear technology can be realized for waste disposal too

Large Test Facilities in HTR Development

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Forschungszentrum Jülich
Institute für Safety Research and Reactor Technology (ISR)
Research Centre Jülich

28. July 2001

Important questions and work of the development program for HTR

- fuel elements for high helium temperatures, high burn-up and good fission product retention
- high temperature alloys
- main components of plants (compressors, turbines, recuperators, hot gas ducts)
- specific components in helium (bearing, penetrations, sealings, insulations)

Large test facilities

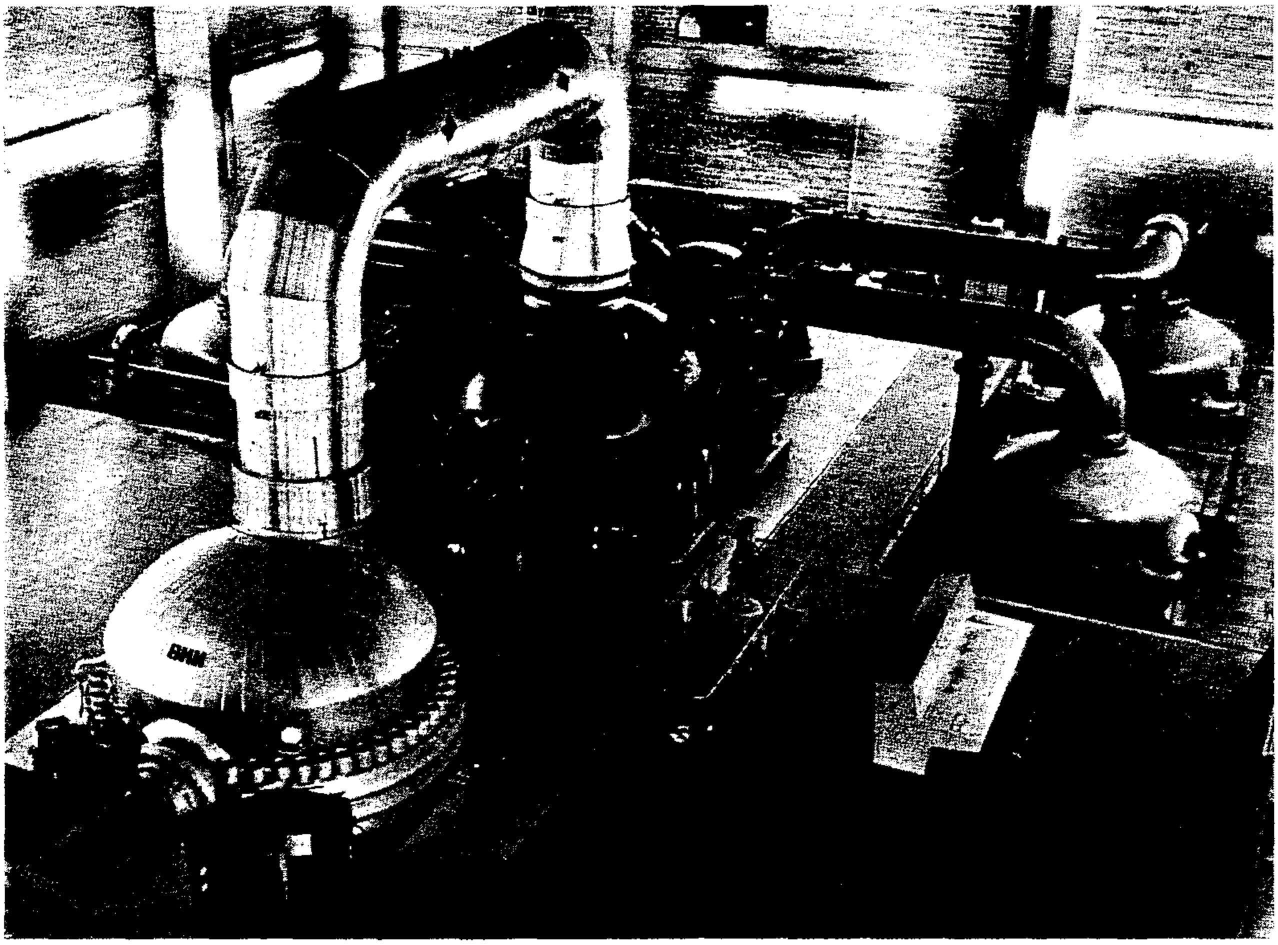
EVO - helium turbine power plant (50 MWel, 750 °C)

HHV - helium turbine test loop (corr. 300 MWel, 850 °C)

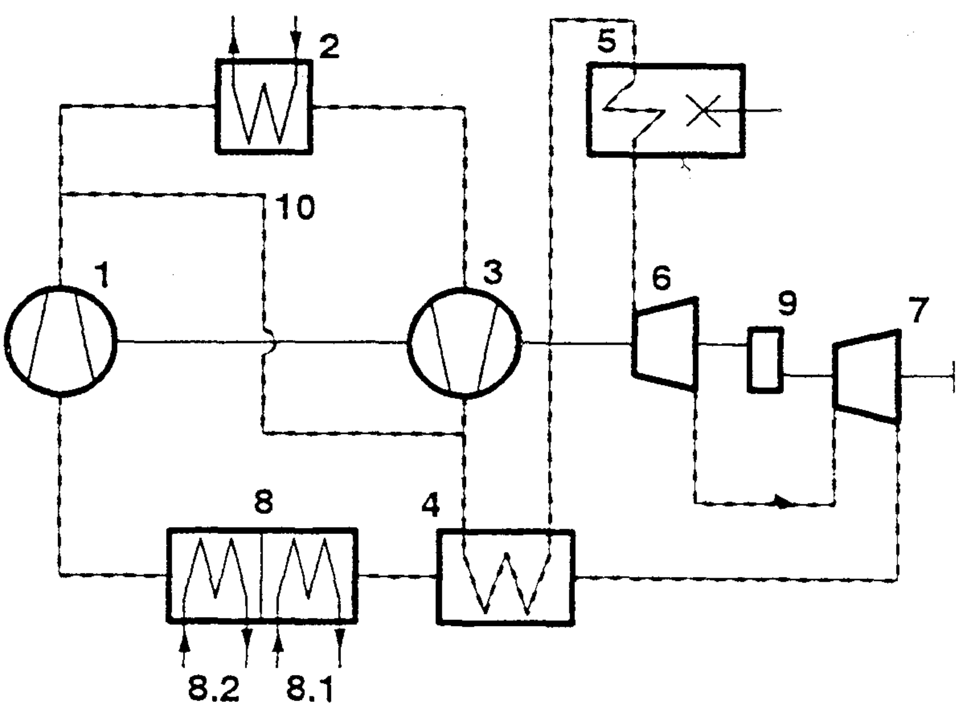
EVA-II - helium heated steam reformer (10 MW, 950 °C)

KVK - helium test loop for intermediate heat exchanger (10 MW, 950 °C)

EVO



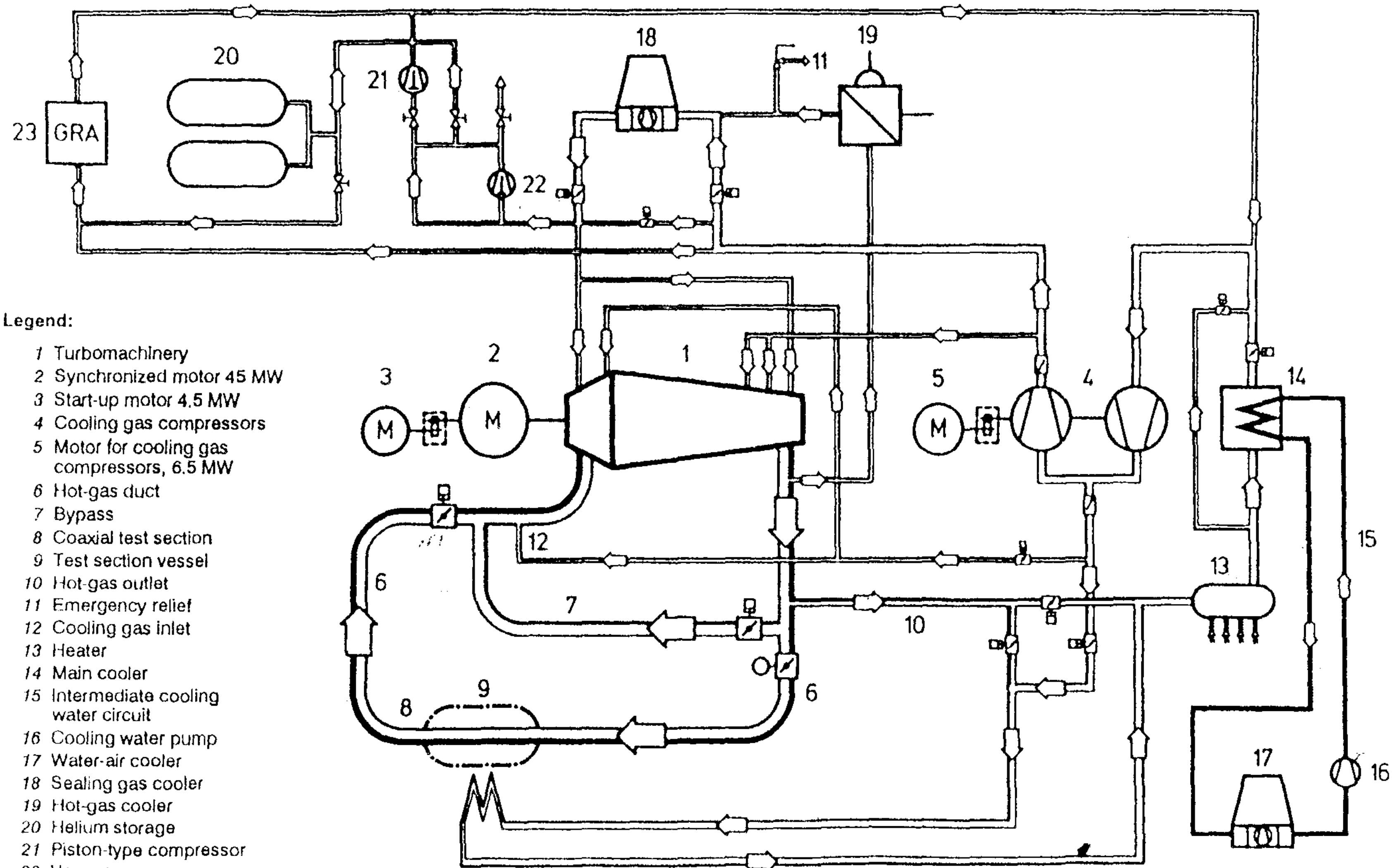
$P_{el} = 50\ MW$, $T_{He} = 750\,°C$



1 Low pressure compressor
2 Intercooler
3 High pressure compressor
4 Recouperator
5 Heater
6 High pressure turbine

7 Low pressure turbine
8 Precooler
8.1 District heat removal section
8.2 Precooler section
9 Gear
10 Regulation bypass

HHV

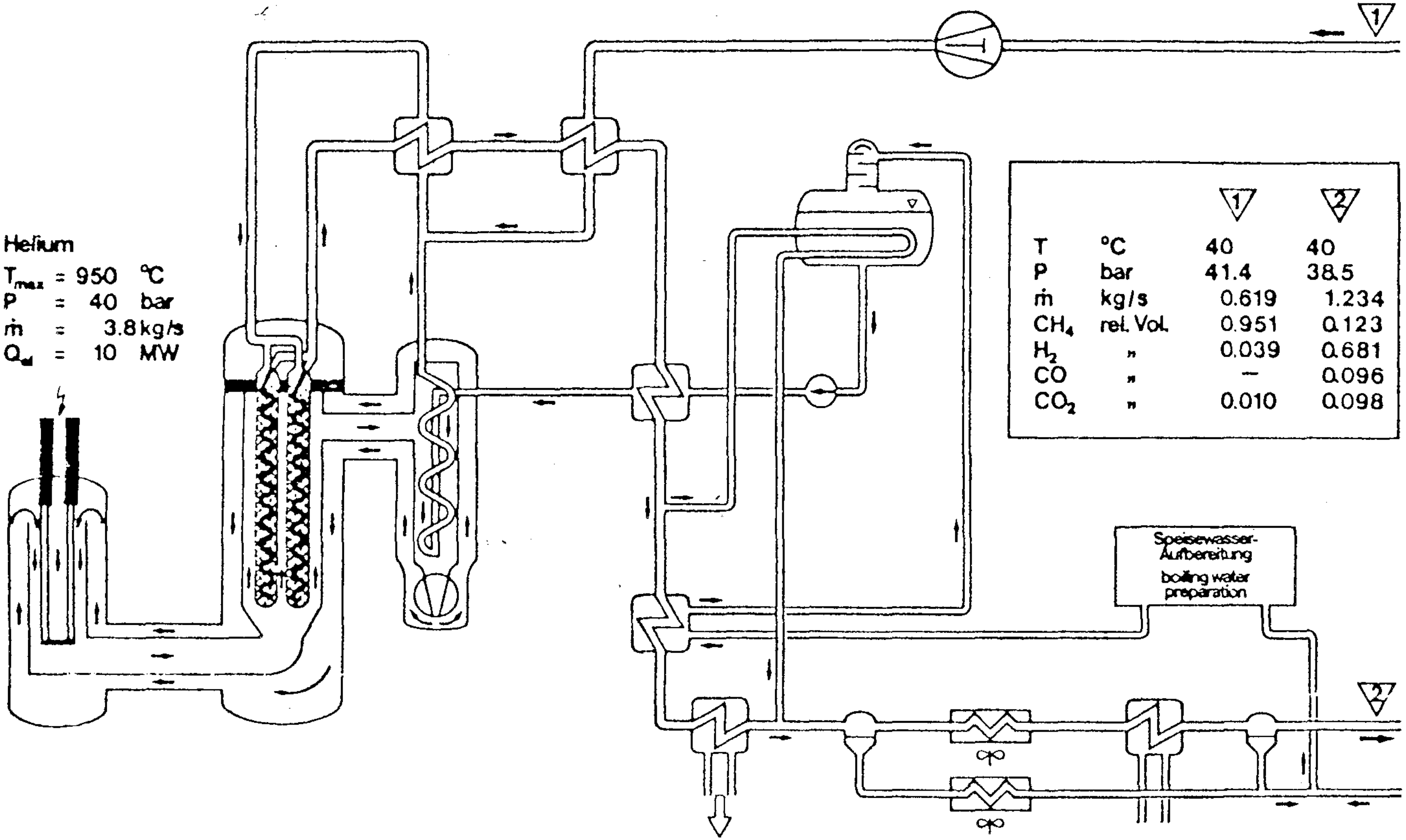


Legend:

1 Turbomachinery
2 Synchronized motor 45 MW
3 Start-up motor 4.5 MW
4 Cooling gas compressors
5 Motor for cooling gas compressors, 6.5 MW
6 Hot-gas duct
7 Bypass
8 Coaxial test section
9 Test section vessel
10 Hot-gas outlet
11 Emergency relief
12 Cooling gas inlet
13 Heater
14 Main cooler
15 Intermediate cooling water circuit
16 Cooling water pump
17 Water-air cooler
18 Sealing gas cooler
19 Hot-gas cooler
20 Helium storage
21 Piston-type compressor
22 Vaccum pump
23 Helium purification system

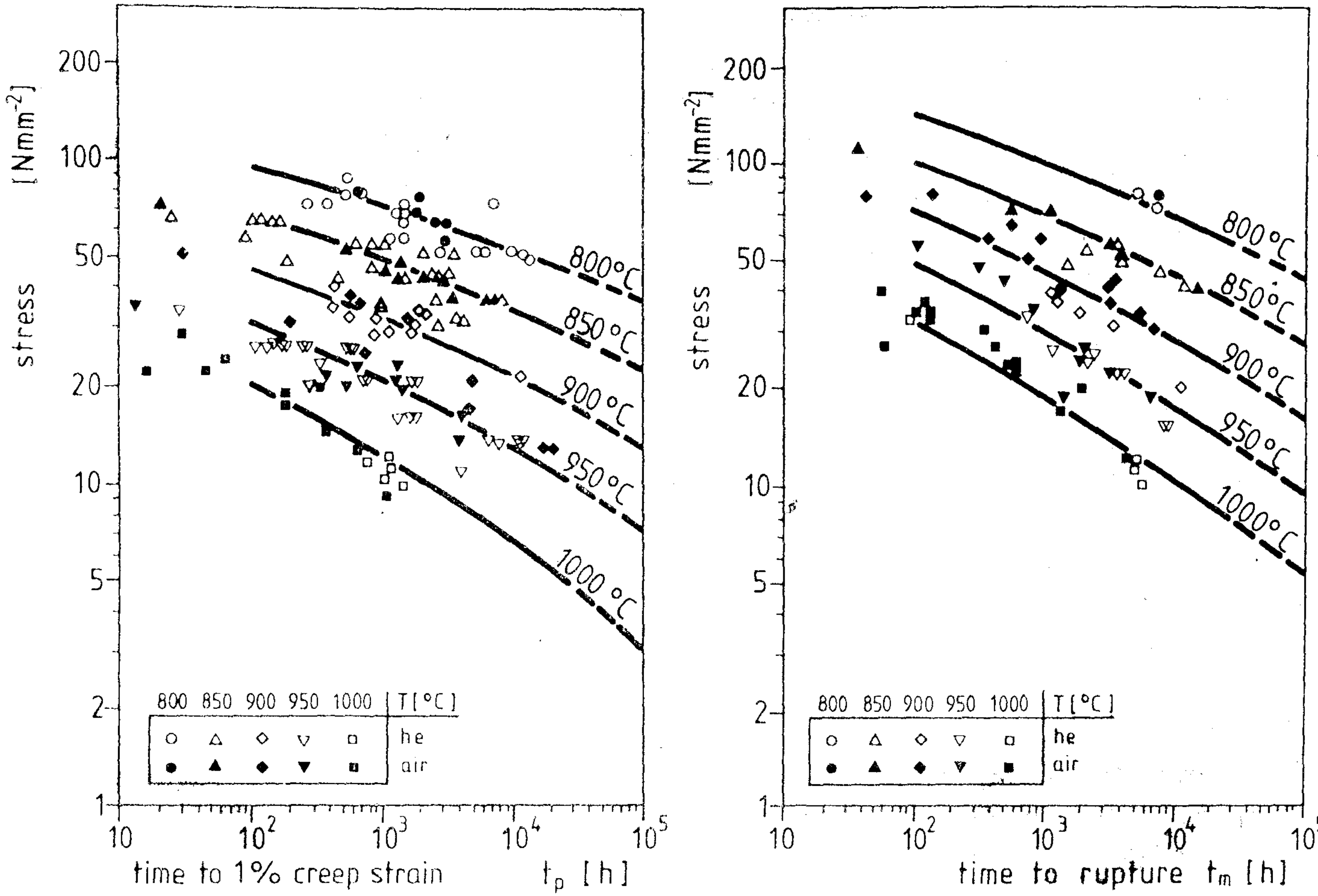
P_{turb} corr. to 300 MWel, $T_{He} = 850°C$

EVA II

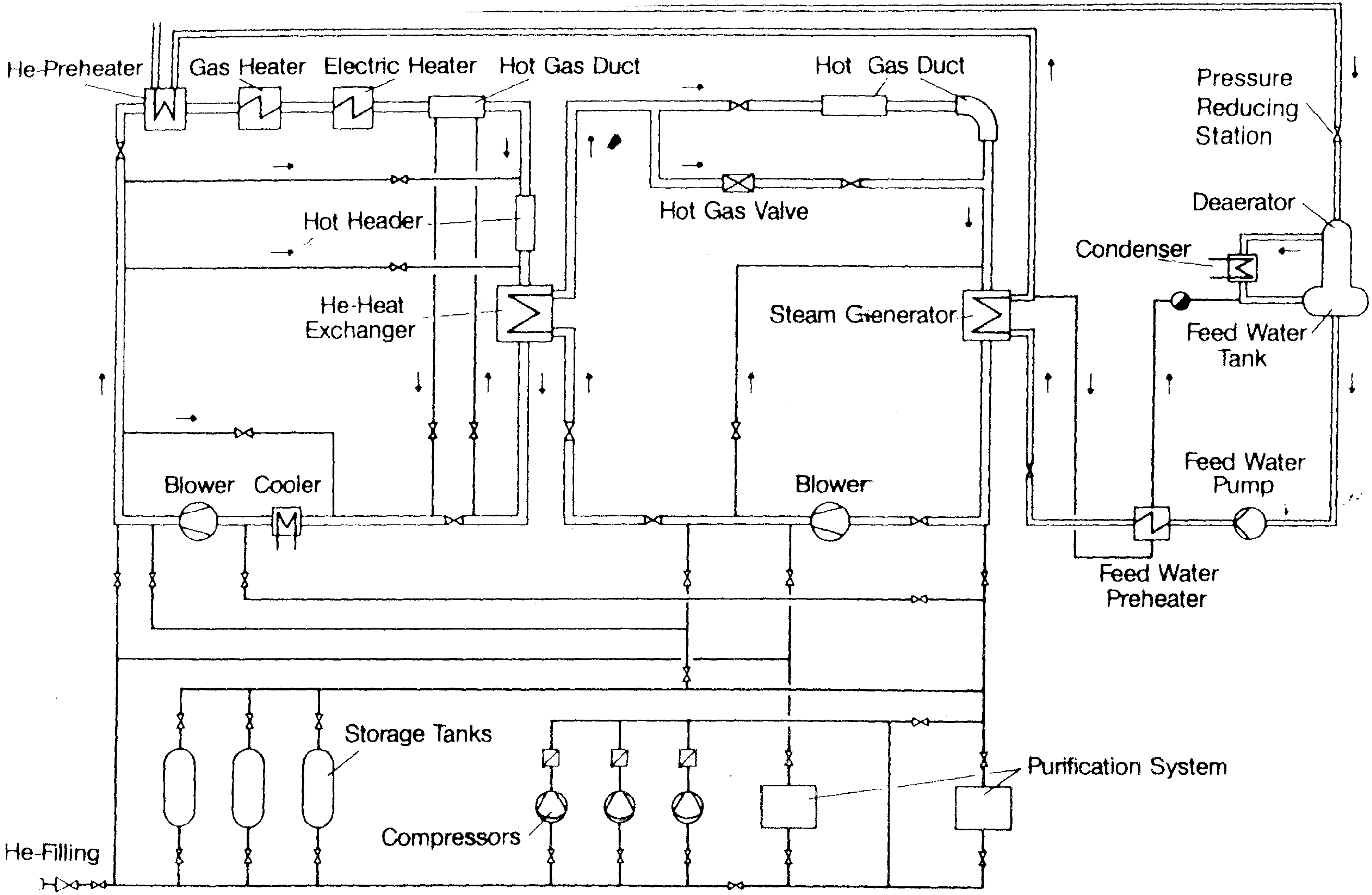


$P = 10\,MW$, $T_{He} = 950\,°C$

Coal Gasification with Heat from HTR materials for the gasifier



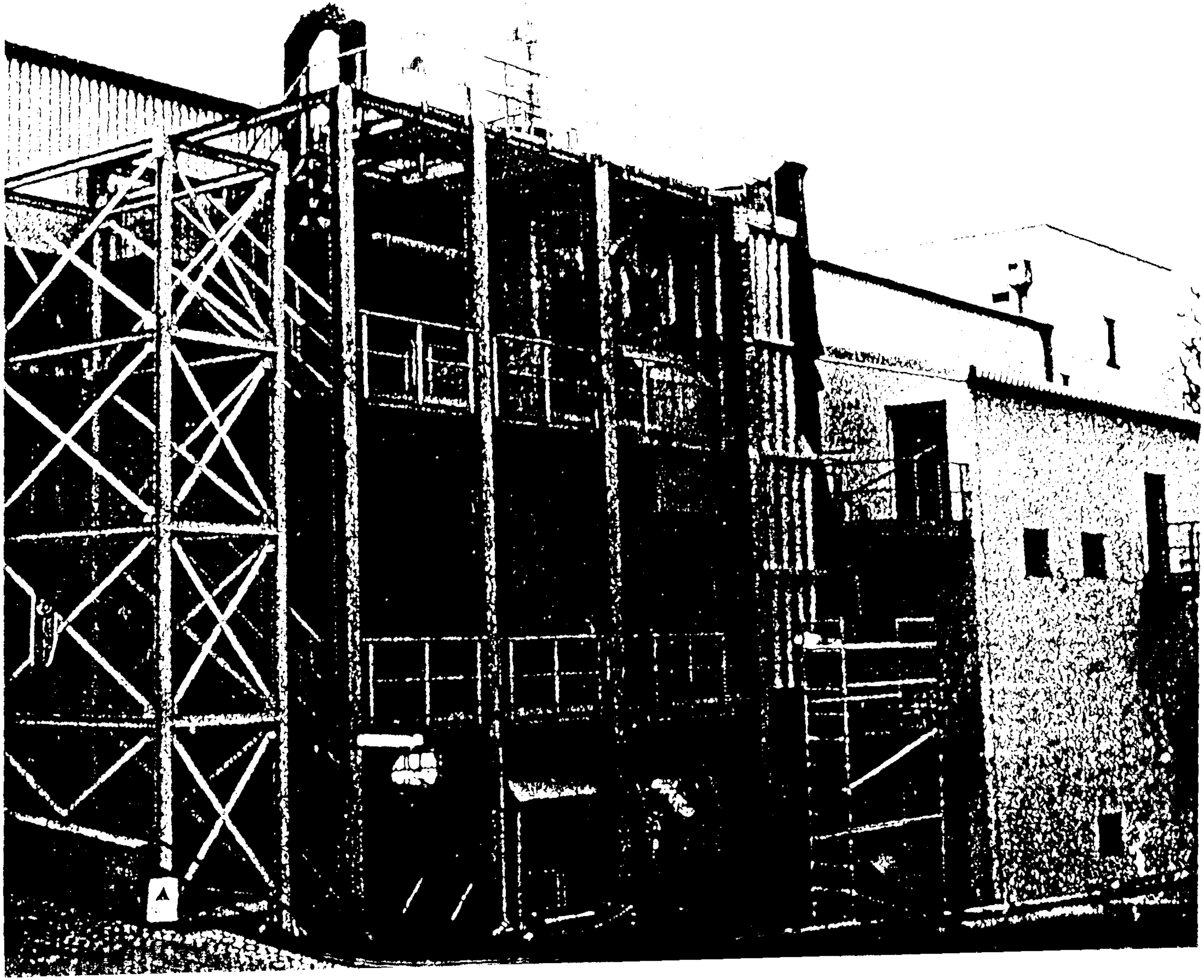
Creep properties of INCONEL 617.

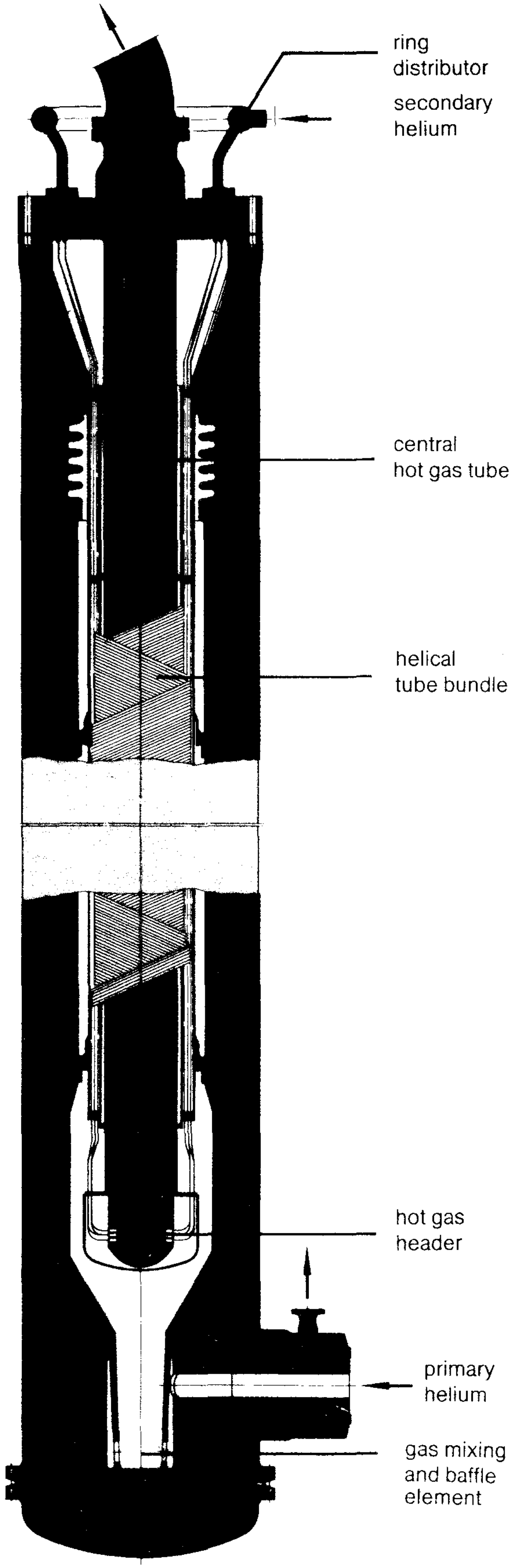


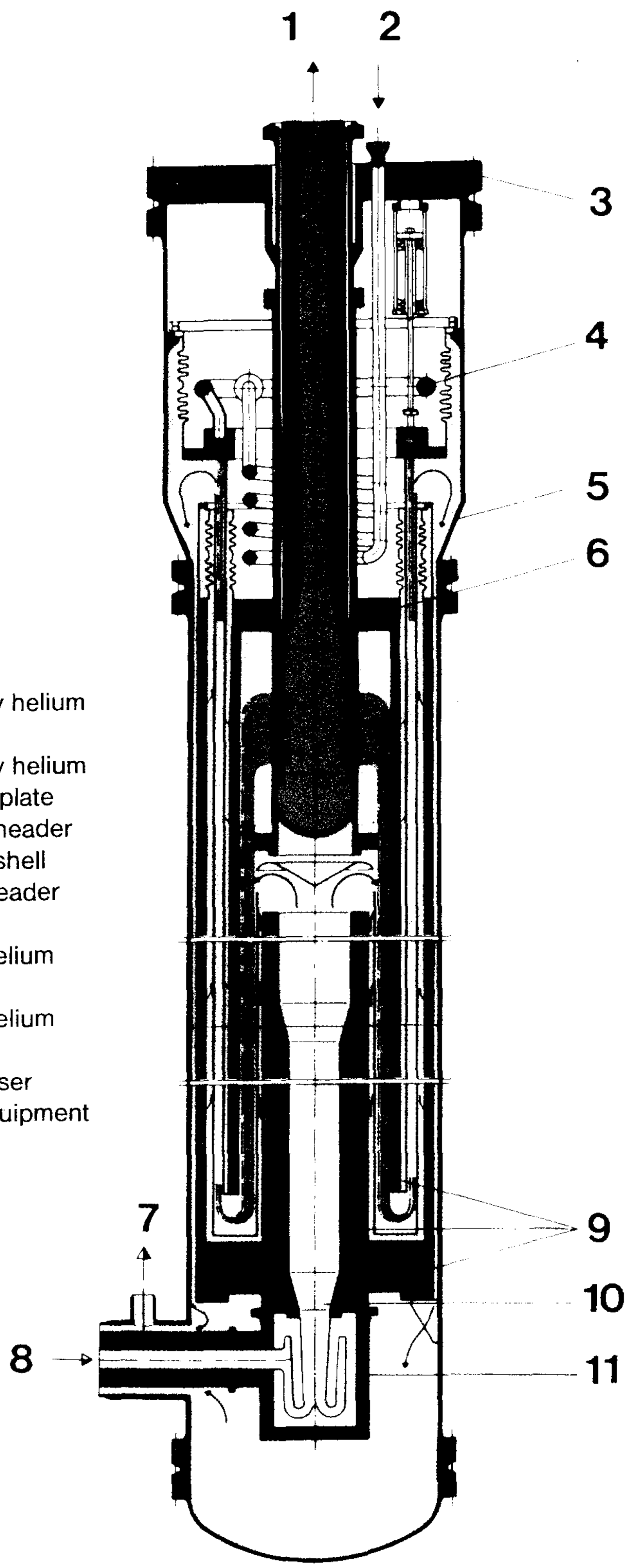
KVK Double Loop Operation With He - Heat Exchanger

Fig. 2

P ~ 10 MW, T_{He} = 950°C




ring
distributor
secondary
helium
central
hot gas tube
helical
tube bundle
hot gas
header
primary
helium
gas mixing
and baffle
element



1. Outlet
   secondary helium
2. Inlet
   secondary helium
3. Mounting plate
4. Cold gas header
5. Pressure shell
6. Hot gas header
7. Outlet
   primary helium
8. Inlet
   primary helium
9. Insulation
10. Hot gas riser
11. Mixing equipment

Conclusions

- more than 3 x 10^9\$ have been spent in Germany in the last 40 years to develop the HTR technology
- the knowledge on HTR is (still) in the head of a lot of people
- in our mind it will be an ad-vantage to use this know how as much as possible for the PBMR

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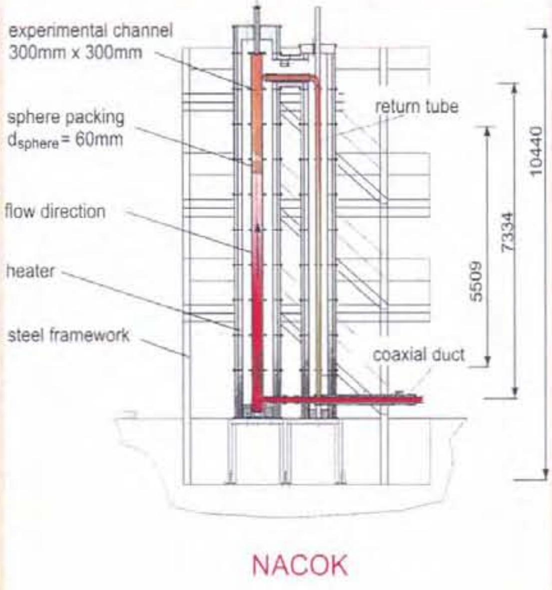
Institut für Sicherheitsforschung und Reaktortechnik

NACOK

Natural Conv. in Core with Corrosion



NACOC – Main data:		
Max. temp. in experimental channel	1200	°C
Max. temp. in return tube	800	°C
Max. through-put of air	17	g/s
Total number of thermo-couples	82	
Total number of gas measurement points	26	
Number of points to measure gas velocity	2	
Max. heating power	147	kW

Grace Time Till Natural Convection Start

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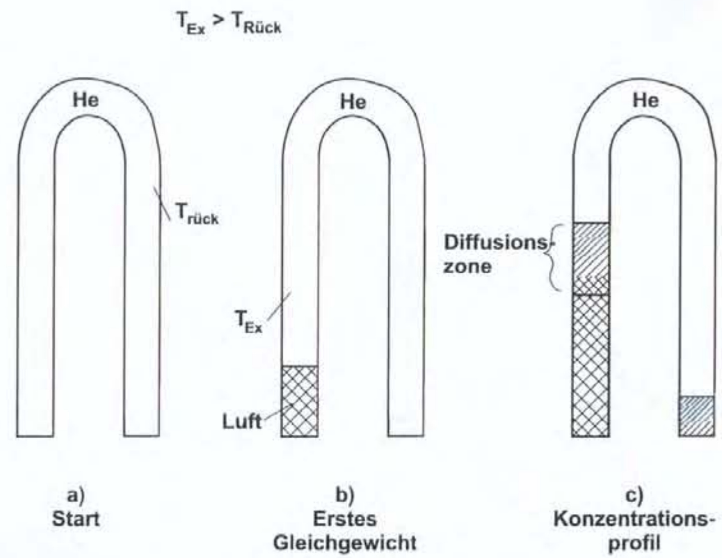
Institut für Sicherheitsforschung und Reaktortechnik



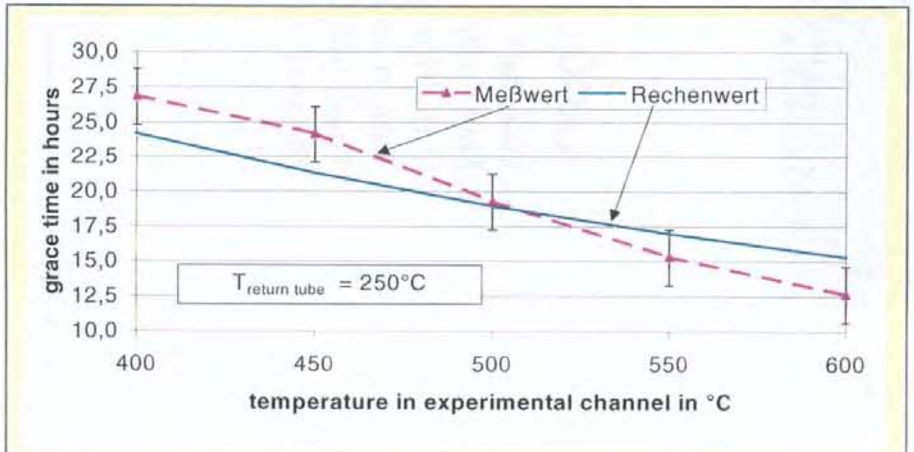
grace time produced by diving bell effekt

$$\boxed{\Delta t = \frac{L^2}{4D}}$$

Δt = *grace time*

L = *length of diffusion (hight)*

D = *coefficient of diffusion*



Δt = 20 h measurement NACOC

Δt = 80 h for HTR-Modul, resp. PBMR

Air Ingress and Corrosion Masses

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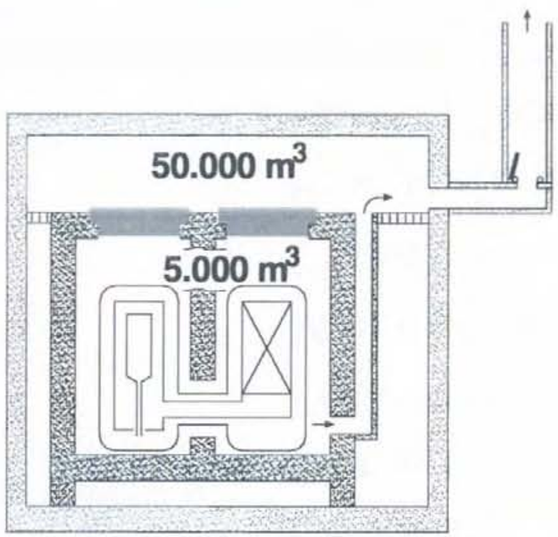
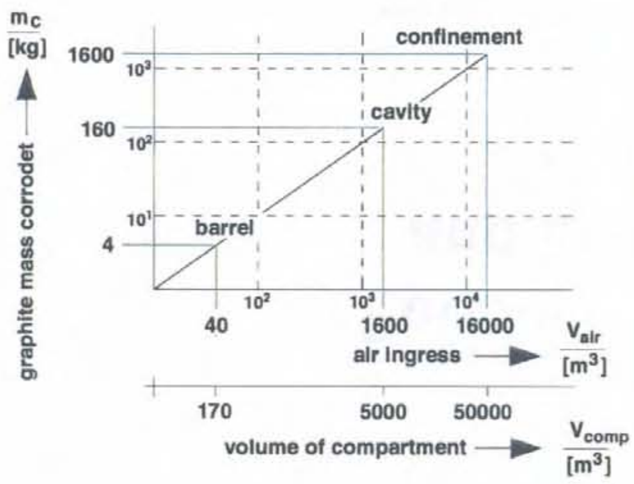
Institut für Sicherheitsforschung und Reaktortechnik





1. **At depressurization accident:**
   Helium from primary circuit (pushes“ air out of confinement.

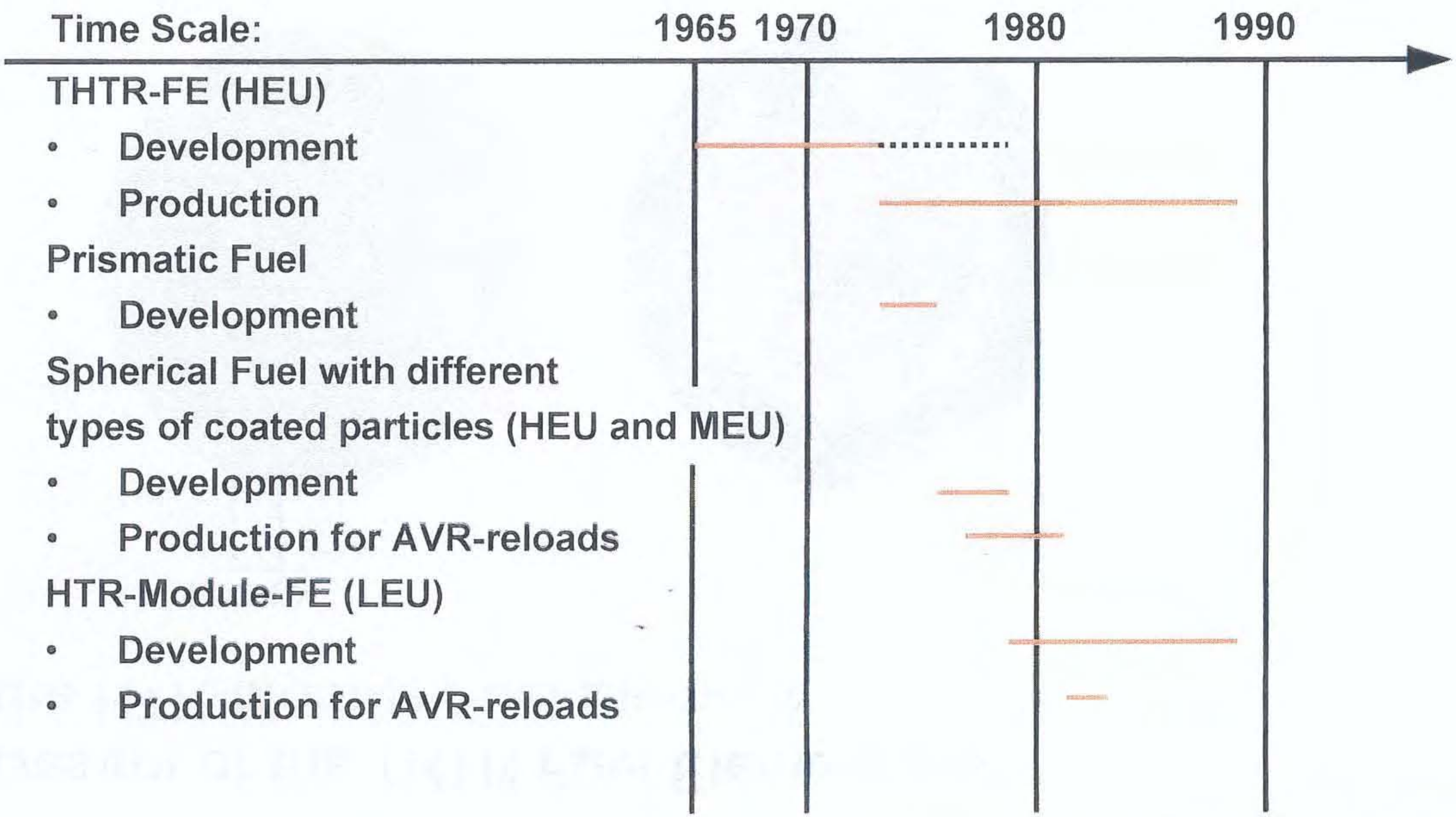
2. **Grace time till natural convection start**
   (due to diving bell effect): 80 h

3. **In Confinement (50 000 m^3) there is max. 16 000 Norm-m^3 air mixed with helium.**
   This air corrodes about 1600 kg C (of in total 500 000 kg).
   If less air, than less corrosion: Accident management measures.

Pebble Bed Fuel Element Research and Development and Industrial Production in Germany

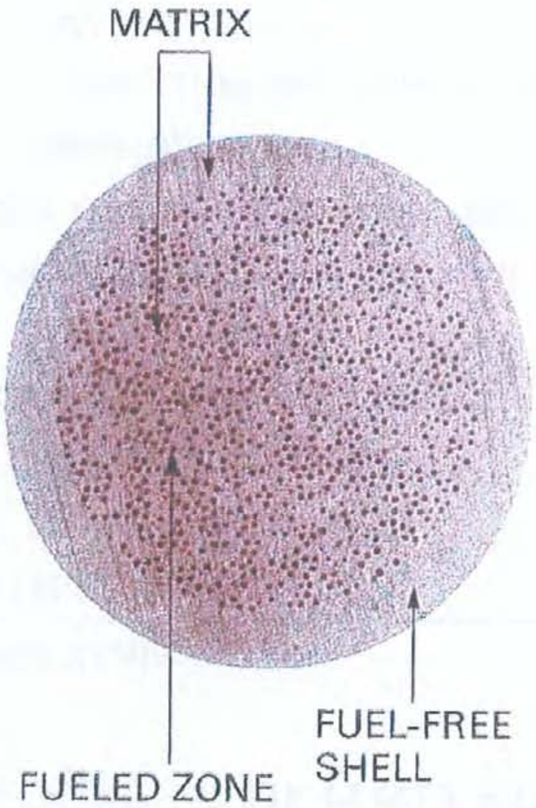
1. Overall View and Progress of R&D and Production
2. Design of the THTR-Fuel Element and the HTR-Module Fuel Element
3. Manufacturing Process
4. Characterization
5. Production Experience
6. Special Quality Assurance System and Philosophy

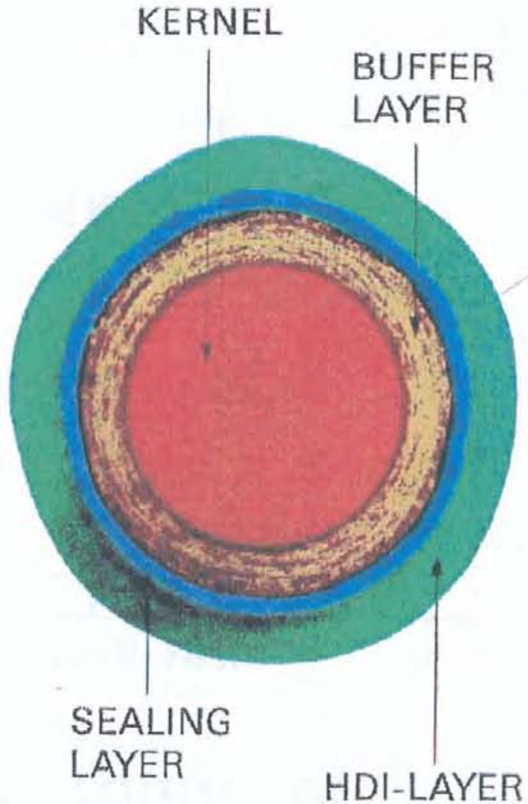
Overall View and Timely Progress of R&D and Production



TESSAG
NUKEM

Design of the THTR Fuel Element and the HTR-Module Fuel Element



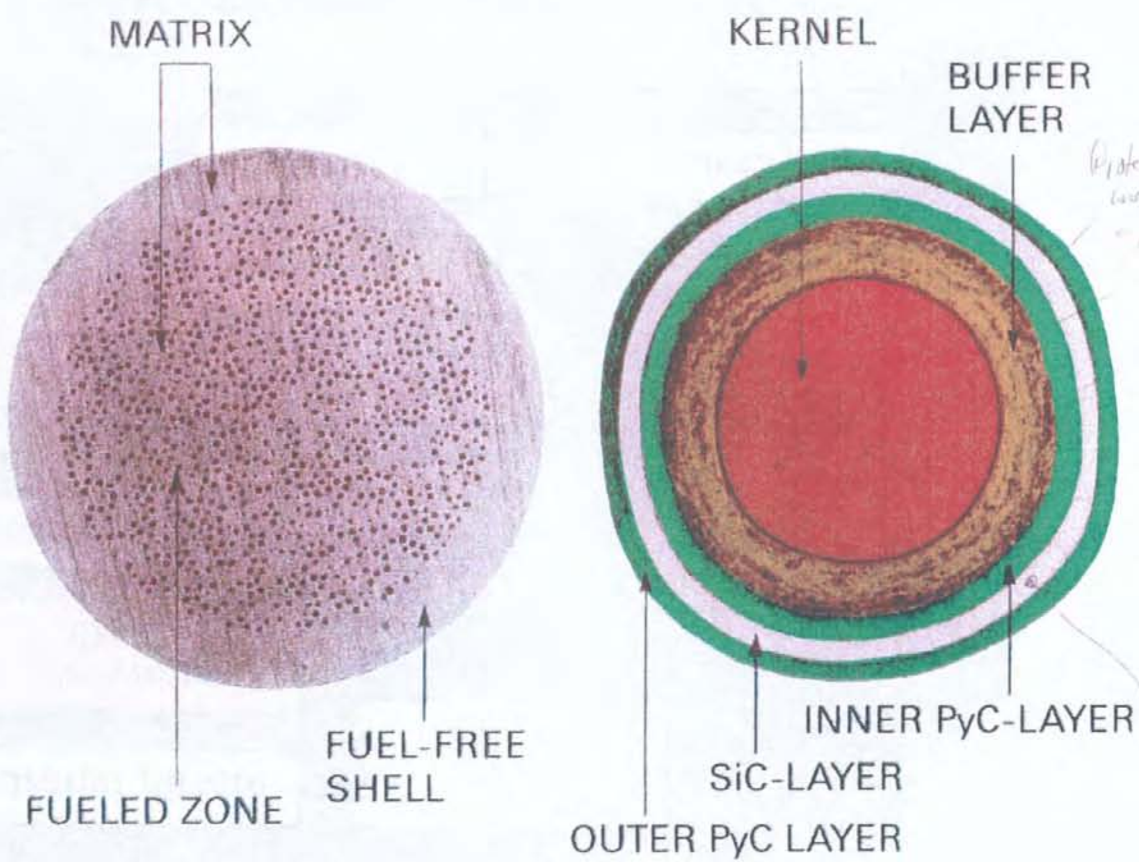


400 micron

99 %

THTR-Fuel Element

Design of the THTR Fuel Element and the HTR-Module Fuel Element

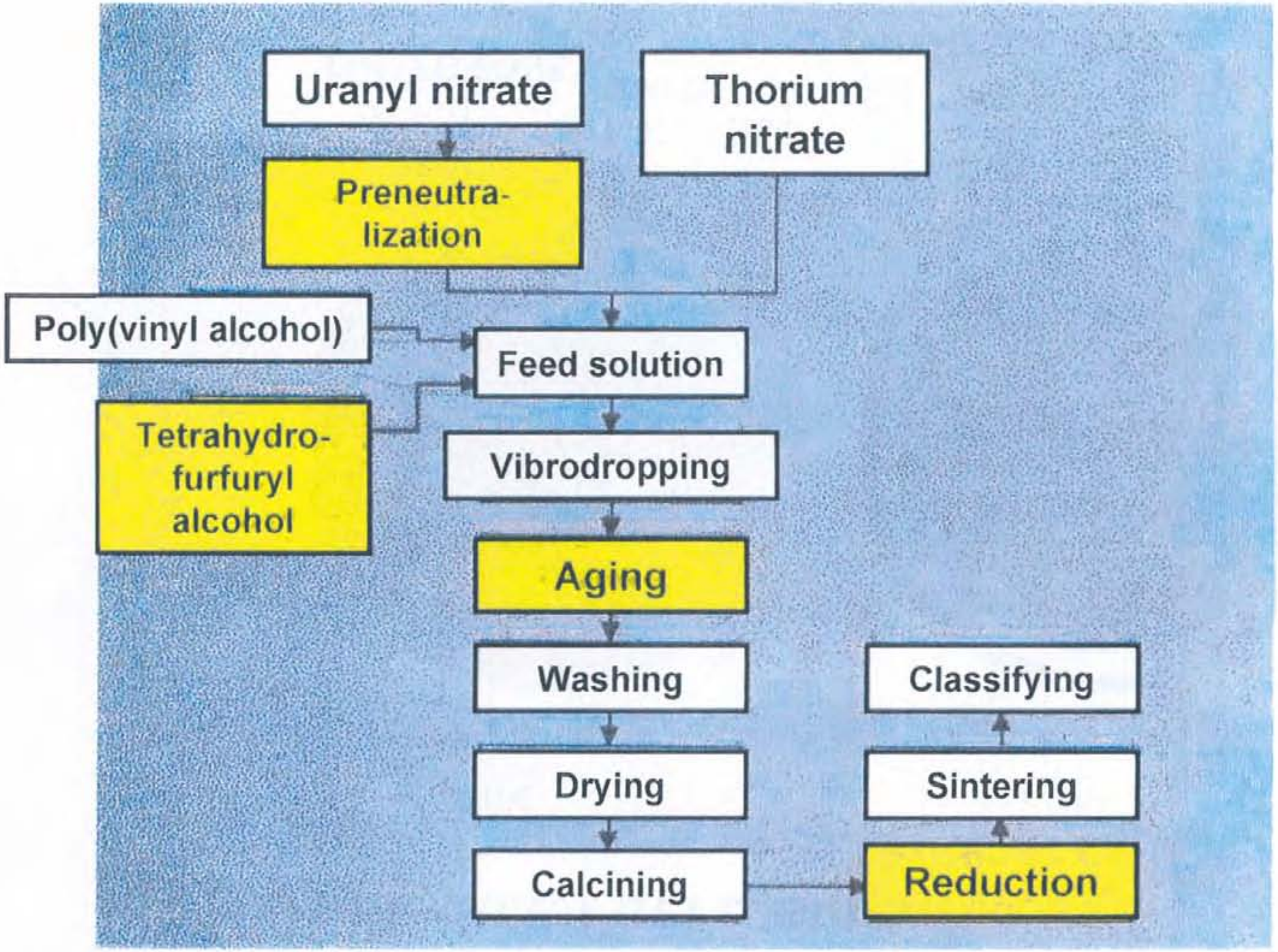


HTR-Fuel Element

Barrier for Fission product

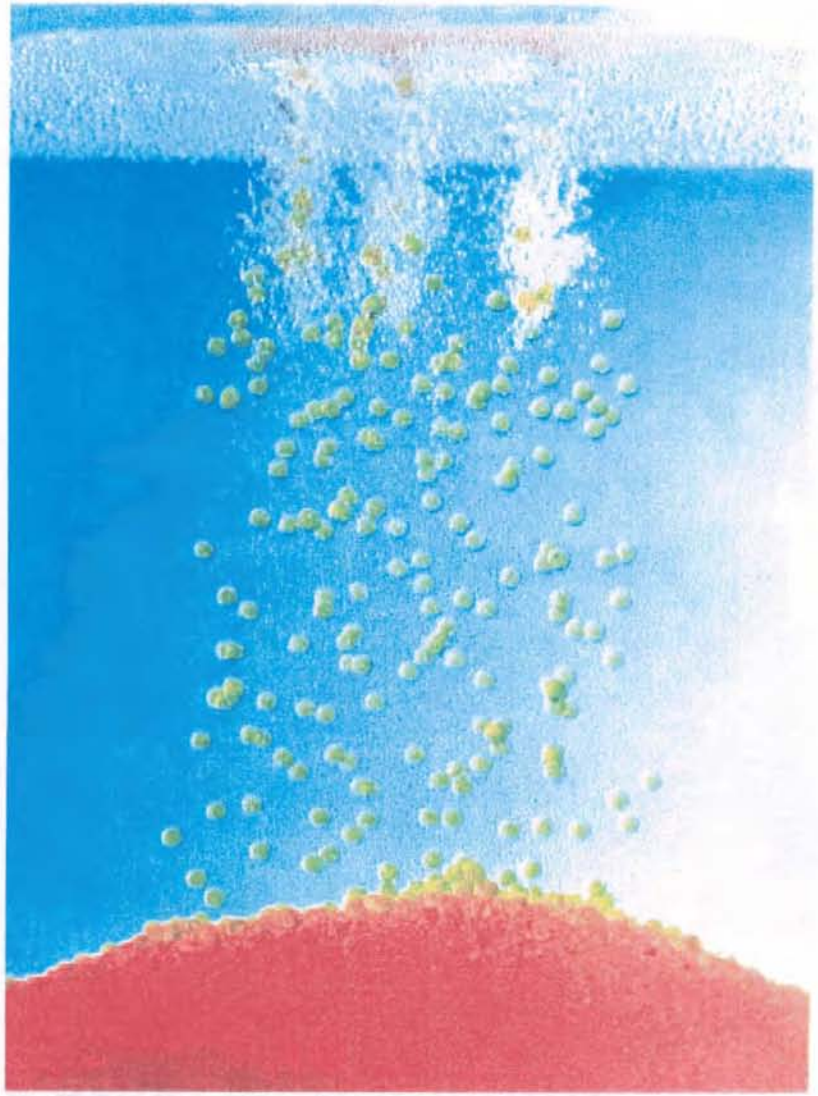
Kernel Manufacture Process Diagram

TESSAG
NUKEM



Fuel Kernel Fabrication

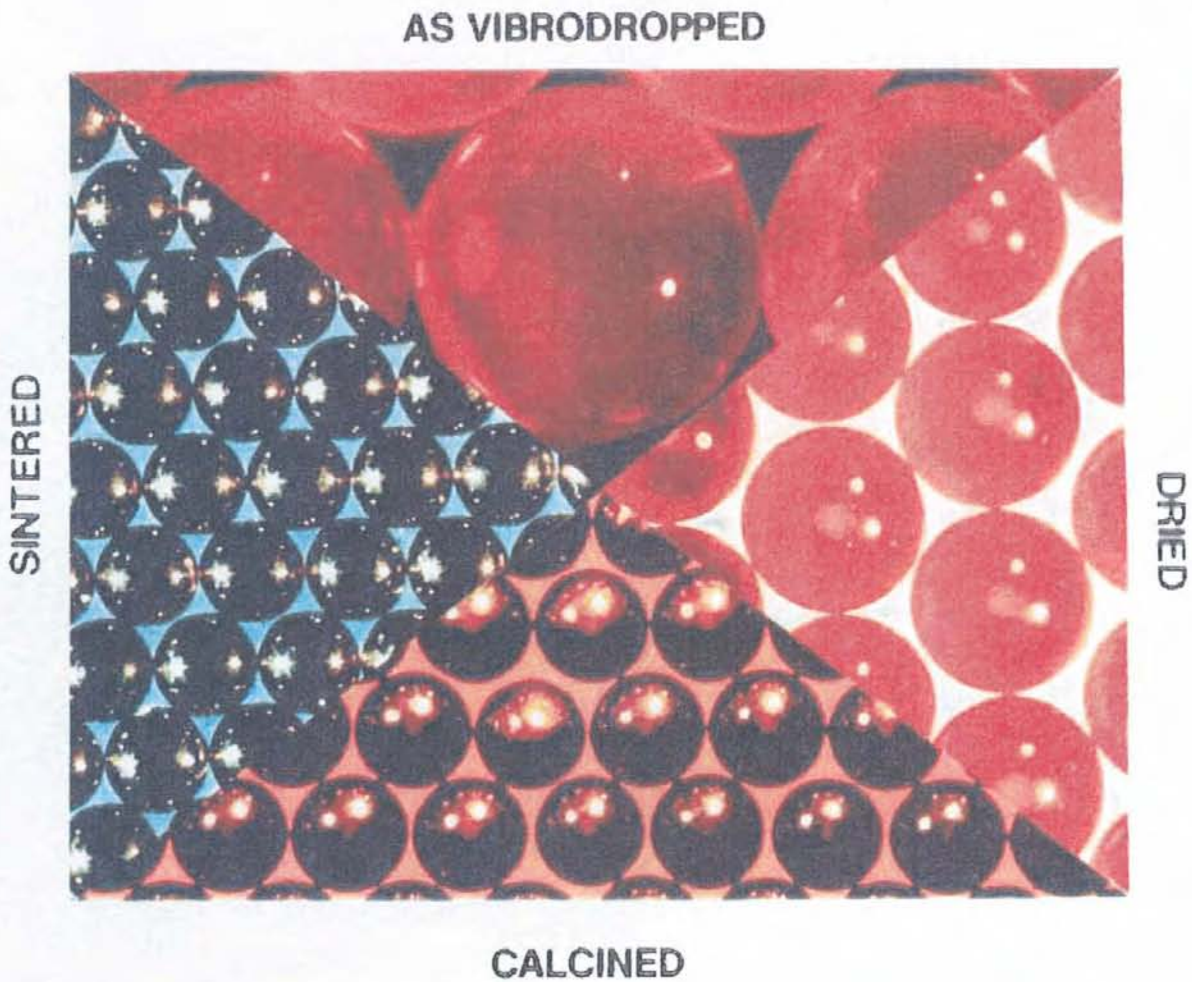
Kernel Manufacture



Kernels in the Precipitating Agent

Kernel Manufacture

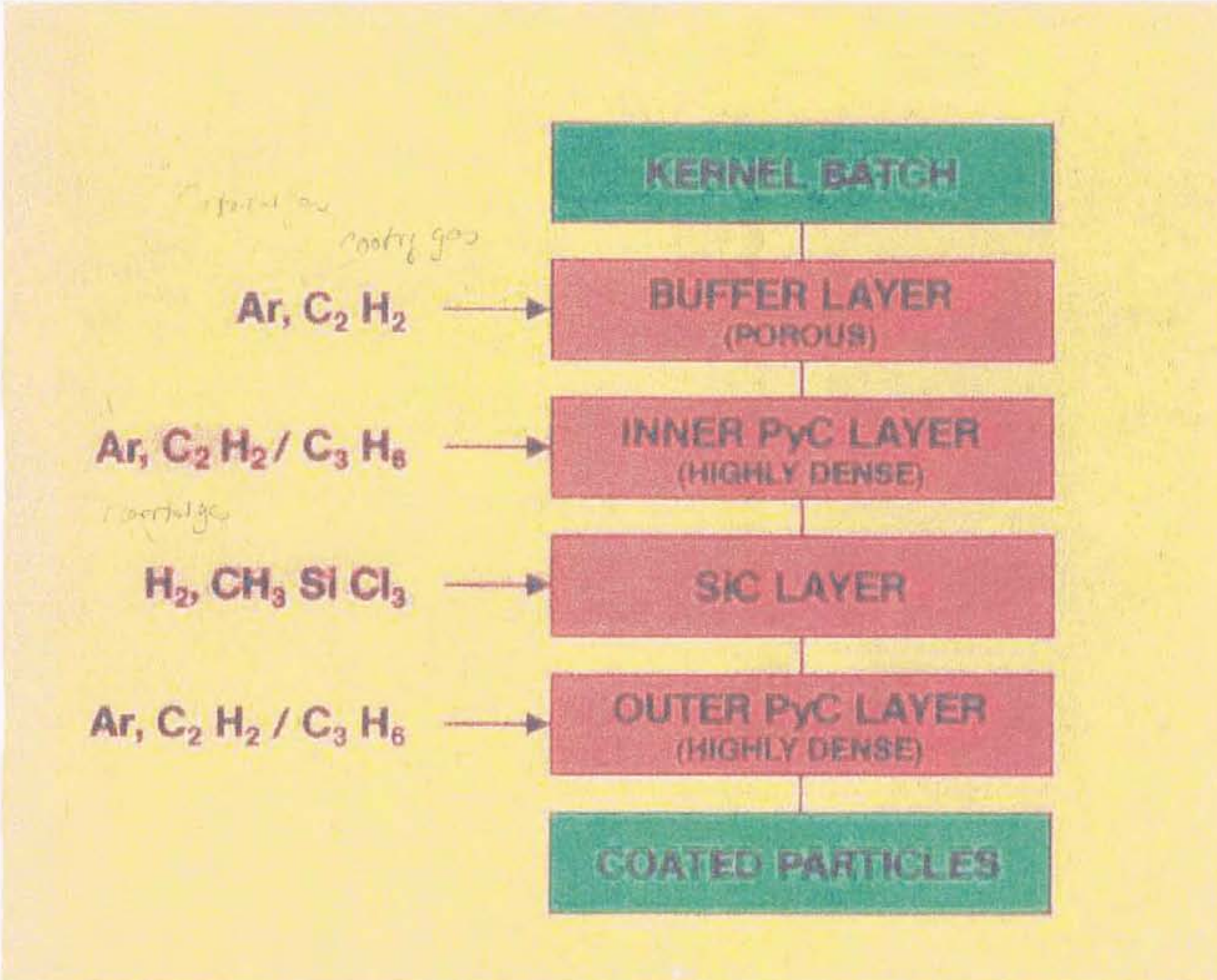
TESSAG
NUKEM



Mixed Oxide Kernels after different Stages of Manufacturing

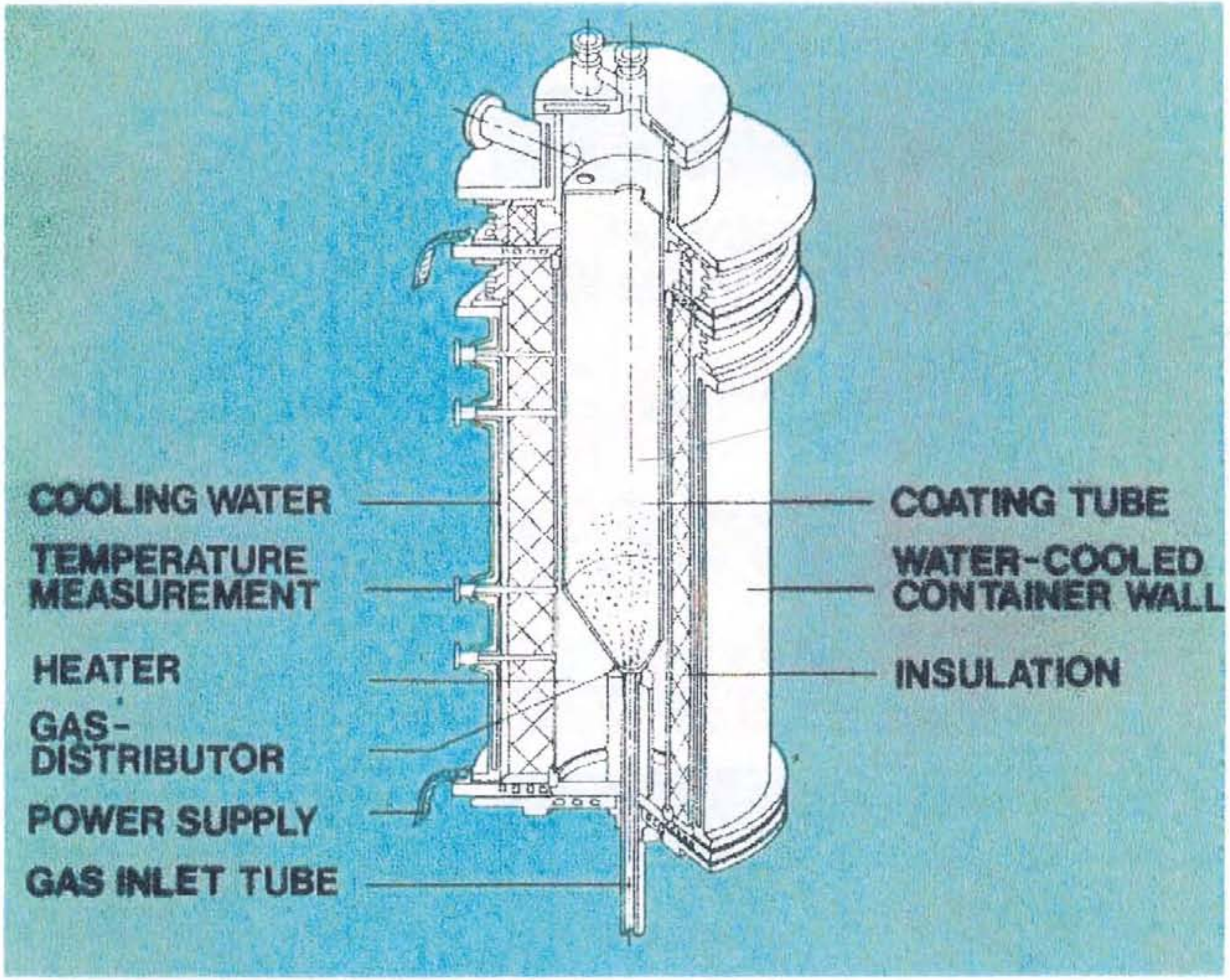
Coating Process Diagram

TESSAG
NUKEM



Coating of HTR Fuel Kernels

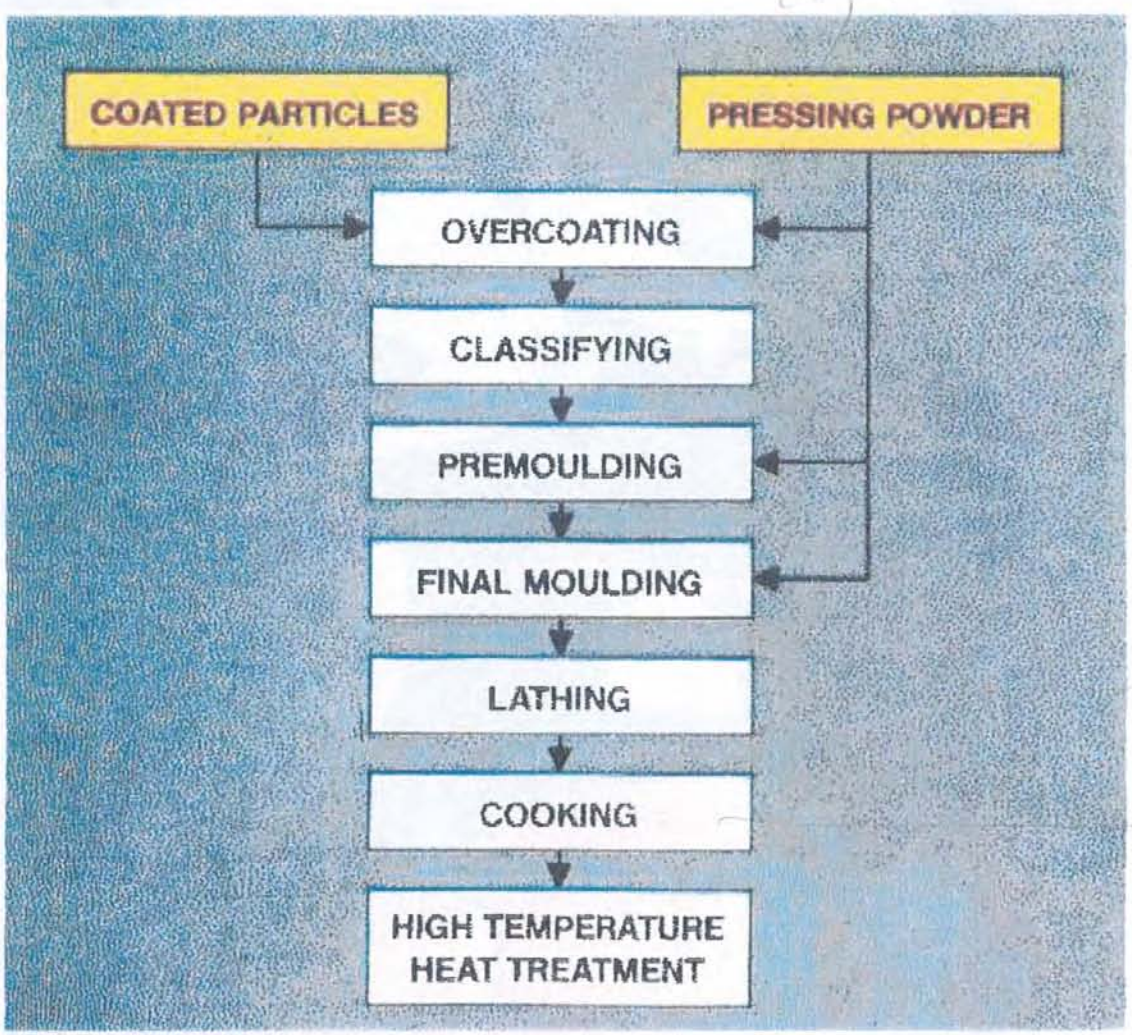
Coating Process



Fluidised Bed Reactor

Fuel Element Manufacturing Process

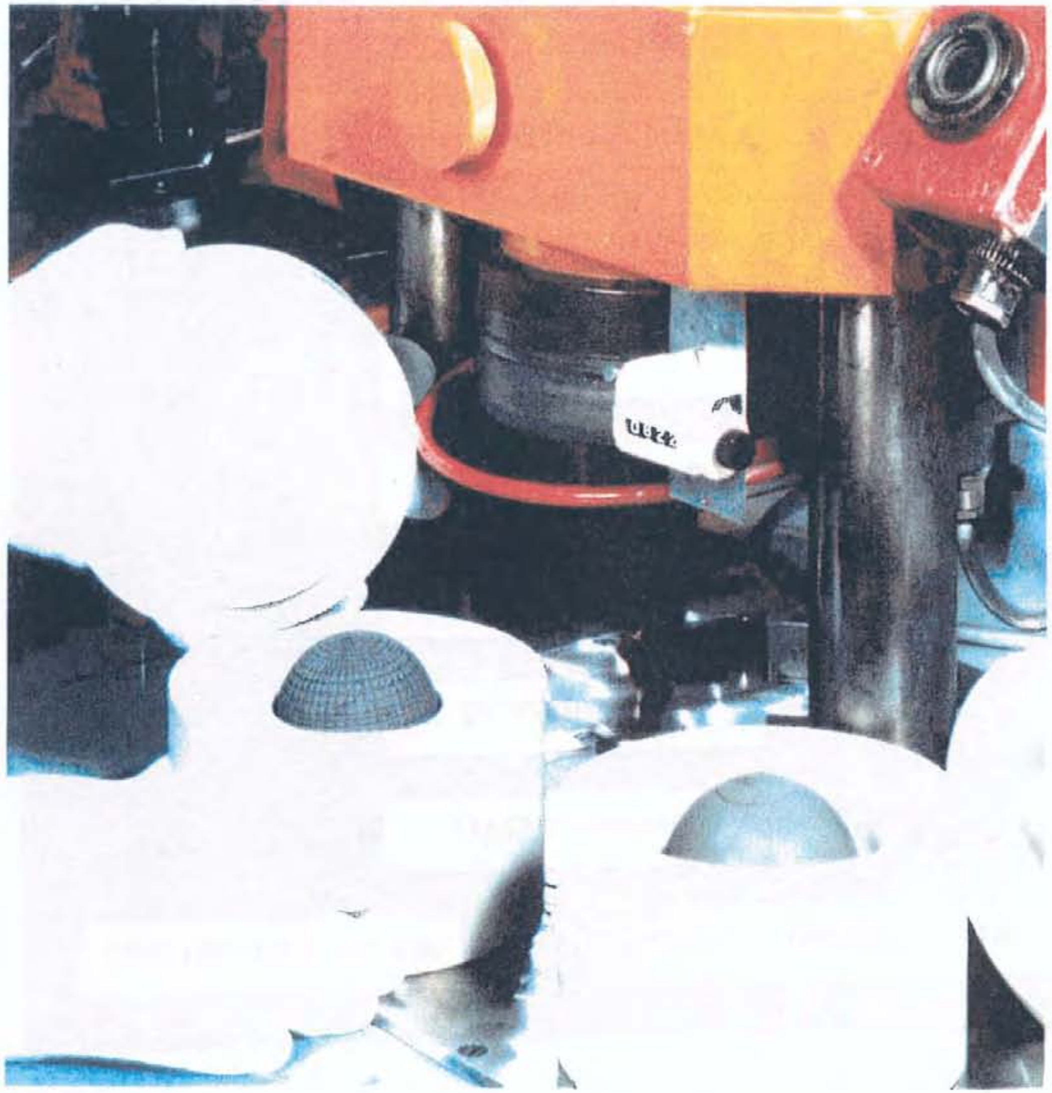
TESSAG
NUKEM



HTR-Fuel Element Fabrication

Fuel Element Manufacturing Process

TESSAG
NUKEM



Rubber Dies for Fuel Spheres

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HTR Fuel Specific Characteristics

Kernels:

- diameter
- roundness

Coated Particles:

- ratio of defect SiC-layers
- diameter
- roundness
- thickness of each layer
- density of each layer
- anisotropy of both dense pyrocarbon layers

HTR Fuel Specific Characteristics

Fuel Element:

- heat conductivity of the graphite matrix at 25 °C and 1000 °C
- ratio of defective SiC-layers (burn-leach test)
- corrosion rate
- crushing strengths
- fuel-free zone thickness
- abrasion rate

THTR-Fuel Element Production Experience

Produced:

Kernels: ~ 1000 batches

Coated particles: ~ 4000 batches

Fuel element: ~ 500 lots

(~ 1.000.000 FE)

Yield:

For each of these products > 95%

Reject:

1 lot of coated particles

1 lot of fuel elements

Safety:

Not a single safety relevant incident

Special Quality Assurance System and Philosophy

Development Targets:
- Target product specification
- Provide suitable manufacturing process
- provide suitable components

↓

Development Stage 1
Standard-Quality 1

↓

Development Stage 2
Standard-Quality 2

↓

Development Stage X
Standard-Quality X
Target product specification achieved

→ **Samples for screening test as part of production batches**

Special Quality Assurance System and Philosophy

TESSAG
NUKEM

- Target product Specification achieved, demonstrated by a "final" standard-quality
- essential targets for process and components achieved (close to the future production conditions)

→ Sample for reference test as part of the standard quality

KFA

Conclusions

1800 – 2500°C

- Increasing number of pressure vessel failure
- Additional to corrosion of SiC at 1800°C, above 2000°C SiC decomposition

Cs High release already at 1800°C, after heatup to 2500°C nearly total release

Kr (I) Release at 1800°C from single pressure vessel failures increases because of additional particle failures and diffusion through already destroyed SiC layer and still intact PyC layers up to 10% at 2500°C

Conclusions of core heatup simulation experimen with FE with UO2 TRISO particles(small HTR)

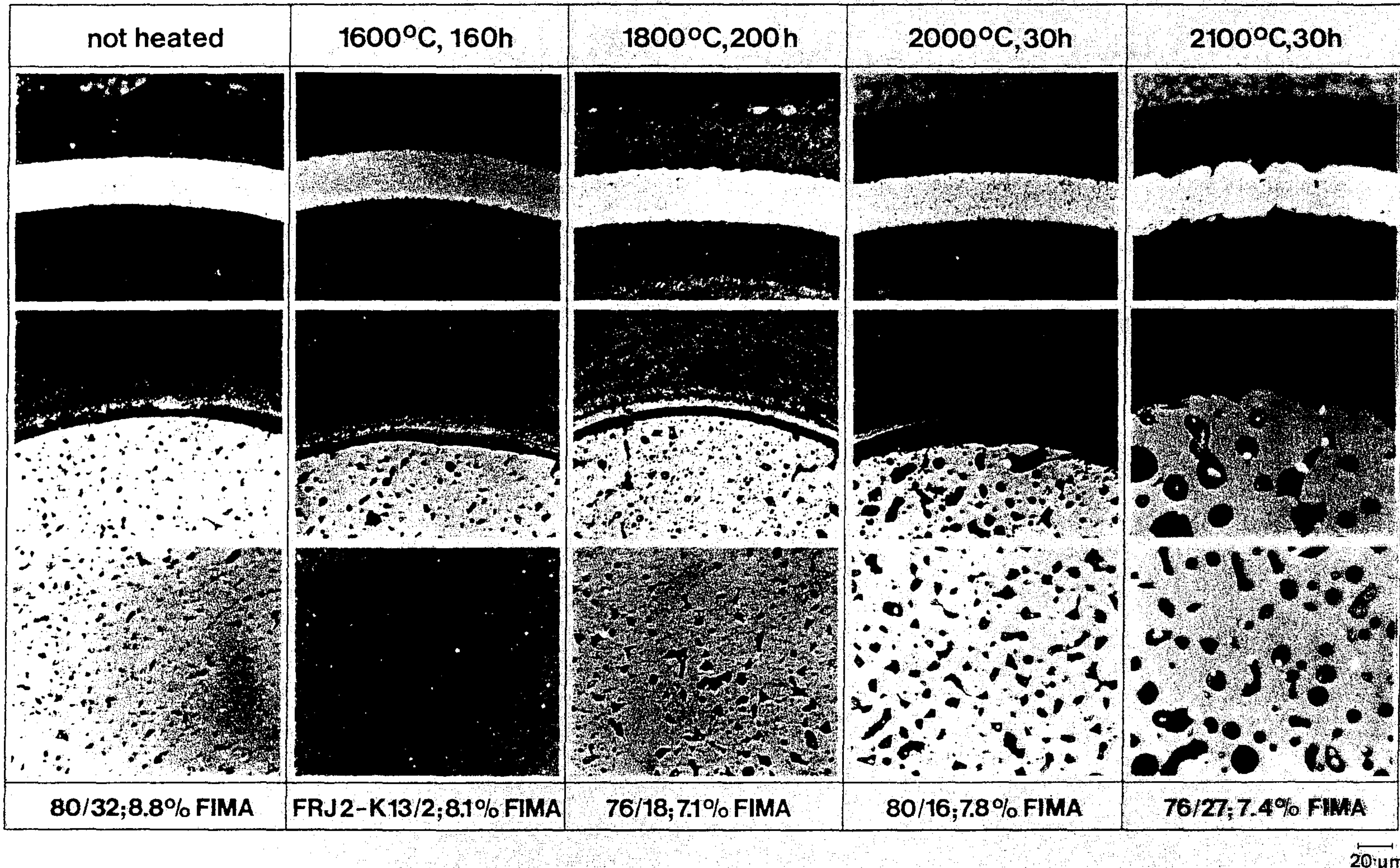
1600° C *F.p. release (except, Ag 110m) from <6E-5 free U from manufactoring*

Changing of SiC structure only after >100h

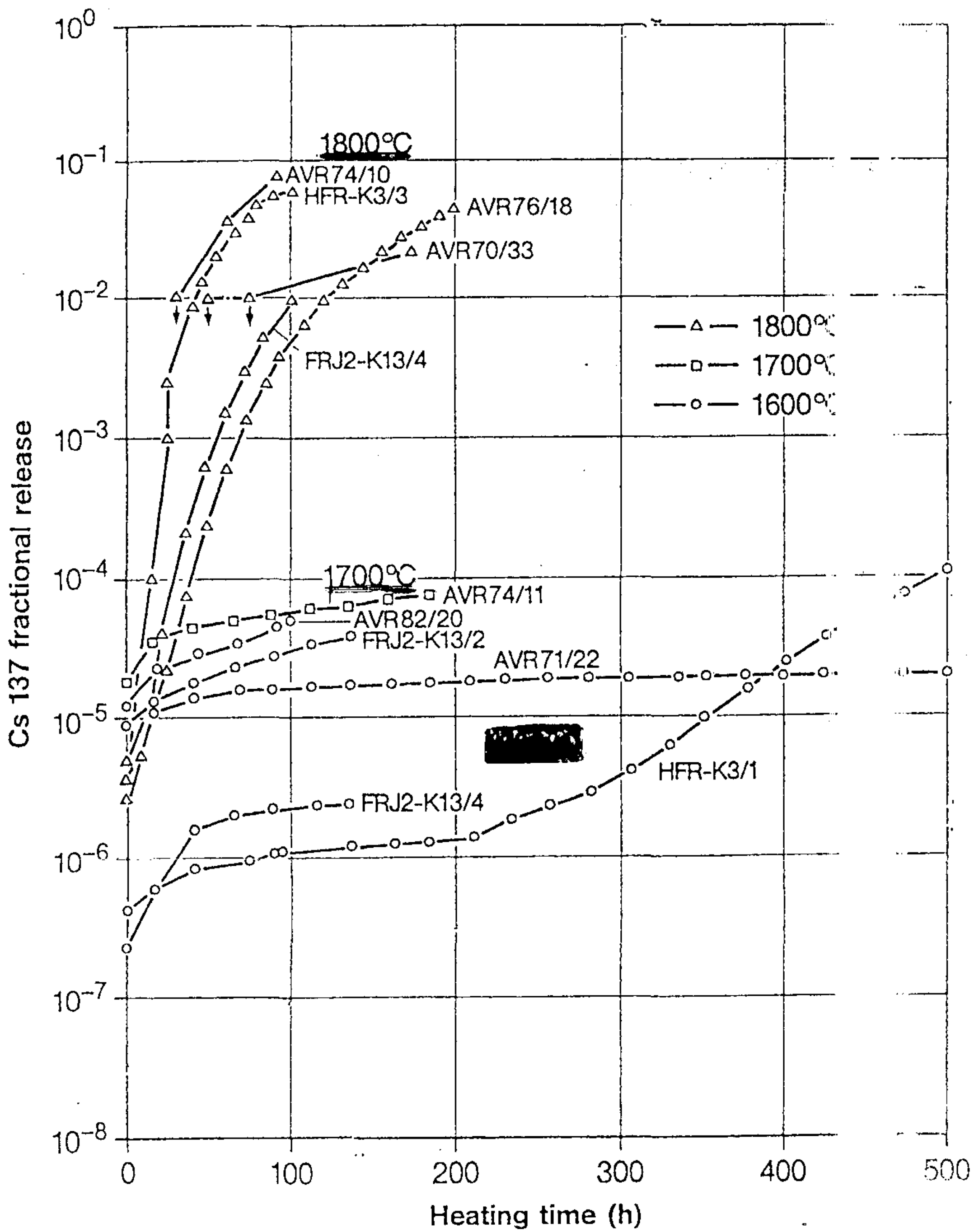
Single pressure vessel failures only after extreme irradiation conditions

1800° C *Single pressure vessel failures and changing of SiC structure lead to increasinc f.p. release in the order:*

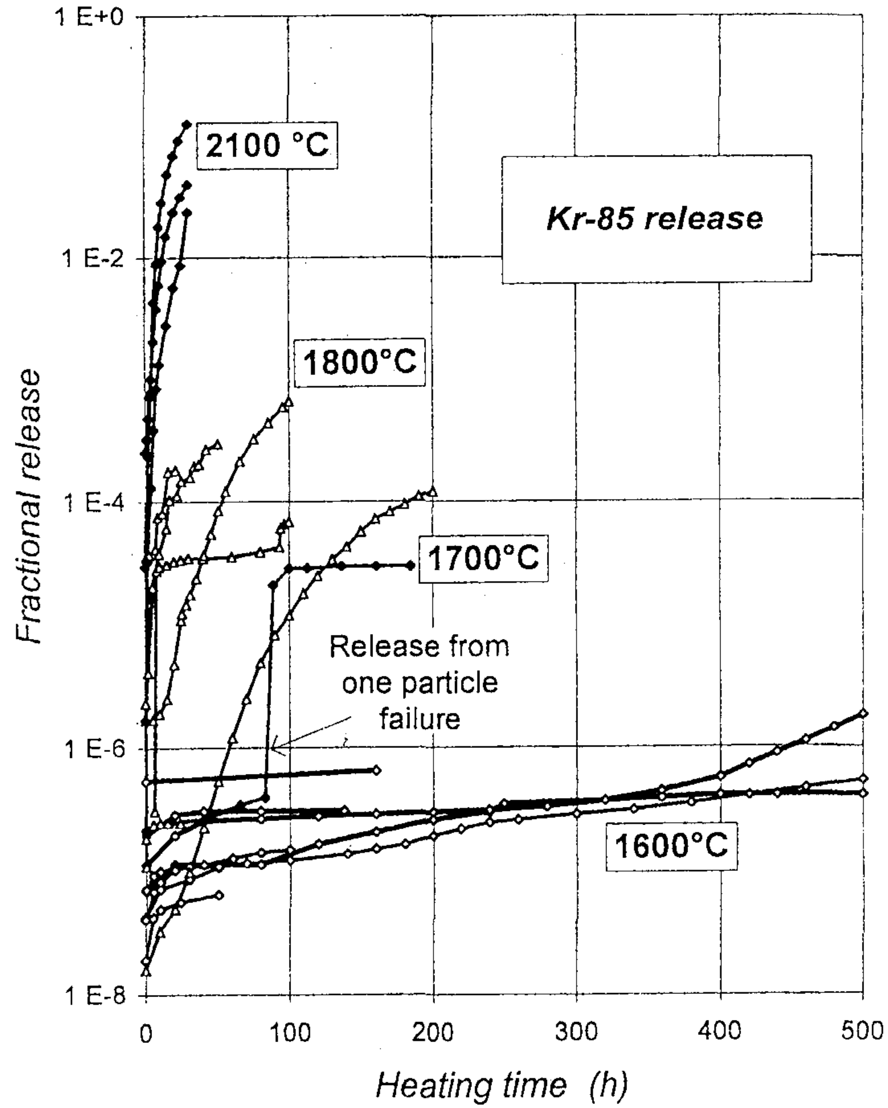
Cs
Sr
Kr / I



Ceramographic sections through UO_2 TRISO particles



Kr-85 release during heating tests with spherical fuel elements (9% FIMA)



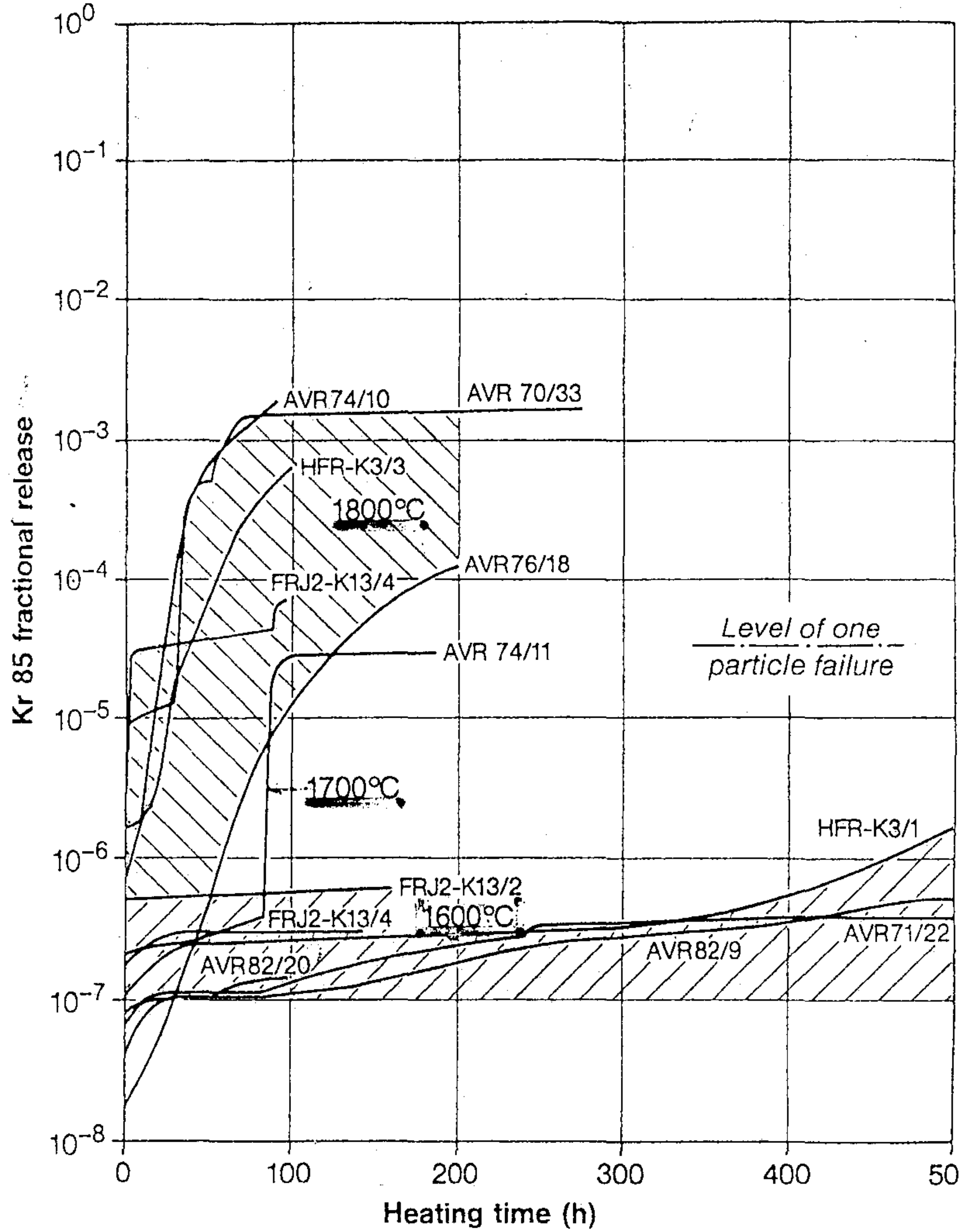
F.p. release during depressurization accidents in small HTR (MODUL)

Experimental data and calculations show Cs, Sr, and other solid fission products are retained in FE and core

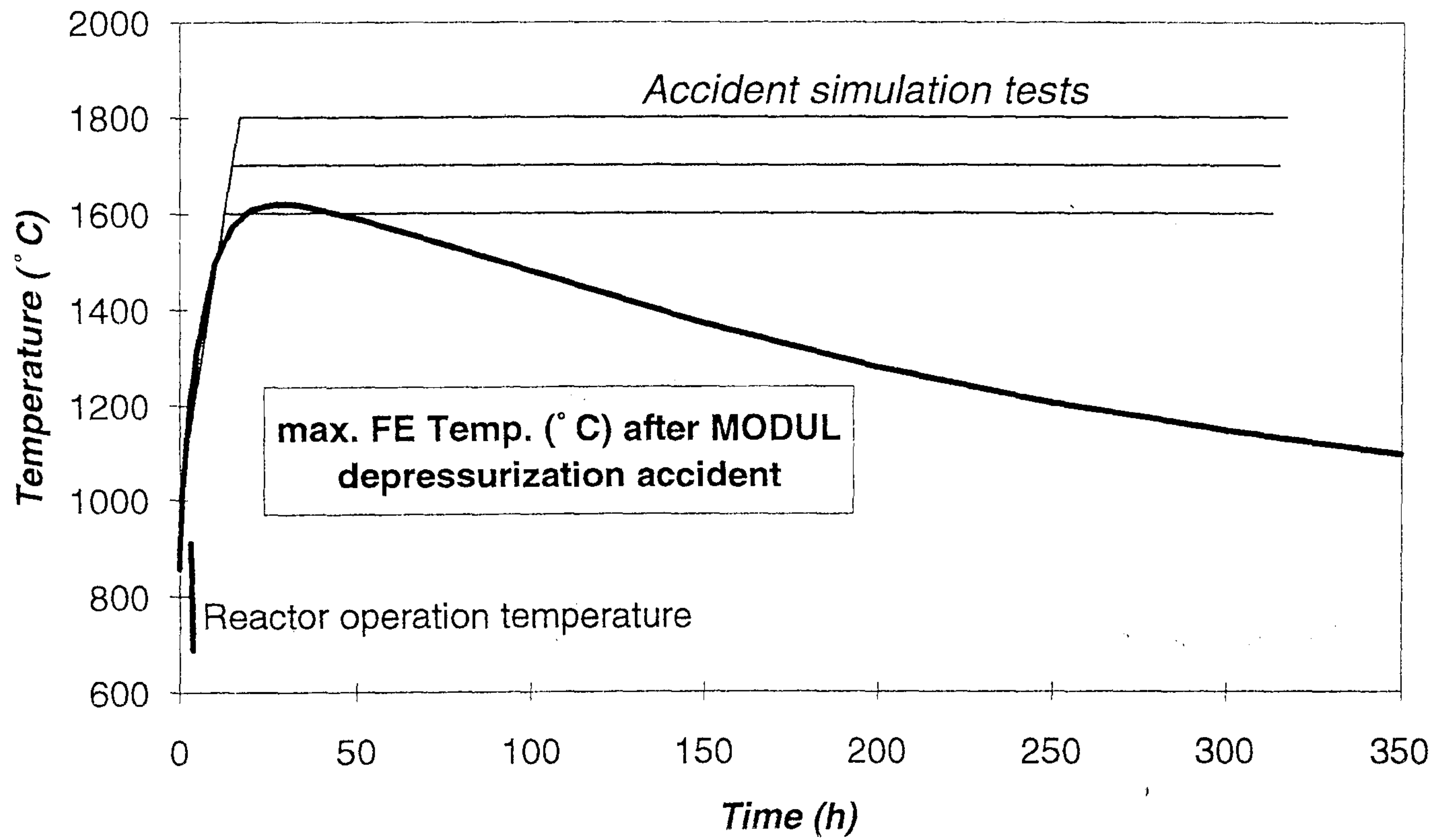
Most important is I release - I release depends on the number of failed particles

The number of defect particles from manufacturing and failed particles during irradiation and accident can only be determined by experimental work

There is a high dependence of failure fraction from particle quality



KFA Kr 85 release from fuel elements with UO_2 TRISO particles



Accident simulation tests
max. FE Temp. (° C) after MODUL depressurization accident
Reactor operation temperature
Temperature (° C)
Time (h)
2000
1800
1600
1400
1200
1000
800
600
0
50
100
150
200
250
300
350

Accident simulation at 1600 – 1800°C

Licensing procedure of MODUL

- Heating tests with UO_2 TRISO FE with high burnup
  - Particle failure fractions
  - Iodine release

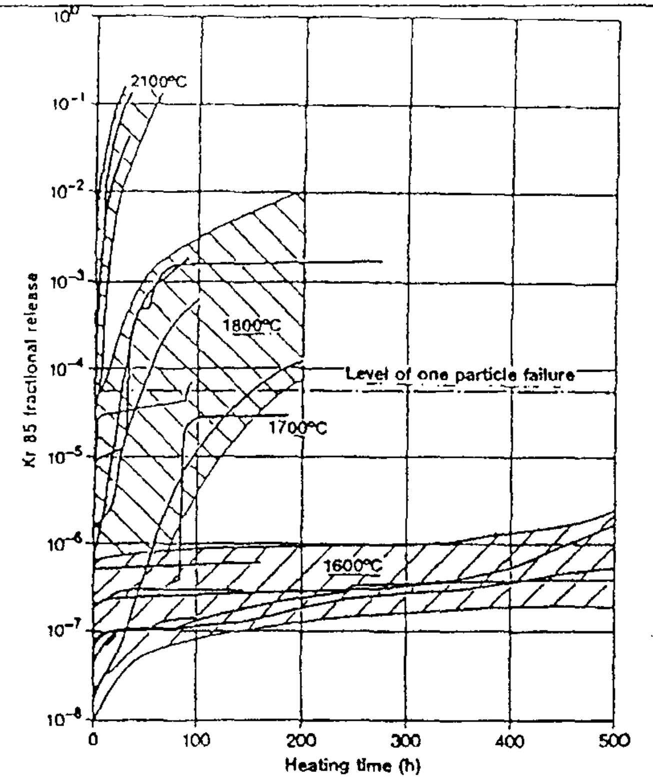
- Post heating examinations
  - Verification of release results
  - Fission product transport
  - Failure mechanisms

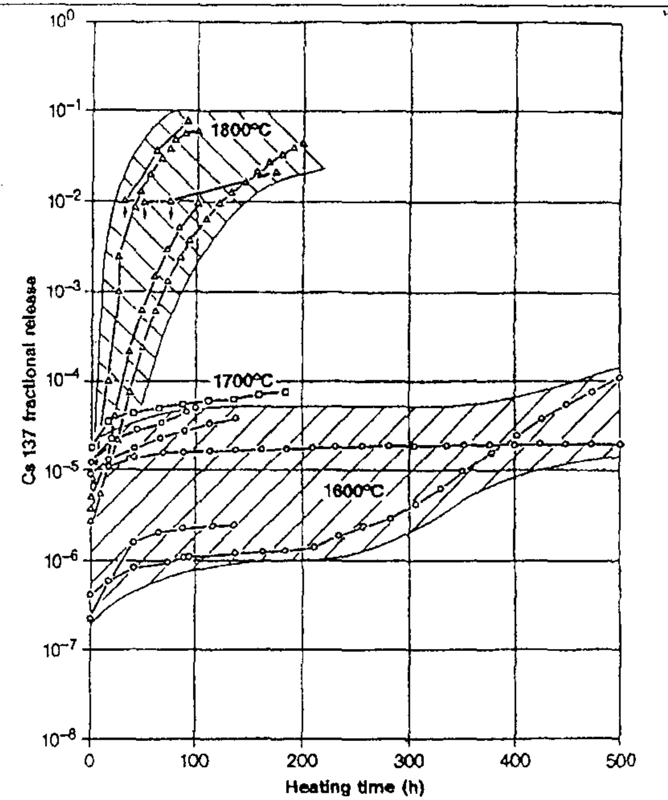
Utilization for licensing and model verification

Table 2: The fission products caesium, strontium and iodine are radiologically significant because, unlike the noble fission gases, they can be incorporated in the human body.

Important fission products		
ELEMENT	ISOTOPE	HALF LIFE
Solid fission products		
Caesium	Cs 137 Cs 134	30 years 2 years
Strontium	Sr 90	29 years
Iodine	I 131	8 days
Fission gases		
Krypton Xenon	Kr 85 Xe 133	11 years 5 days

Heating tests at 1600-2100°C





Krypton release during tests with irradiated spherical fuel elements at 1600 to 2100°C.	**Caesium release from heated spheres as a function of heating times up to 500 hours.**

The Work of Dr Werner Schenk

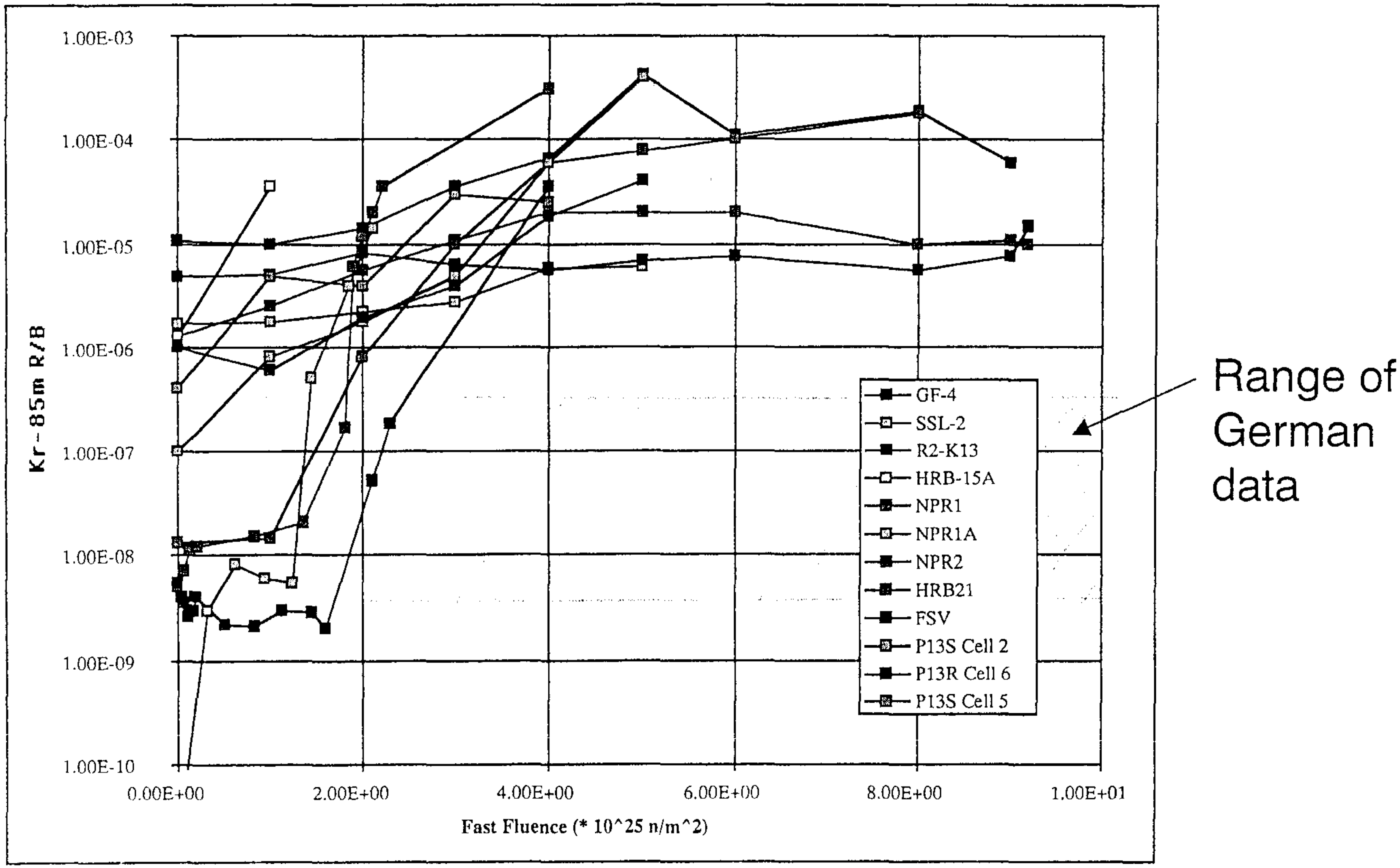
Suggested HTR fuel work, to be discussed:

(i) ^{110m}Ag: Re-evaluate release data during normal operations for better source term data base in direct cycle applications.

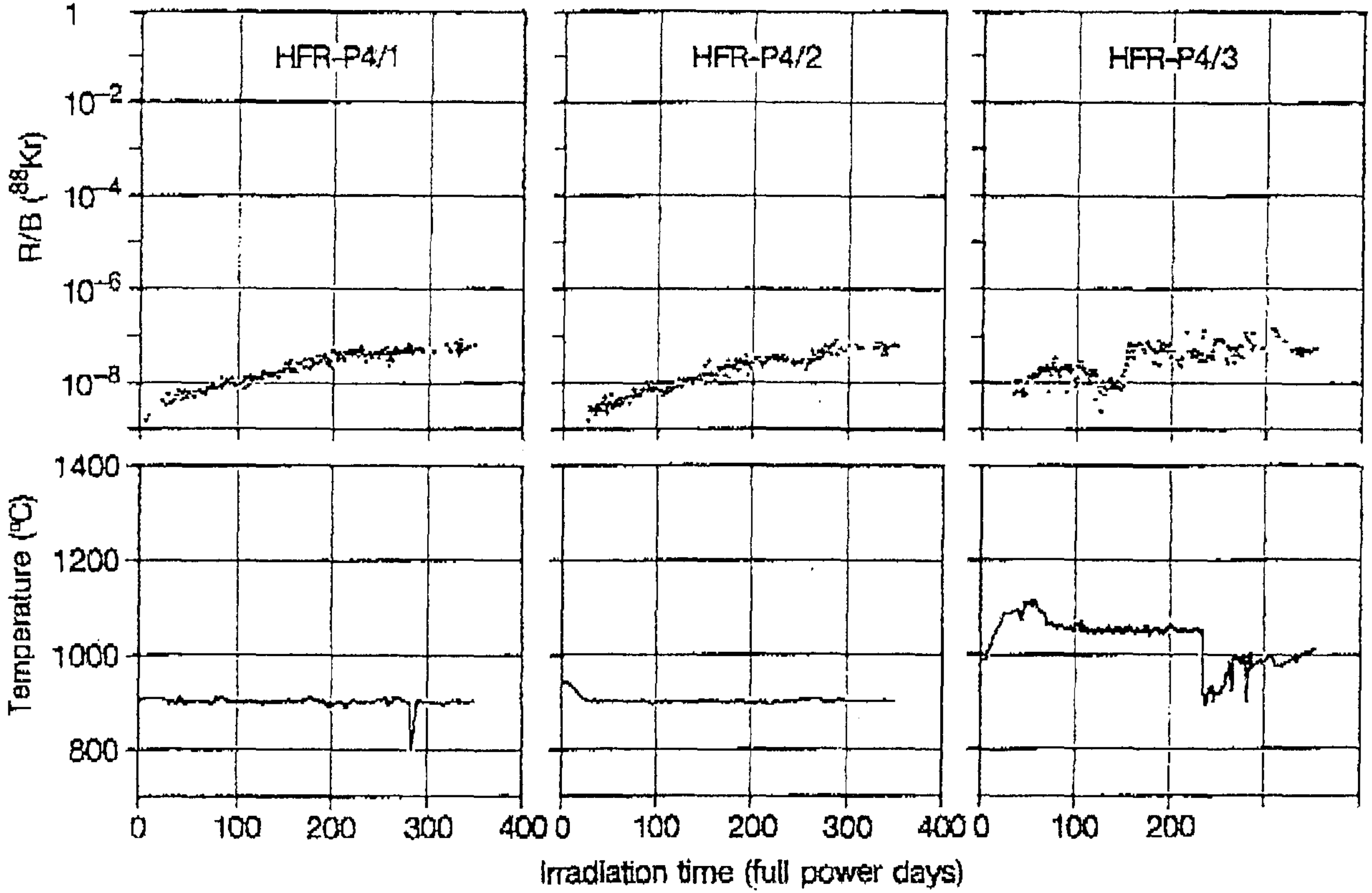
(ii) Determine influence of burnup > 10% FIMA on irradiation performance, in particular for potential reduction of 1600°C capability.

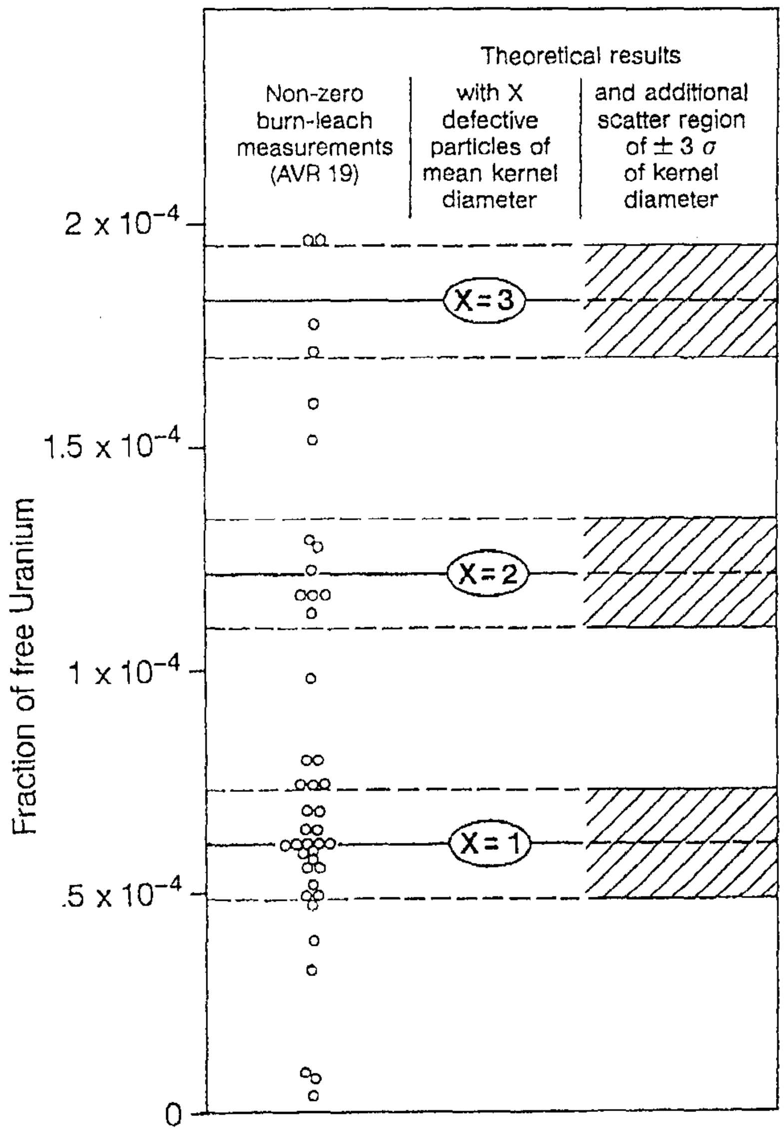
(iii) Analyse accident condition performance > 1600°C for an improved coated particle model.

Compilation of US Gas Reactor Fuel Behavior Experience



• Irradiation to near 15 % FIMA





Burn-leach with spherical fuel elements:

1. burn graphite and oPyC at 800°C
2. leach with HNO_3
3. determine U in solution

Diagram shows non-zero free uranium measurements in the seventy burn-leach tests from NUKEM quality control of the AVR 19 (GLE 3) production of 24,600 AVR fuel elements. This is a destructive test on 5 FEs per lot from the 14 lots in this production.

Measured free uranium corresponds to the contents of an integer number of coated particles; here zero, one, two or three out of 16,400 particles in a sphere.

HTR fuel: criteria required for ...

- Manufacture
- Irradiation tests ← normal operation conditions
- Heating tests ← off-normal conditions

Criteria for irradiation testing in order of relevance

1. temperature
2. burnup
3. fluence
4. power/ temperature gradients
5. transients
6. real time

Coated Particle Modelling		
Classical		**Alternate**
Pressure vessel models like PANAMA or STRESS3	Geometry + material properties: failure when gas pressure exceeds strength	Goodin-Nabielek for 1500-1800°C off-normal conditions
	Chemical effects by thinning of coating layers	Ogawa et al. for 1800-2400°C extreme accidents
FRESCO	Set of diffusion coefficients determines release from intact, defect and broken particles	...

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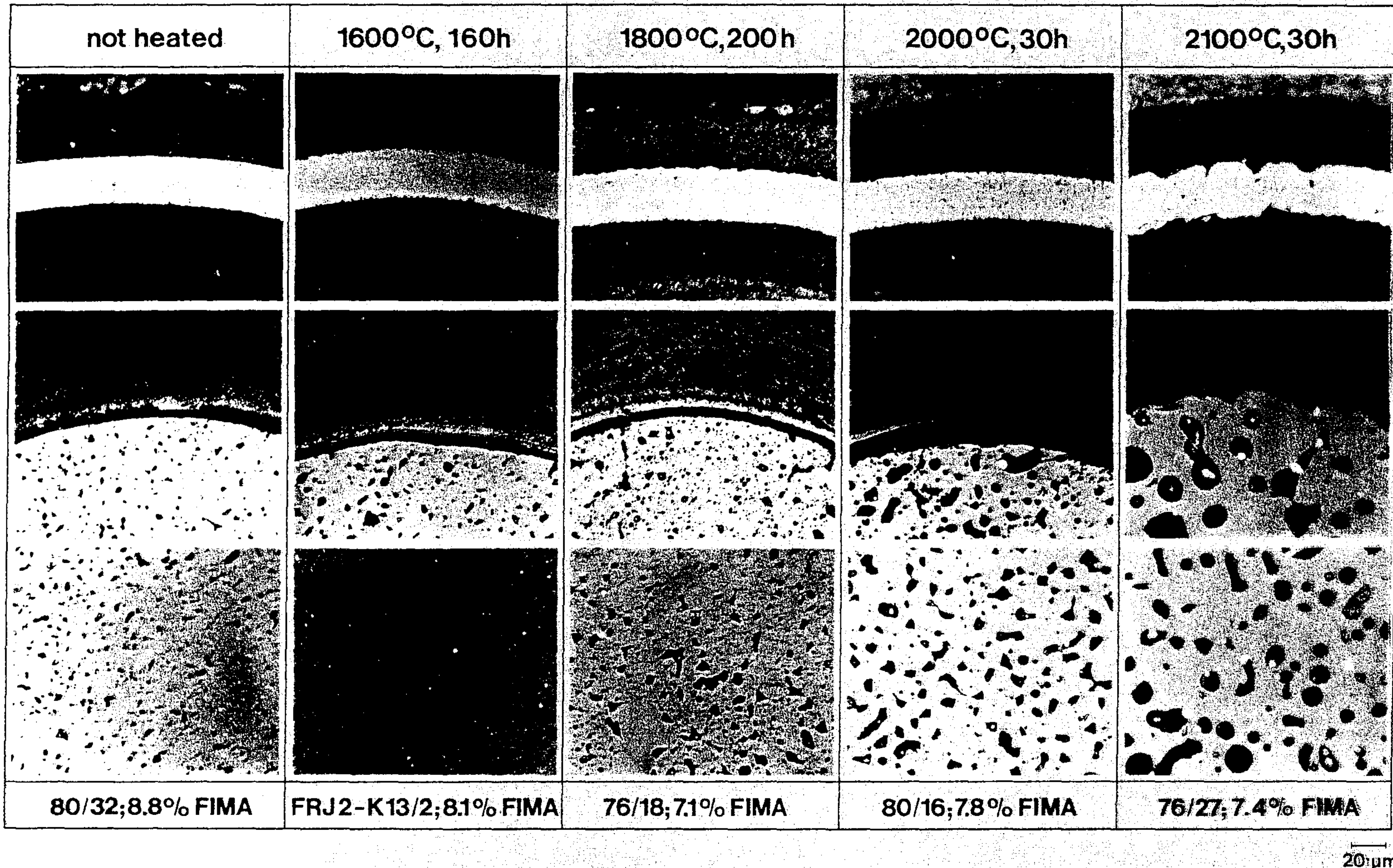

KFA

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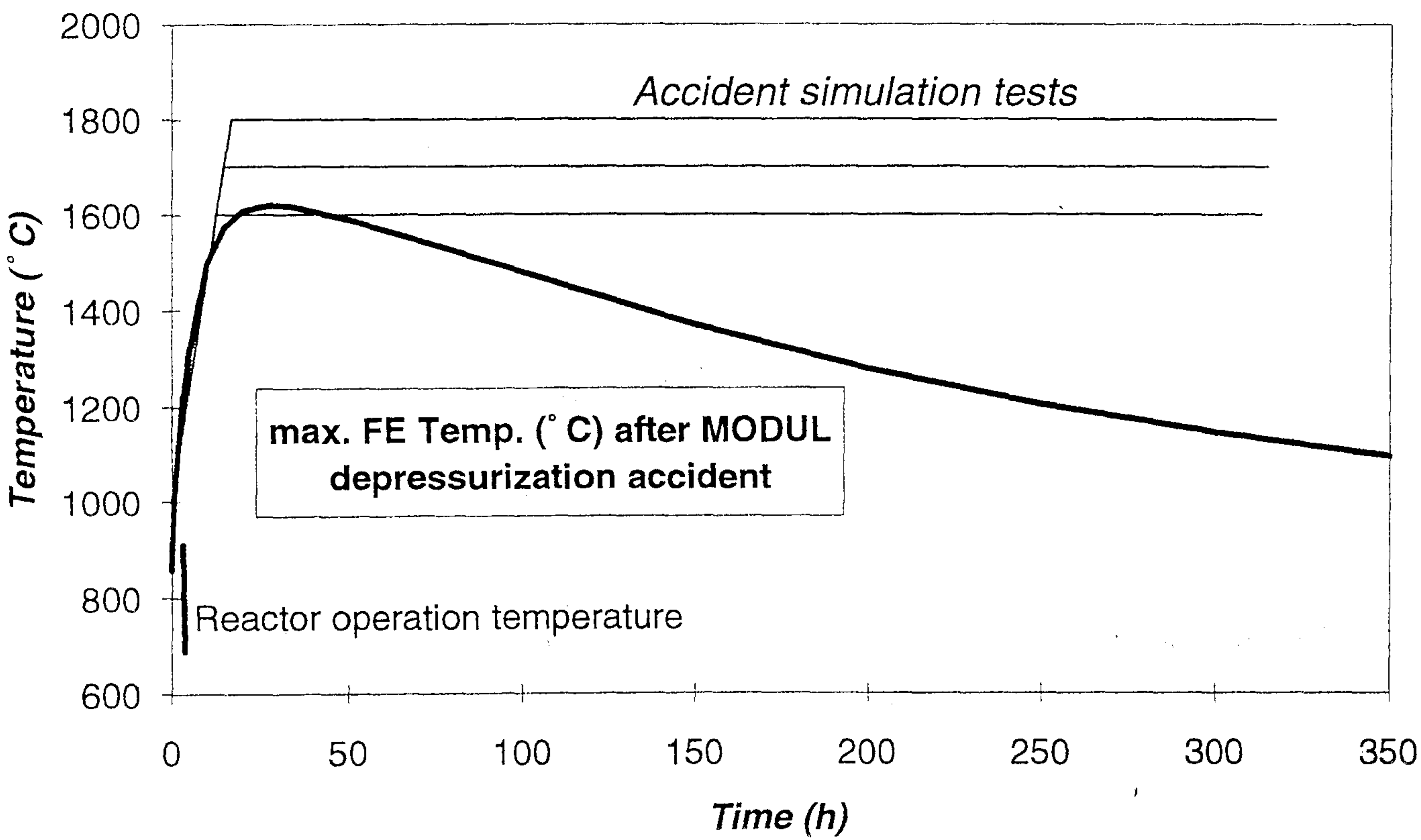

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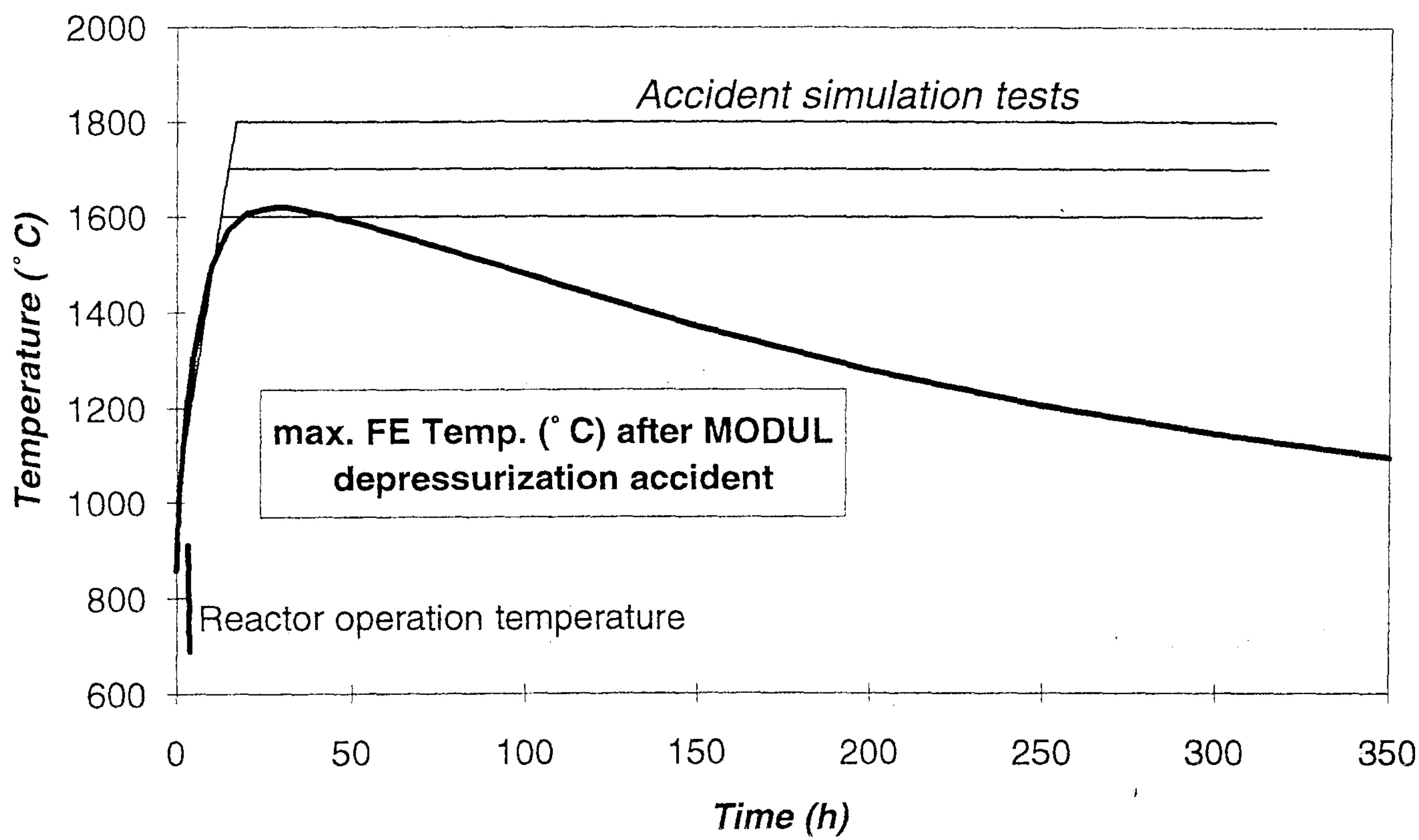

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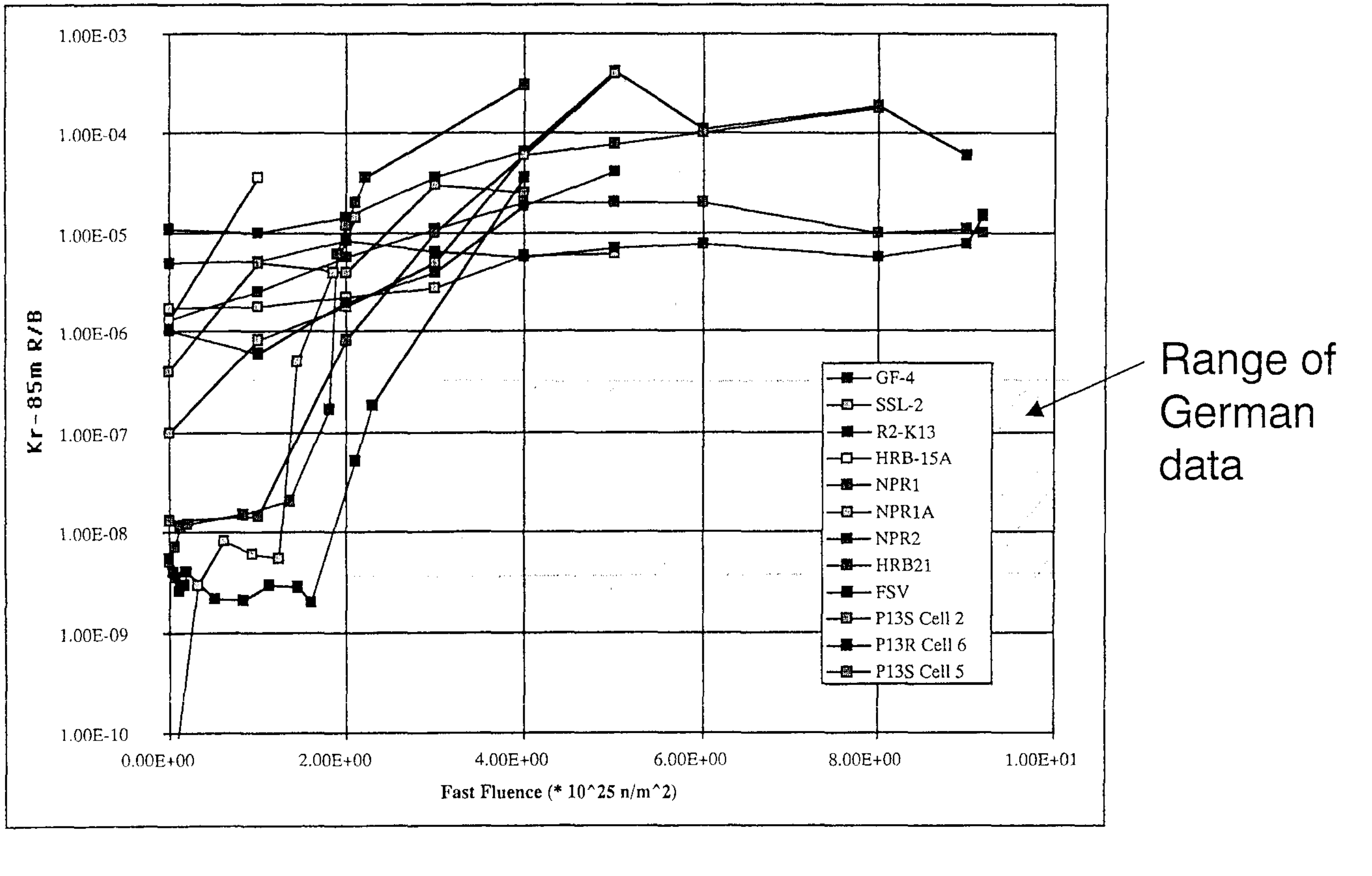
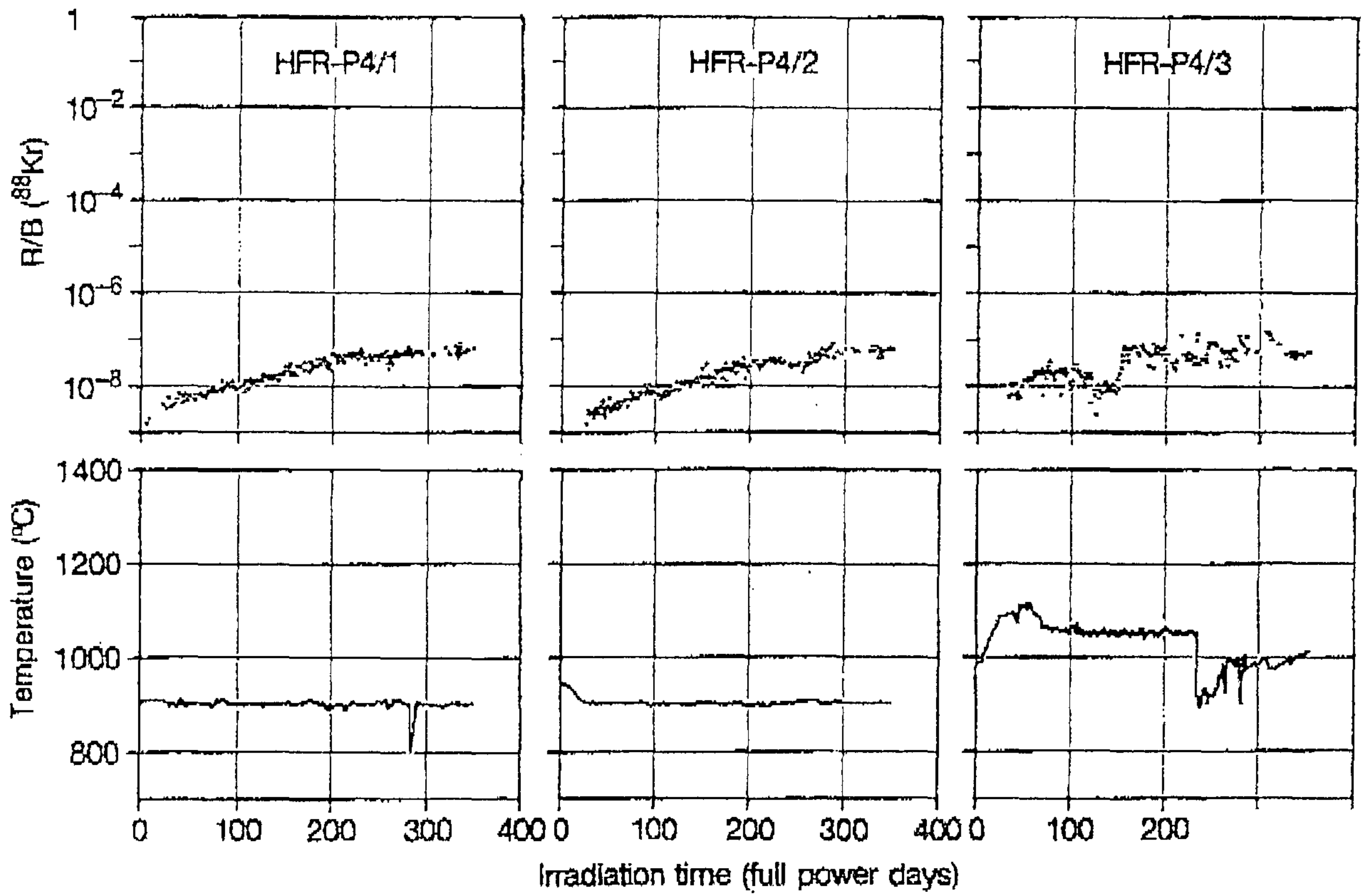
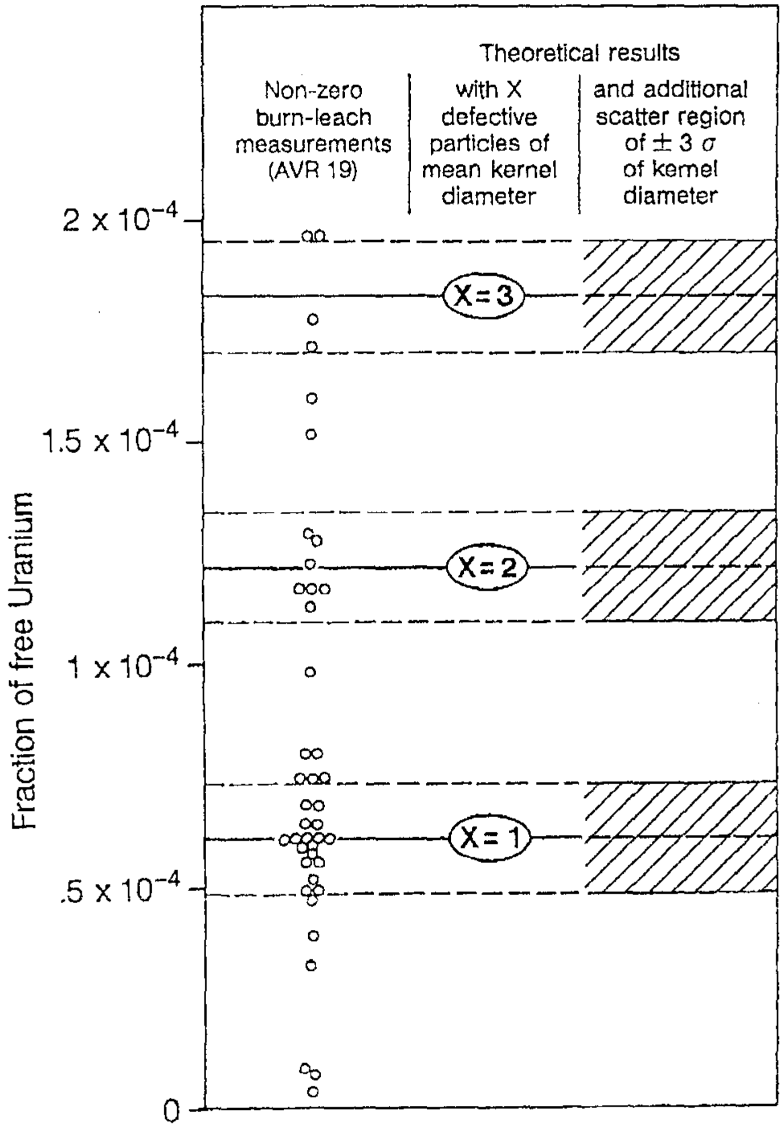
6. real time

Compilation of US Gas Reactor Fuel Behavior Experience



• Irradiation to near 15 % FIMA





Burn-leach with spherical fuel elements:

1. burn graphite and oPyC at 800°C
2. leach with HNO_3
3. determine U in solution

Diagram shows non-zero free uranium measurements in the seventy burn-leach tests from NUKEM quality control of the AVR 19 (GLE 3) production of 24,600 AVR fuel elements. This is a destructive test on 5 FEs per lot from the 14 lots in this production.

Measured free uranium corresponds to the contents of an integer number of coated particles; here zero, one, two or three out of 16,400 particles in a sphere.

HTR fuel: criteria required for ...

- Manufacture
- Irradiation tests ← normal operation conditions
- Heating tests ← off-normal conditions

Source terms

for fission products into the primary circuit of an HTR are:

(i) heavy-metal contamination;
(ii) particle defect and/ or failure;
(iii) release from intact particles.

Sequence of release is

^{110m}Ag, ^{137}Cs, ^{134}Cs, ^{85}Kr, ^{90}Sr, ^{106}Ru, ^{95}Zr

Visit of NRC delegation to Germany
Forschungszentrum Jülich, 24 July 2001

HTR Fuel
Manufacture, Irradiation and Accident Condition Testing

By

Heinz Nabielek, Forschungszentrum Jülich, Germany

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HTR FUEL
MANUFACTURE, IRRADIATION AND ACCIDENT CONDITION TESTING

By

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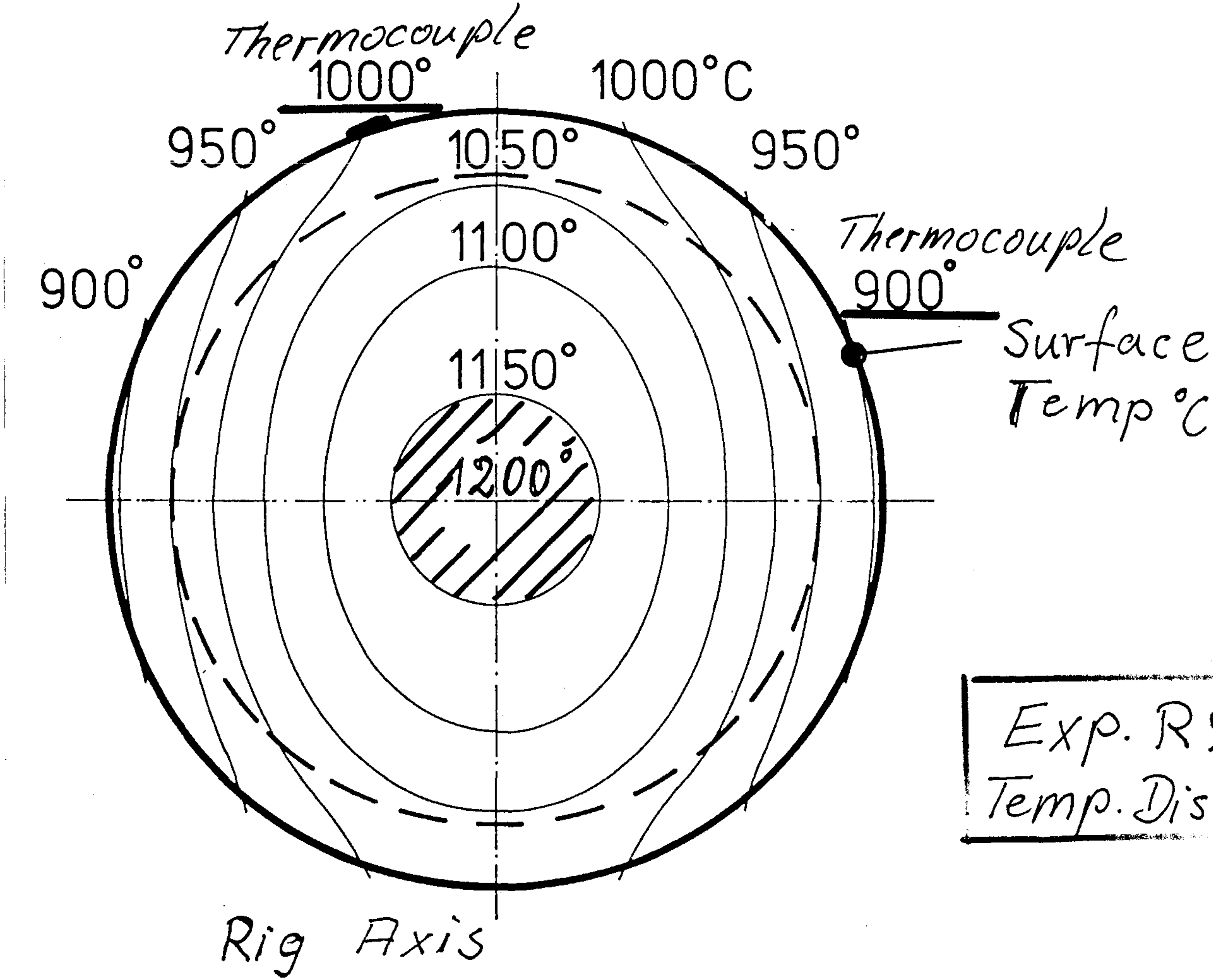


Exp. R2-K3
Temp. Distribution

Forschungszentrum Jülich GmbH
FZJ

FUEL PEBBLES OPERATIONAL EXPERIENCES
IRRADIATION AND POSTIRRADIATION EXAMINATION

G. Pott

H. Nabielek

Jülich, 09.July 01

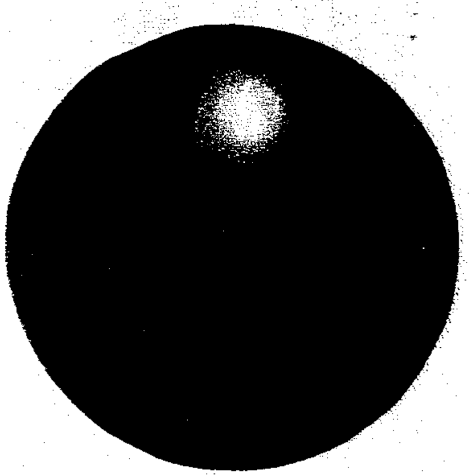
- Reference fuel ,TRISO coated particles
- Irradiation tests in research reactors
- PIE, heating tests

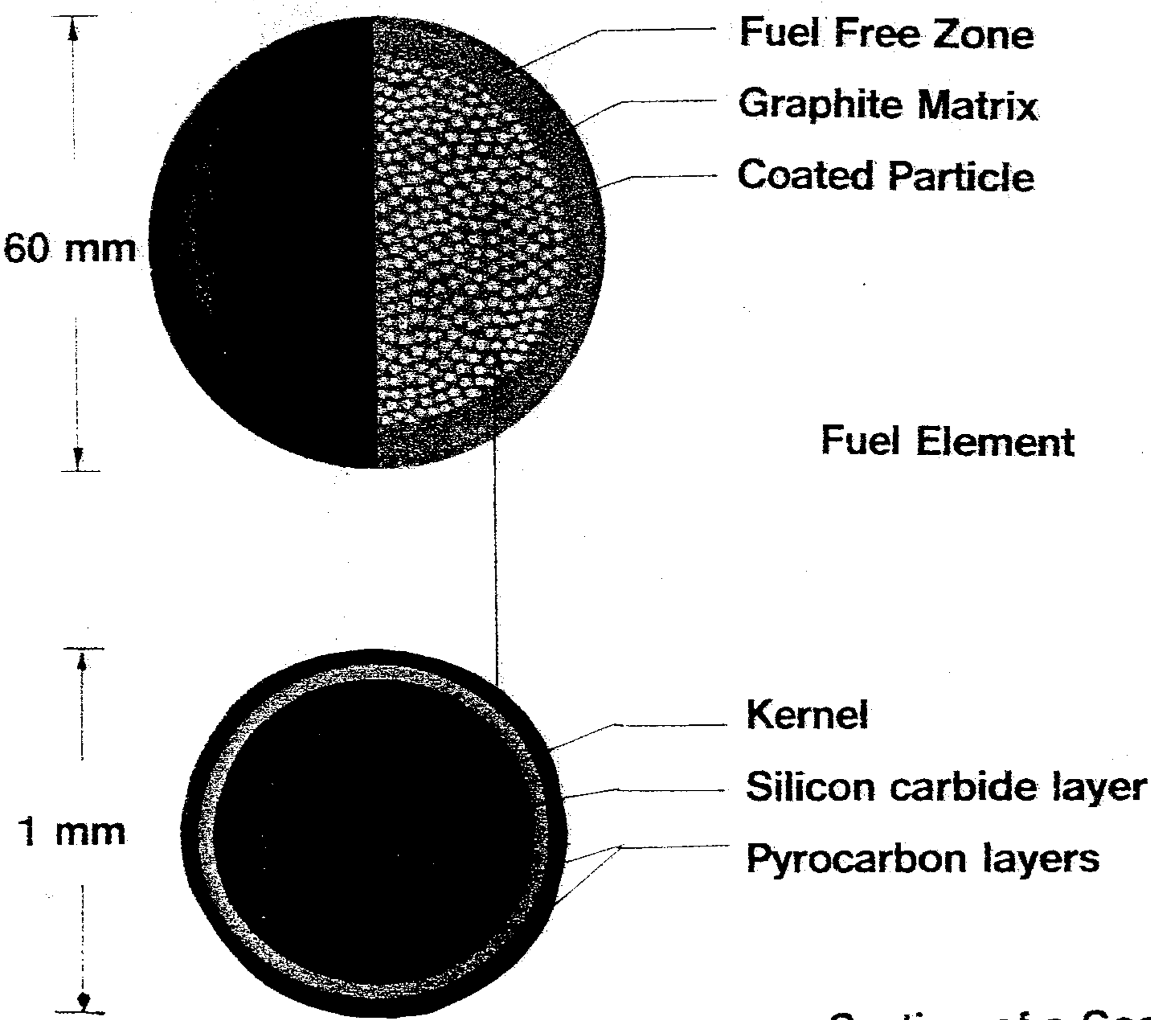
Executive Statements Summary

1. The design of modern HTRs is based on high qualified fuel. This fuel designed in the 1960s and 1970s had been perfected for steam cycle applications in the 1980s and early 1990s enabling the design of small inherently safe modular HTRs with self-limiting temperatures of < 1600 ^{0}C.
2. In the past for normal reactor conditions, irradiation testing has been performed in material test reactors and in the AVR. Parameters such as burn up, operating temperature and fast neutron fluence are varied to assess fuel performance. Continuous monitoring of released fission gas during irradiation tests gave a direct indication of the integrity of fuel coatings.
3. **In the German program, relevant irradiation tests with more than 2x10^5 particles were performed without a single coated particle failure during irradiation. Statistically, this result corresponds to a 95% confidence level that the coating failure fraction is less than 2x10^{-5}.**

4. Postirradiation examinations had been carried out in the FZJ – Hot Cell Laboratories. One of the most important examination method are the heating tests for simulating accident conditions in special designed and constructed furnaces.(e.g. **KÜFA** *cold finger furnace*) These tests under off-normal conditions has provided fuel performance information as a function of burn up, fast neutron fluence, heating time and temperature up to 2500 ^{0}C.
5. **Kr 85 gas release fractions during accident condition testing up to 1600 ^{0}C were low at <10^{-6}, even at 1800 ^{0}C for 50-100 h. With >11% FIMA fuel, release remains at this low level throughout a 350 h test at 1600 ^{0}C . At 1800 ^{0}C, 10^{-3} release fractions are reached as a consequence of diffusion through degraded SiC.**
6. **At 1600 ^{0}C the fuel does not suffer irreversible changes and continues to retain all safety- relevant fission products (e.g. Cs, I, Sr). Ag 110m diffuses at 1200 – 1600 0C through intact SiC, but the amount of the generated silver is low.**
7. Know how transfer with ESCOM representatives is going on by the author. **Additional experiments** should be performed with higher temperatures, longer heating time and with fuel from accelerated tests to establish the performance margins under accident conditions of new designed reactors. This means also to irradiate actual fuel produced for the new ESCOM reactors.

SPHERICAL FUEL ELEMENT





Fuel Element

Section of a Coated Particle

IRRADIATION QUALIFICATION OF HTR FUEL ELEMENTS

- TEST FOR DETERMINATION OF PARTICLE DEFECT RATES UNDER CONDITIONES EXCEEDING NORMAL OPERATING CONDITIONS

  800-1200° C

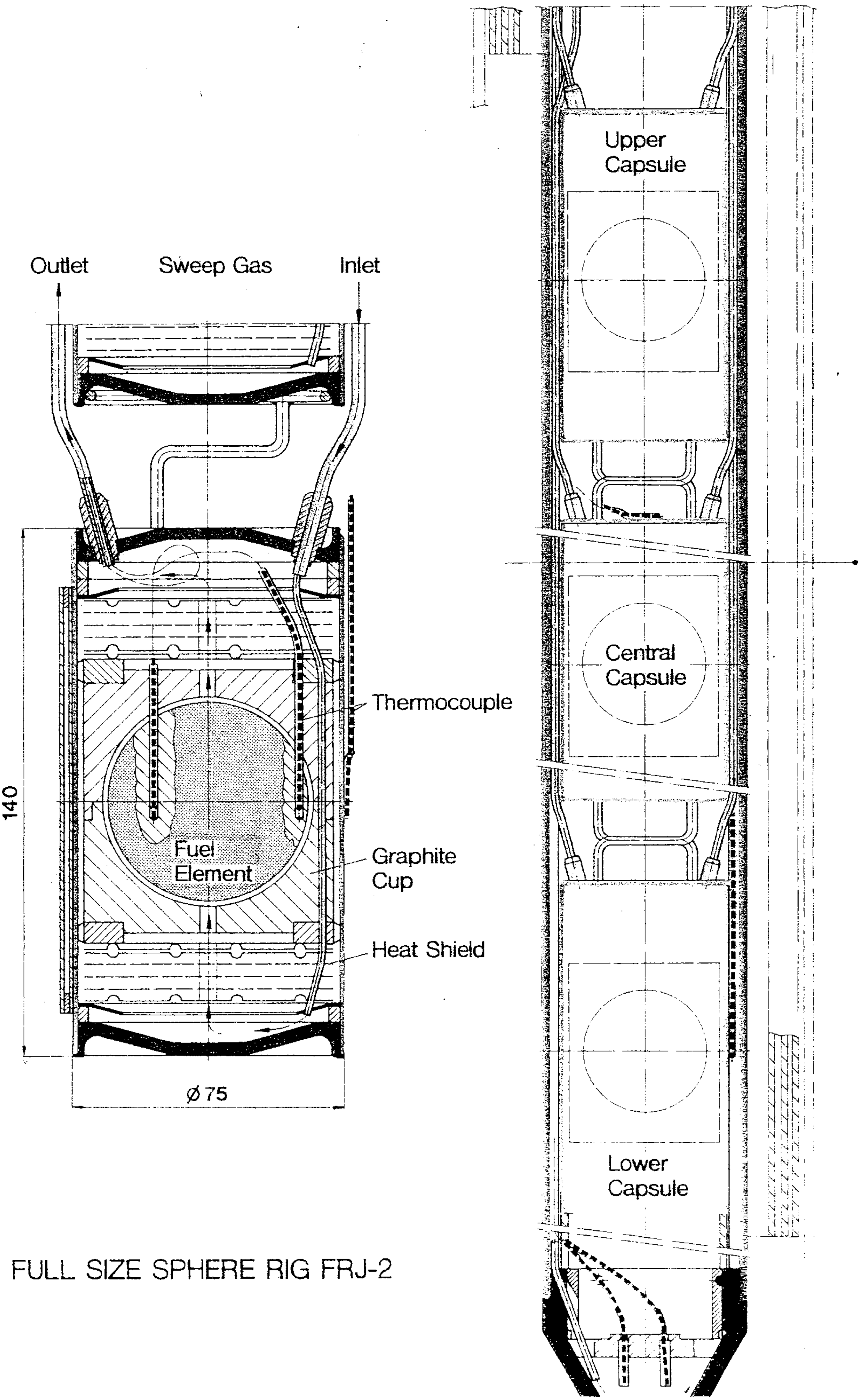
- IRRADIATIONS OF FUEL PARTICLES WITH KNOWN FAILURE FRACTION

  800-1300° C

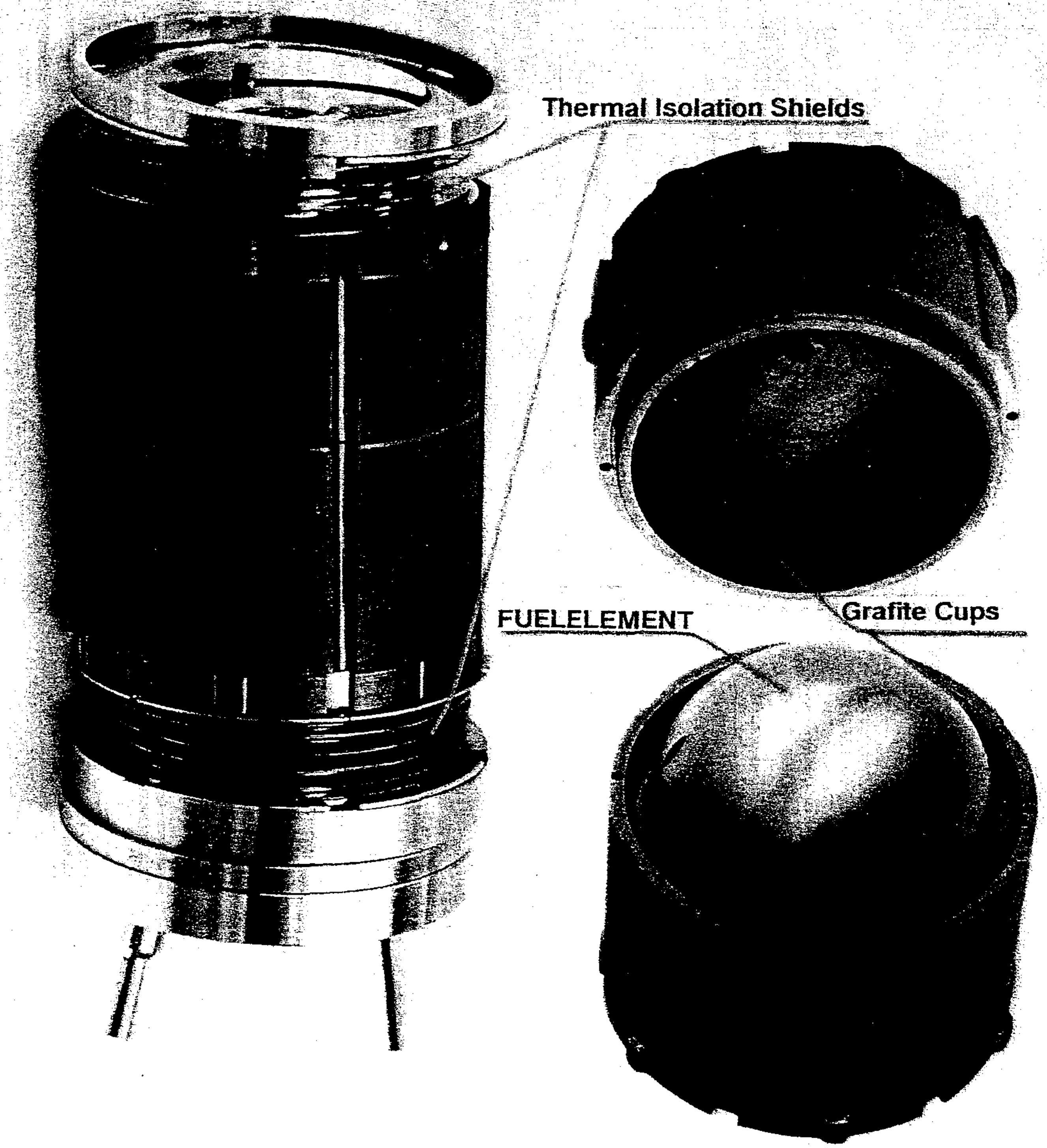
- TEST FOR DETERMINATION OF BURN UP INFLUENCES ON DEFECT RATES

  1000-1200° C

- FUEL ELEMENT REFERENCE TESTS



FULL SIZE SPHERE RIG FRJ-2

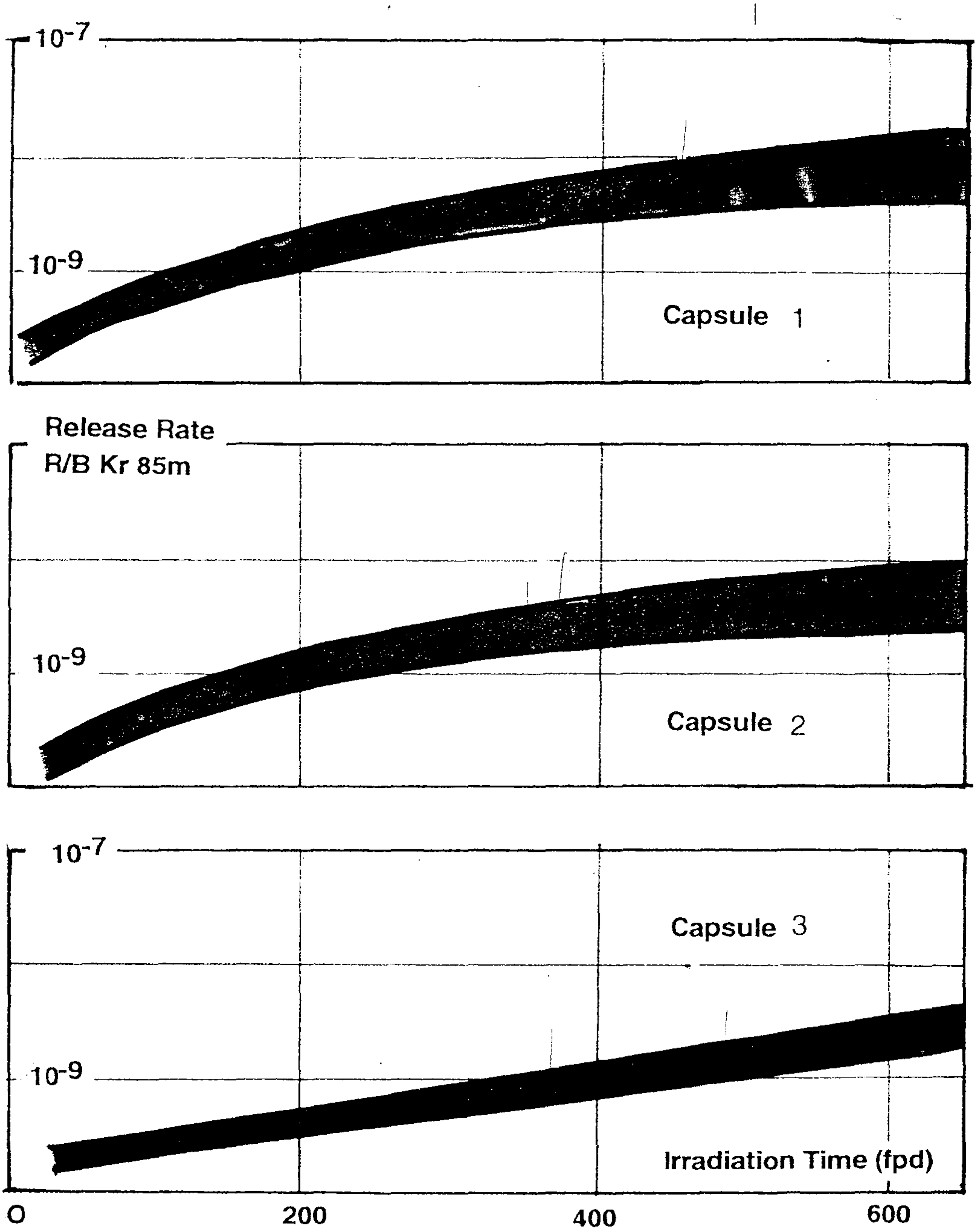


HTR FULL SIZE SPHERE RIG

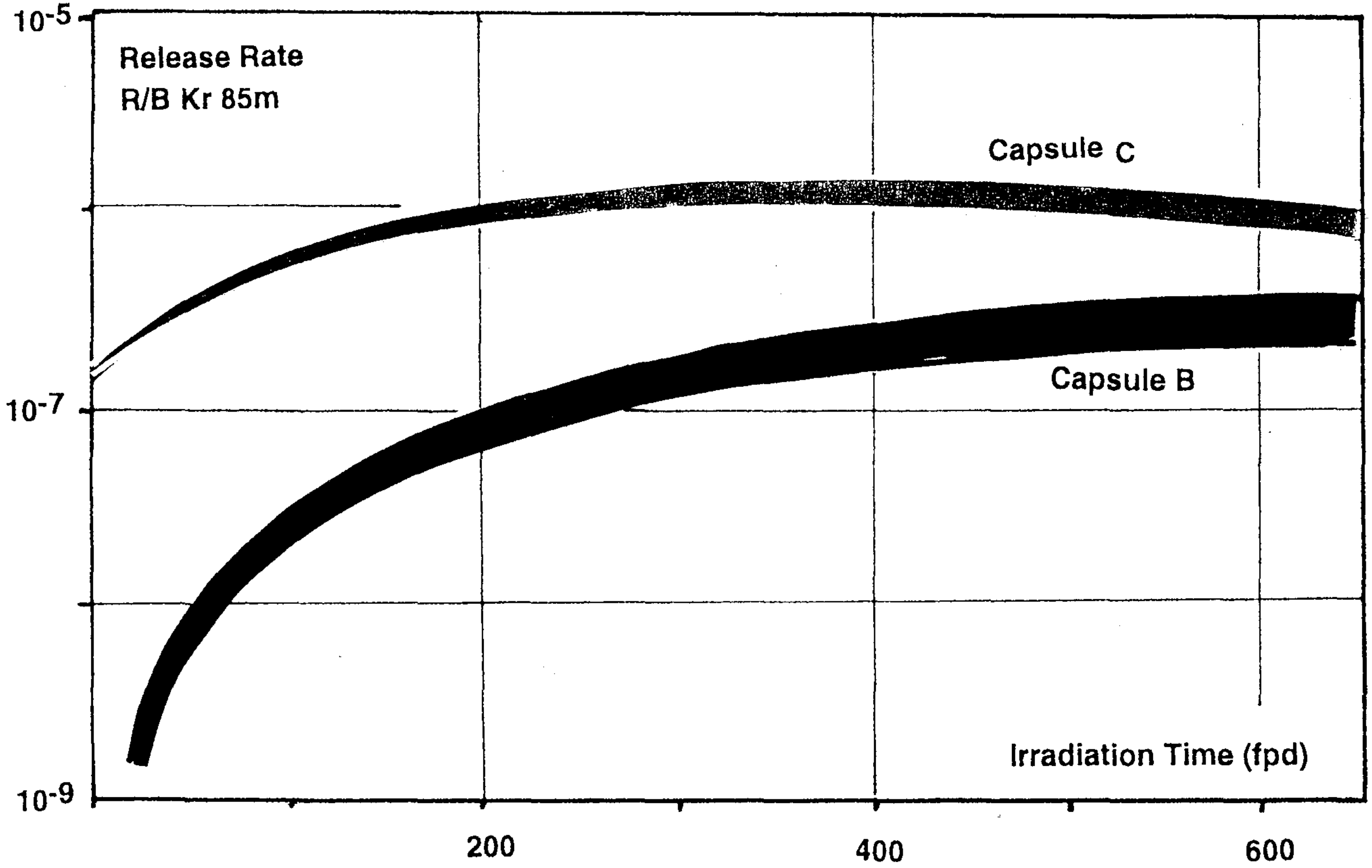
CAPSULE COMPONENTS

Experiment FRJ 2-K15
- Irradiation-Data -

Capsule Nr.	1	2	3
Irradiation time (fpd)		650	
Burn up (% fima)	14,8	16,0	15,2
Fast Neutron Fluence (E > 0,1 MeV x 10^{25}) m^{-2}	0,2	0,3	0,2
Fuel Element Power (kW) Start	1,9	2,2	2,0
Fuel Element Power (kW) End	0,6	0,6	0,6
R/B Kr 85m Start	2 E -10	2 E -10	1 E -10
R/B Kr 85m End	1,2 E -8	4 E -9	3 E -9
Power / CP (W)		0,23 / 0,06	
Fuel Element Surface Temp. (°C)	800	900 - 1000	800



Experiment FRJ 2-K15 - Fission Gas Release



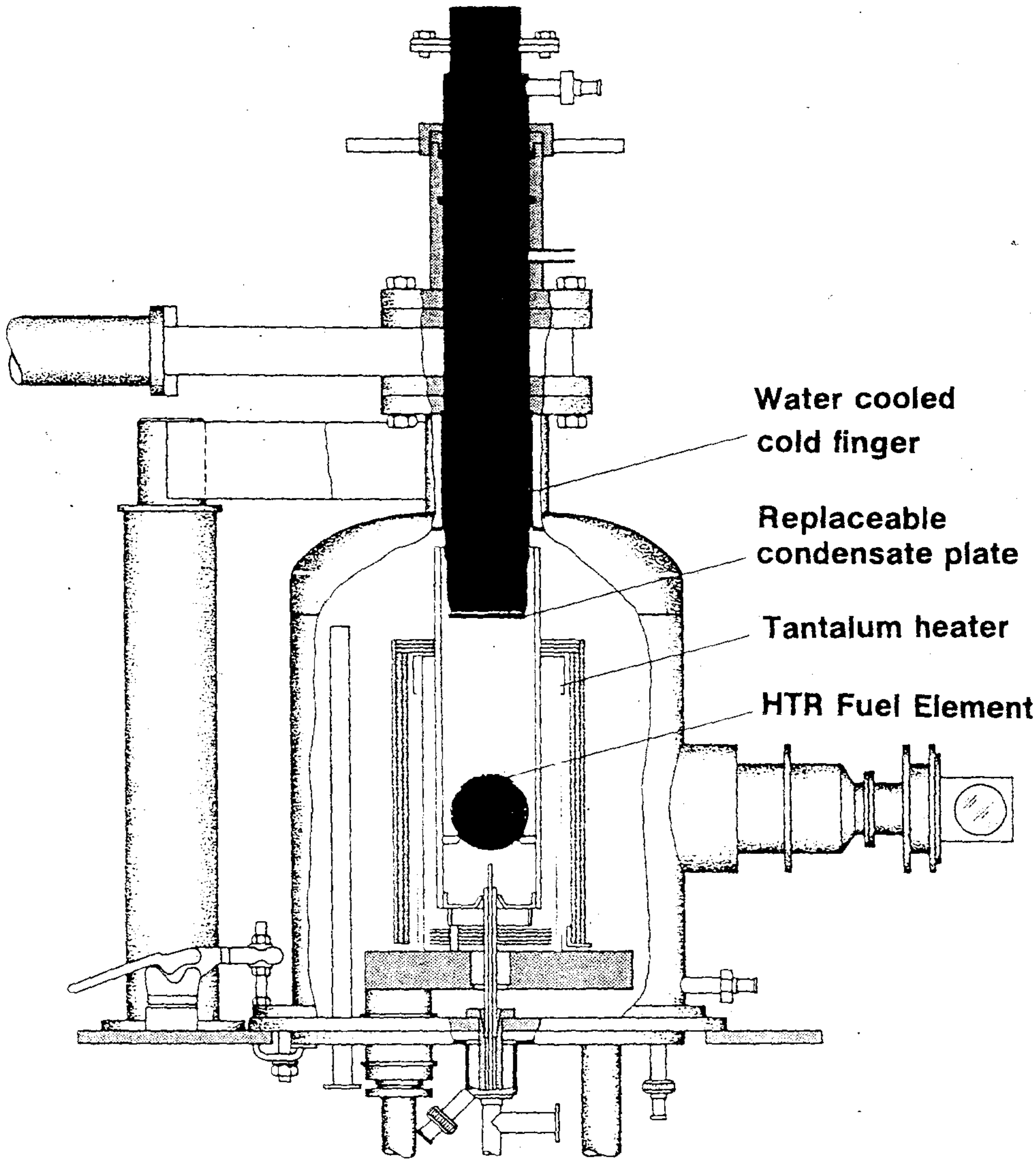
HTR REFERENCE TEST , FISSION GAS RELEASE
(HFR-K6)

Experiment	Specimens per capsule	Number of particles	Irrad. time (efpd)	Temp. centre (°C)	Burnup (% FIMA)	Fluence (10^{25} m^{-2} E>16 fJ)	In-pile Release R/B ^{85m}Kr
HFR-P4	36 small sph.	58800	351	940-1075	9.6-14.9	5.5-8.0	8E-8 - 9E-8
SL-P1	12 small sph.	19600	330	800	8.6-11.3	5.0-6.7	1E-6
HFR-K3	4 fuel sph.	65600	359	920-1220	7.5-10.6	4.0-5.9	1E-7 - 3E-7
FRJ2-K13	4 fuel sph.	65600	396	1120-1150	7.5-8.0	0.2	2E-9 - 2E-8
FRF2-K15	3 fuel sph.	28800	533	970-1150	14.1-15.3	0.1-0.2	3E-9 - 1E-6
FRJ2-P27	3 compacts	22020	232	1080-1320	7.6-8.0	1.3-1.7	1E-7 - 1E-5
HFR-K5	4 fuel sph.	58400	359	cycled	6.7-9.1	4.0-5.9	1E-7 - 3E-7
HFR-K6	4 fuel sph.	58400	359	cycled	7.2-9.7	4.0-5.9	1E-7 - 3E-7

Parameters and results from irradiation tests with modern UO_2 TRISO fuel

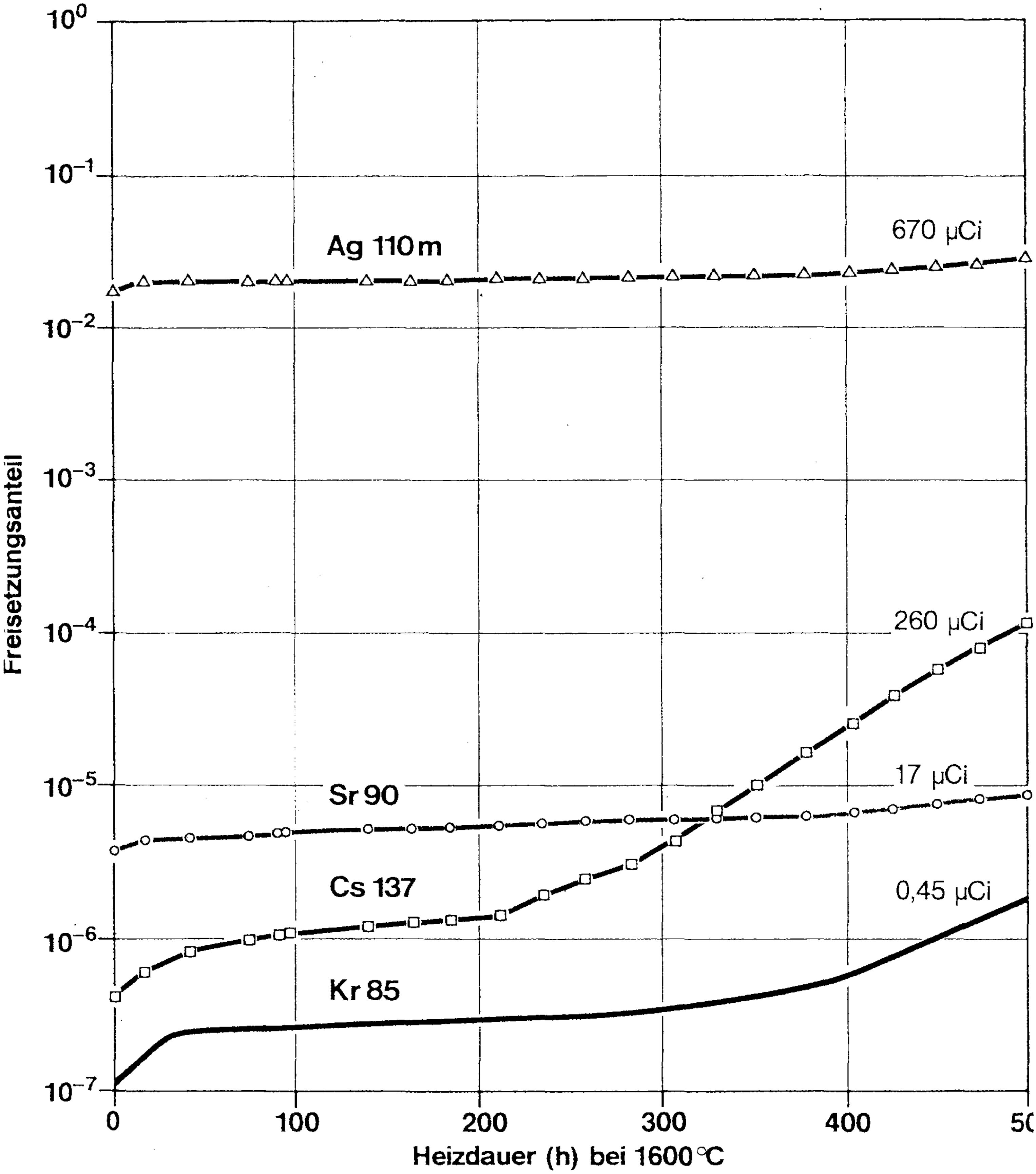
Post Irradiation Examinations for HTR Fuel Elements

- Neutron Radiographie (Irradiation Device)
- Gamma Scan (Flux Distribution)
- Examination of Neutron Fluence Monitors
  (Fast Fluence, Burn up)
- Dismantling of Rig and Capsules
- Inspection, Photodocumentation
- Dimensional Measurements of Fuel Ball
- Burn up Measurement (Comparison with Calculation)
- Gamma Spectrometrie - Fission Product Distribution
  (Fuel element, Components)
- Corrosiontest
- Compressive Strength (generally not necessary)
- Ceramographie / REM
- Accident-Simulation-Tests
  (Corrosion, High Temperature >1600°C)

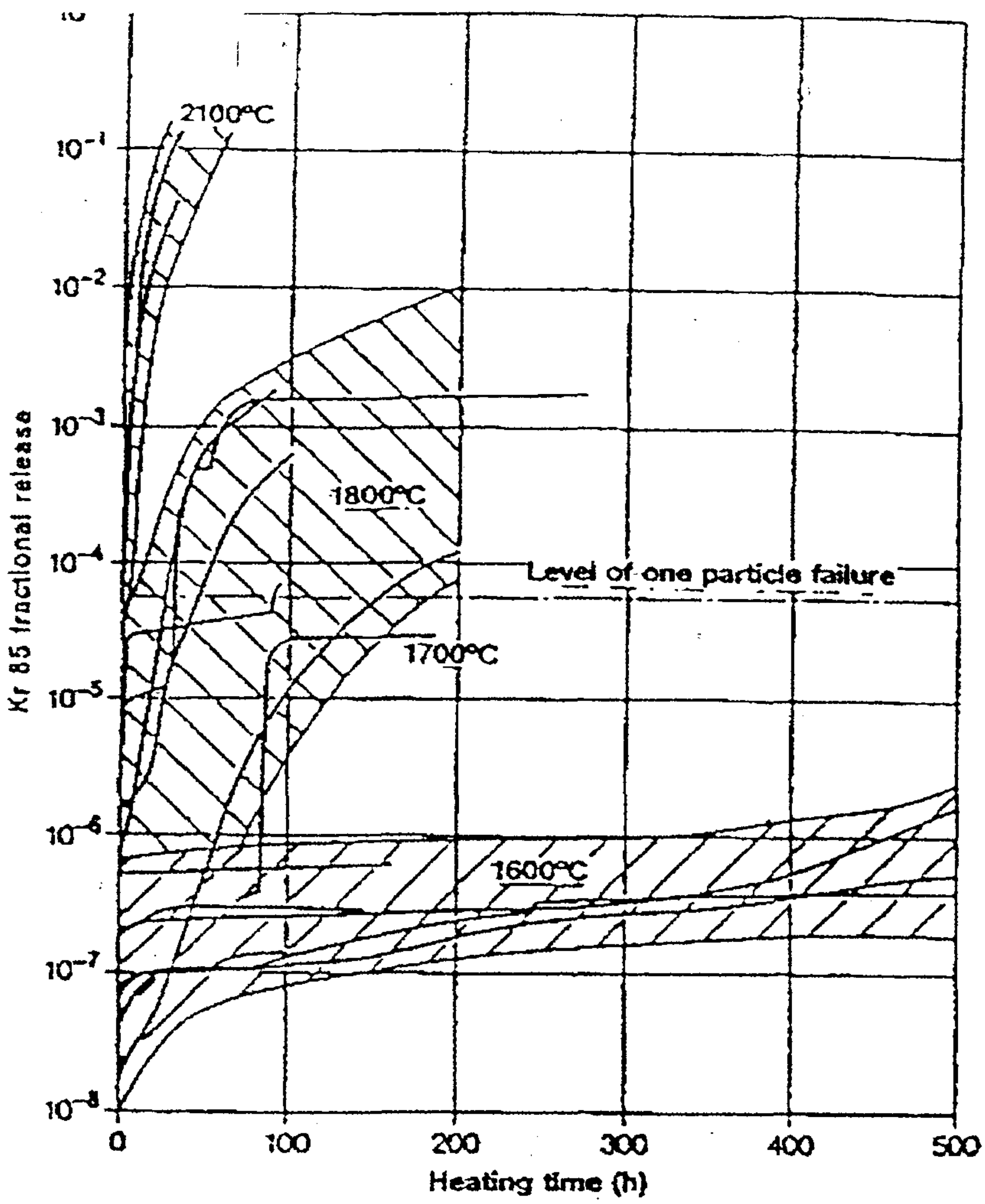


Heating furnace used in accident simulation tests
with irradiated HTR fuel elements

KFA

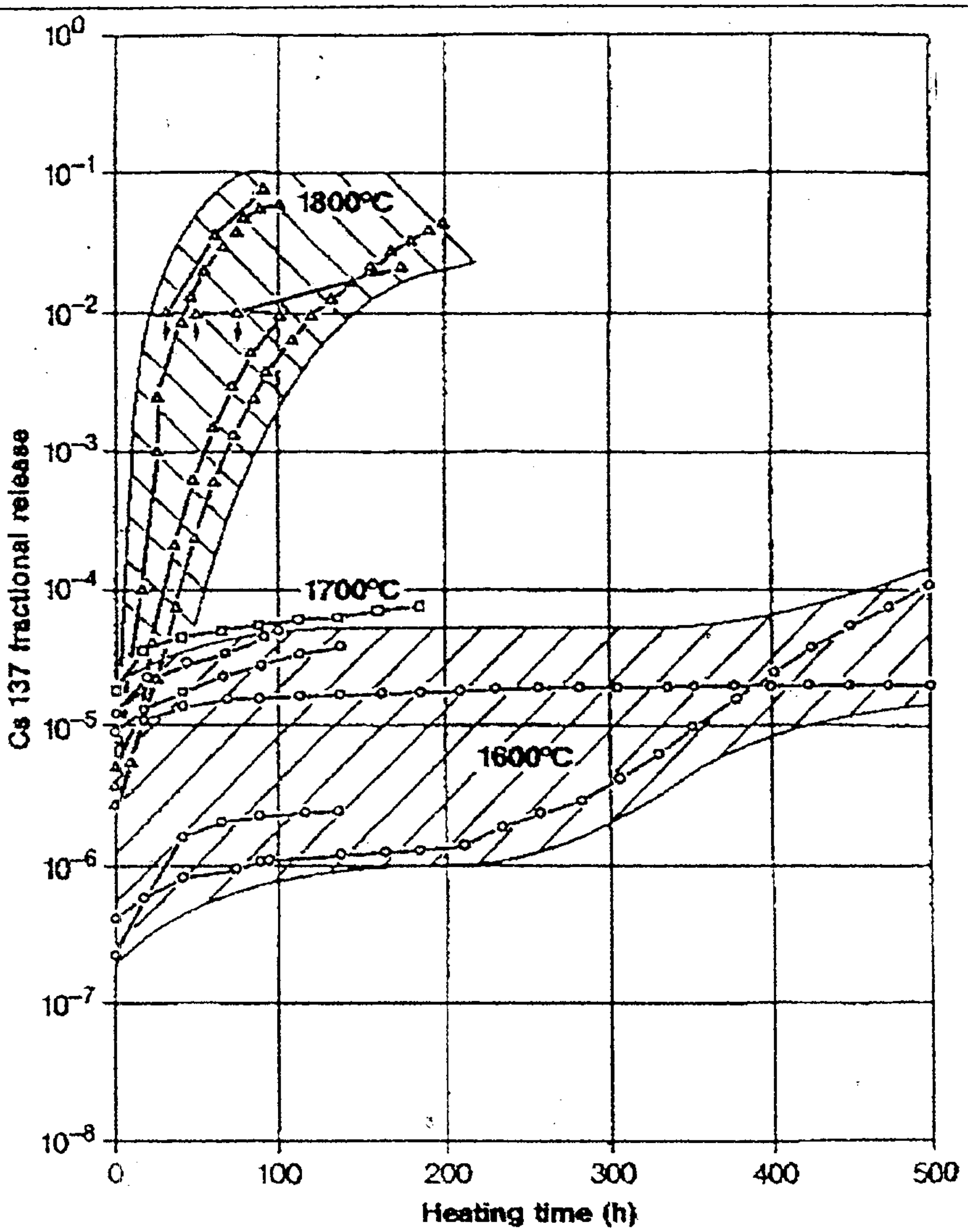


Spaltproduktfreisetzung aus einem BE mit UO_2-TRISO Partikeln (HFR-K3/1) bei 1600 °C



Krypton release during tests with irradiated spherical fuel elements at 1600 to 2100°C.

Accident condition performance of German fuel



Caesium release from heated spheres as a function of heating times up to 500 hours.

Accident condition performance of German fuel

Results of accident simulation tests with irradiated fuel elements containing UO_2 Triso

Fuel Element	Burnup %FIMA	Fast Fluence $10^{25}m^{-2}$ E>0.1 MeV	Heating test		Number of failed particles **		Fractional release				
			Temp (°C)	Time (h)	manuf.	heating	^{85}Kr	^{90}Sr	^{110m}Ag	^{134}Cs	^{137}Cs
AVR 71/22	3.5	0.9	1600	500	no	no	4.0E-7	5.3E-6	9.0E-4	6.9E-5	2.0E-5
HFR-K3/1	7.7	3.9	1600	500	no	no	1.8E-6	1.8E-7	2.7E-2	1.3E-4	1.1E-4
FRJ2-K13/2	8.0	0.1	1600	138 (160)	no	no	6.4E-7	3.3E-7	2.8E-3	1.0E-4	3.9E-5
AVR 82/20	8.6	2.4	1600	100	no	no	1.5E-7	3.8E-6	4.4E-3	1.2E-4	6.2E-5
AVR 82/9	8.9	2.5	1600	500	no	no	5.3E-7	8.3E-5	1.9E-2	5.9E-4	7.6E-4
AVR 89/13	9.1	2.6	1620 *	~10	no	no	2.0E-7	***	8.3E-4	1.3E-5	1.1E-5
			1620 *	~10		no	1.3E-9	***	1.5E-2	1.6E-6	1.4E-6
AVR 85/18	9.2	2.6	1620 *	~10	no	no	1.4E-7	***	6.5E-3	1.0E-5	1.3E-5
AVR 90/5	9.2	2.7	1620 *	~10	no	no	1.9E-7	***	1.1E-3	7.7E-6	9.0E-6
			1620 *	~10		no	6.6E-9	***	9.0E-4	3.5E-6	3.3E-6
AVR 90/2	9.3	2.7	1620 *	~10	1	2	1.0E-4	***	3.7E-2	5.0E-5	4.6E-5
AVR 90/20	9.8	2.9	1620 *	~10	2	3	2.4E-4	***	7.6E-2	5.6E-6	6.5E-6
AVR 91/31	9.0	2.6	1700 *	~10	2	18	1.2E-3	***	6.2E-1	3.7E-3	2.4E-3
AVR 74/11	6.2	1.6	1700	184.5	1	no	3.0E-5	7.2E-6	4.8E-2	8.4E-5	7.6E-5
FRJ2-K13/4	7.6	0.1	1600	138	no	no	3.0E-7	2.0E-8	4.5E-4	5.7E-6	2.5E-6
			1800	100		2	7.2E-5	1.4E-3	5.3E-1	9.7E-3	9.9E-3
AVR 88/33	8.5	2.3	1600	50	no	no	1.0E-7	8.4E-6	1.2E-3	1.1E-4	1.2E-4
			1800	20		~4	1.8E-4	2.3E-4	2.1E-1	4.4E-4	4.6E-4
AVR 88/15	8.7	2.4	1600	50		no	6.3E-8	***	9.1E-3	8.8E-6	1.2E-5
			1800	50	1	~6	2.9E-4	1.1E-2	8.1E-1	1.3E-2	1.4E-2
AVR 76/18	7.1	1.9	1800	200	no	~3	1.2E-4	6.6E-2	6.2E-1	5.3E-2	4.5E-2
AVR 88/41	7.6	2.0	1800	24	no	no	2.4E-7	1.2E-4	7.7E-2	1.4E-4	1.5E-4
HFR-K3/3	10.2	6.0	1800	100	no	~12	6.5E-4	1.5E-3	6.7E-1	6.4E-2	5.9E-2

* simulating calculated core heatup curve ** out of 16400 particles *** not measured

HTR-TN International HTR Fuel Seminar,
Brussels, Belgium, February 1-2, 2001

Long Time Experience with the Development of HTR Fuel Elements in Germany

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Research Centre Jülich[1], Technical University Aachen[2],
Advanced Nuclear Fuels GmbH Duisburg[3], Germany

Abstract
The development of spherical fuel elements for HTR-designs in Germany are discussed. Special attention is given to the development, production and characterisation (incl. kernel and coatings) as well as to the irradiation and post irradiation examination of the different coated particle systems. It has been demonstrated in various irradiation tests which were supplemented by heating tests that for a modular HTR power plant (with a thermal output of 200 MJ/s) during the specified normal operation as well as in the case of incidents and even accidents, where the maximum fuel temperature will be below 1620°C, the fission product release is very low.
In this context, it must be mentioned that the present coated particle design has not yet been optimised for the combination of high burn-up and high temperature resistance under accident conditions. The TRISO fuel available is a result from fuel development for large HTRs with gas turbines in a time when the modular concept was not yet been invented although its capabilities inspired the design of modular reactors. Thus, there is still a huge potential for improvement of coated particles especially when plutonium or actinide burning is also taken into account.

1) Introduction

The HTR utilises an all-ceramic core, a graphite core structure, ceramic-coated particle fuels and complete ceramic fuel elements. The use of refractory core materials combined with a single phase inert helium coolant allows high coolant temperatures and results in a number of significant advantages including high thermal efficiency of the HTR and its inherent safety advantages resulting from the low-power density and large thermal capacity of the core, the absence of coolant phase changes, and the prompt negative temperature coefficient. These features ease reactor siting constraints by reducing both cooling water requirements and the consequences of postulated accidents.
The development of the HTR has proceeded in two directions: a) the pebble bed concept in the Federal Republic of Germany and Russia (now also in China and South Africa), and b) the prismatic core in the United States, the United Kingdom, Japan and, recently with the GT-MHR, also Russia. The fuel elements for the pebble bed system consist of 60 mm diameter spheres made up a fuel-free carbon outer zone and an inner-fuelled region with coated particles uniformly dispersed in a graphitic matrix. The prismatic fuel element consists of a machined hexagonal graphite block ~750 mm long and 350 mm across flats. Alternate fuel and coolant holes are drilled in a hexagonal array. Fuel rods, consisting of coated particles bonded in a close-packed array by a carbonaceous matrix, are stacked in the fuel holes.

accidents remain below 1600°C without active control mechanisms. This modular design are intended to replace water-cooled reactors for electricity generation and to provide environment-friendly process heat for application such as heavy oil recovery, coal gasification and liquefaction, etc.
The German Reactor Safety Commission made in their recommendation in January 1990 e.g. the following statements to the HTR Modular Power Plant Concept developed by Siemens/Interatom. This system is characterised by the fact that several standardised nuclear heat production units of 200 MW_{th} output are combined to form a power plant. The limitation of the reactor power to 200 MW_{th} and of the mean power density to 3 MW_{th}/m^3 in connection with the core geometry has particular the following advantages: In the case of a failure of the main heat sink in the HTR Modul, residual heat removal is effected via passive heat conduction, heat radiation and natural convection to the surface coolers provided on the outside of the reactor barrel. Residual heat removal does not require any forced circulation inside the primary system. A maximum fuel temperature of 1620 °C is not exceeded, irrespective of whether residual heat removal remains intact per design intent during an incident or there is an additional failure of the residual heat removal via the surface coolers. Adherence to this maximum fuel element temperature is inherent safety feature of this reactor concept [1].

The HTR development is still on the way in different countries as it will be shown in the following:

In Japan has developed the experimental reactor HTTR with a thermal power of 30 MW which became critical in November 1998 and is on its way to full power. The major specifications of the HTTR are: Prismatic Block Core; Low enriched UO_2; TRISO; He pressure: 4 MPa; He inlet/outlet temperature: 395/850 and 950°C; Steel containment; Heat removal IHX and PWC (parallel loaded).

China has built the test reactor HTR-10. The HTR-10 with a thermal power of 10 MW represents the features of modular HTR design, it became critical in the end of the year 2000. The HTR-10 main design parameters are: Modular HTR with a Pebble Bed Core; Low enriched fuel with 17 % U-235 UO_2; TRISO; He pressure: 3 MPa; He inlet/outlet temperature: 250/300 and 700/900°C. Reactor core and steam generator are housed in two steel pressure vessel which are side-by-side with a connecting vessel between.

In South Africa, ESKOM as the national utility sees a nuclear future in the HTR pebble bed system. ESKOM successful operates the two unit Koeberg PWR station, but it does not see LWRs as a solution for the present. Rather, it is putting its technical and financial resources behind a HTR project which sees as the best approach to take. The concept design is concentrated on a 100 MW_{el} Pebble Bed Modular Reactor (PBMR) with a direct cycle gas turbine.

2) Coated Fuel Particles

Coated particles are in themselves miniature fuel elements on the order of a millimetre in diameter. A commercial reactor core contains between 10^9 and 10^{10} individual fuel particles. The coatings provide the primary barrier to fission product release. The very small size of coated particles is an advantage in testing, since statistically significant numbers of "fuel elements" can be tested. Individual tests typically contain 10^3 to 10^5 coated particles. As it will be shown through properly designed fuel devel-

2.1.1) The uranium-thorium kernel fabrication was mainly based on the sol-gel process. This process has been developed for the production of the (Th,U)O_2 kernel and the ThO_2 kernel as well as for the uranium-dioxide kernel fabrication. Using this specified process a total of approximately 400 kg UO_2 kernels was successfully produced for the manufacture of all fuel elements scheduled for AVR reloads [3,4].

2.1.2) For coating of the microspheres the fluidized bed technique is used. The pyrocarbon (PyC) is a unique material that has been central to coated particle development from the earliest days. Results showed that the structure and irradiation behaviour of PyC coatings are highly dependent on deposition conditions, which in turn determine coating properties such as density and crystalline anisotropy. Many activities has been done in this area emphasising the optimisation processes and the development of improved characterisation techniques, also for the post-irradiation examination of the coated fuel particles and fuel elements [2].
The UO_2 or the other kernels are batch-wise coated in fluidized bed furnaces. An inert gas, usually argon, is used for fluidisation. First a porous buffer layer of C_2H_2 is deposited. This layer supplies a free volume for kernel swelling and fission gas production during burn-up and protects the following highly dense layers from recoil atoms. Next a high density inner pyrocarbon layer is deposited from a mixture of C_2H_2 and C_3H_6. The layer SiC deposited from CH_3SiCl_3, predominantly for retaining the solid fission products, is brittle and therefore protected finally by an outer highly dense pyrocarbon layer [3,4].

2.1.3) The production of spherical fuel elements for HTRs consists behind the (i) fuel kernel casting and the (ii) coating of microspheres of the following steps: (iii) overcoating of particles; (iv) matrix powder preparation; (v) fuel element fabrication, i.e. pre-moulding of fuel zone, high-pressure isostatic pressing of complete element, machining, and 800/1950°C heat treatment; and (vi) quality control [3,4].
Table 1 shows the main particle and fuel data with German reference HEU and LEU particles.

3) Irradiation Testing of Coated Particles, Graphitic Matrix and Spherical Fuel Elements

The overall objective of the HTR fuel element development program was to qualify an element which minimises fission product release under normal and transient conditions for all types of HTR application as well as under accident conditions for small HTRs with a pebble bed core. Apart from fuel elements, the coated particles, the graphite matrix and the reflector graphite have been tested in several MTRs like HFR-Petten, R2-Studsvik, BR2-Mol, Siloé-Grenoble, FRJ2-Jülich as well as in the HTR test reactors AVR and Dragon. Long time tests have been carried out over more than 20 years with about twelve different spherical fuel element types in the AVR as a large-scale test bed [7].
A typical irradiation program for testing was directed to (i) "determination of particle failure rate" under conditions exceeding the demands of the HTR projects with reference to fast fluence and burn-up to investigate performance margins at irradiation temperatures (800-1200°C); (ii) "investigation of burn-up influence", irradiation of fuel in thermal test reactors with low fast neutron fluxes to separate burn-up controlled effects from neutron-induced effects (800-1300°C); (iii) "reference tests", demonstration of reference fuel elements under condition enveloping the demands on of differ-

3.3) The irradiation testing of spherical fuel elements in accelerated and long-time tests was carried out as described above. During irradiation of spherical HTR fuel elements, important information about fuel performance could be obtained from the tests in the MTRs as well as from the AVR operation. But all essential properties of the fuel elements have been controlled mainly in the post-irradiation examinations in the Hot Cells e.g. (i) dimensional change of fuel elements; (ii) mechanical (crushing) strength; (iii) corrosion resistance; (iv) fission product behaviour, (v) accident simulation heating tests.
The source terms for fission products in the primary circuit of an HTR are:
(i) heavy-metal contamination;
(ii) particle manufacturing defects and in-pile particle failure;
(iii) release from intact particles.
The important isotopes are Cs, Sr, Ag, I, Xe and Kr. From all the available data it can be pointed out that fission product release from the spherical HTR fuel elements during normal operation is insignificant. In the German HTR-Modul, for instance, the calculated release of Cs-137 accumulated during normal operation conditions after 32 operating years is approximately 2.6×10^{12} Bq (70 Ci). The coolant gas activity of the AVR after 20 years operating was approximately 5.5×10^{11} to 1.1×10^{12} Bq (20-30 Ci) of the noble gases, a few tenths of a MBq of aerosols and non-measurable amounts of iodine.
In the case of a core heatup accident, higher temperatures lead to enhanced diffusion of fission products out of the particle kernel through the TRISO coating and through the graphite [8,9]. One of the most important diffusion coefficients is that of caesium in SiC. On the basis of a number of heating experiments, it has been shown that an increased permeability of the SiC layer for caesium at temperatures of more than 1600°C. Fig. 6 shows results from fission product experiments in the form of diffusion coefficients in UO_2, in pyrocarbon and silicon carbide coating layers, and A3 matrix as a function of temperature.
Summarising it can be pointed out that for the German reference LEU-TRISO fuel elements the release of solid fission products, e.g. Cs-137 from coated particles into the fuel element matrix and from there into the reactor core equals to the low release levels of gaseous fission products [3,7].

4) Heating Tests for Accident Condition Performance
Accident simulation tests have been performed since the mid-seventies whereby the Research Centre Jülich has concentrated on heating complete spherical fuel elements rather than single particles or small numbers of coated particles. An early experimental program had consisted of heat-up ramp tests with $(U,Th)O_2$ BISO fuel up to 2500°C. This program was followed by work with fuel elements containing $(U,Th)O_2$ TRISO and UO_2 TRISO particles. Special attention has been given to accident performance testing of the UO_2 TRISO particles for small HTRs [10, 11].
The fission gas release data from spheres during heating tests are shown in Fig. 7. The measured isotope is Kr-85 which give the same release as Xe-133 and I-131. As expected, release increases with heating temperature and duration. All 1600°C release results remain below the level of one particle failure (6×10^{-5} fraction for 16400 particles). The shape of the release curves can be explained by the following two phenomena: (i) Deterioration of the SiC layer leads to permeability to fission products, but the remaining intact outer pyrocarbon layer delays the release of noble gases and iodine; (ii) On rare occasions can a burst of gas release be observed which is due to pressure induced complete coating failure.

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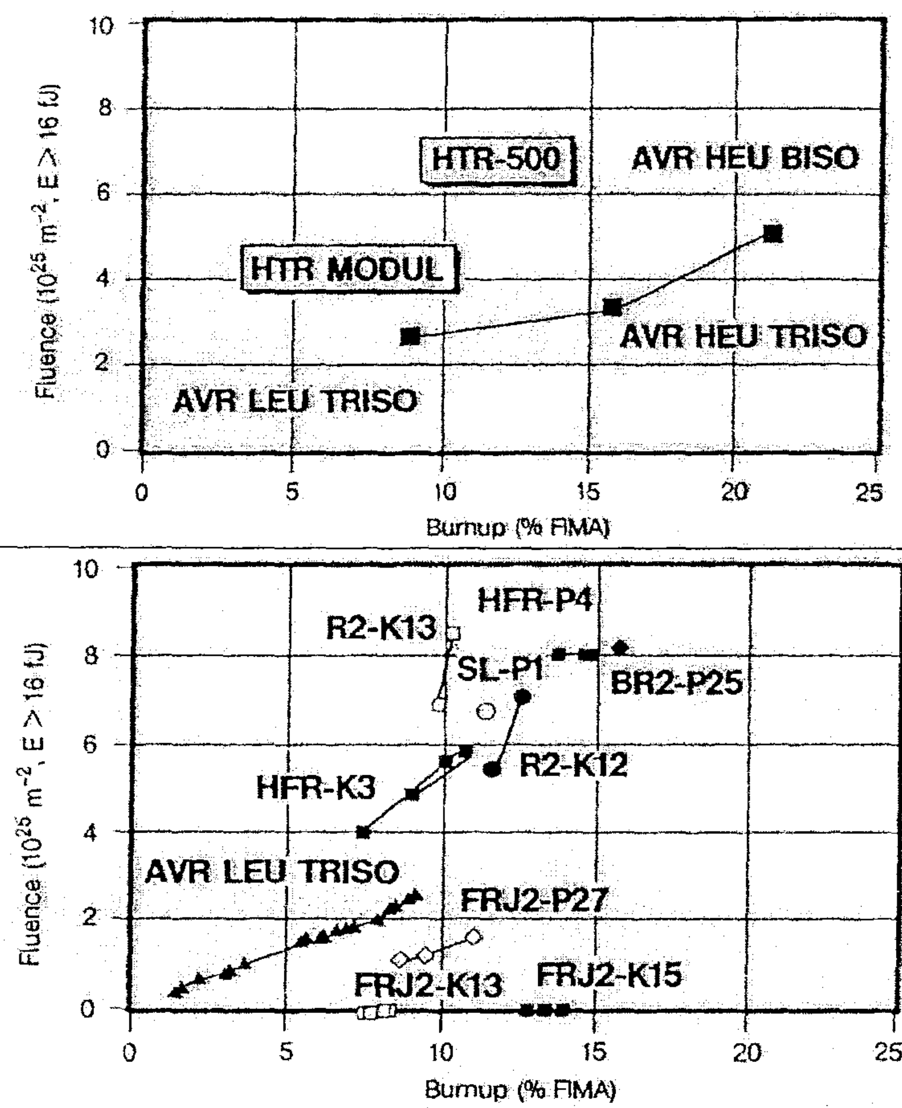

Fig. 3) The main parameters in fuel irradiation testing are heavy metal burn-up and accumulated fluence of fast neutrons. Irradiation temperature is usually around 1000°C in the tests shown here. AVR testing of LEU fuel covers MODUL conditions (top diagram), MTR irradiation tests (bottom diagram) cover a wide field of potential applications

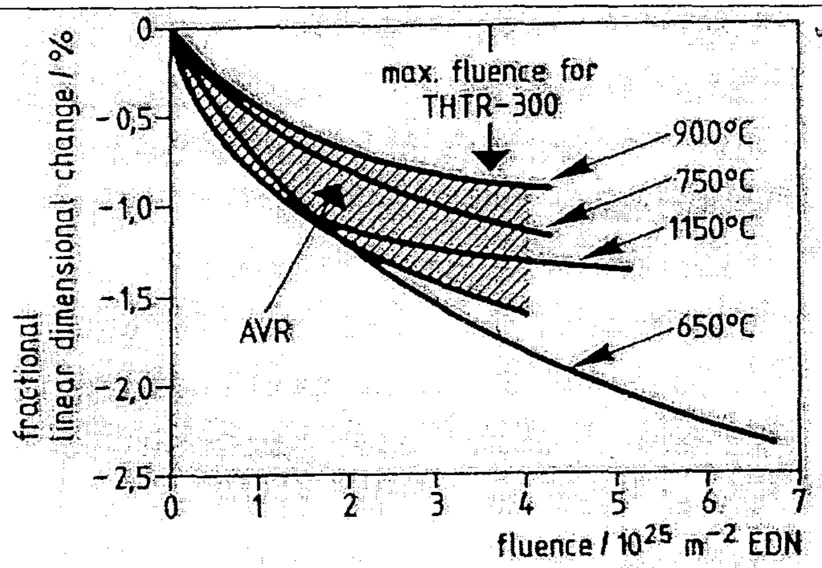

Fig. 5) Dimensional change of A3-3 matrix samples and AVR fuel elements during irradiation as a function of the fluence.

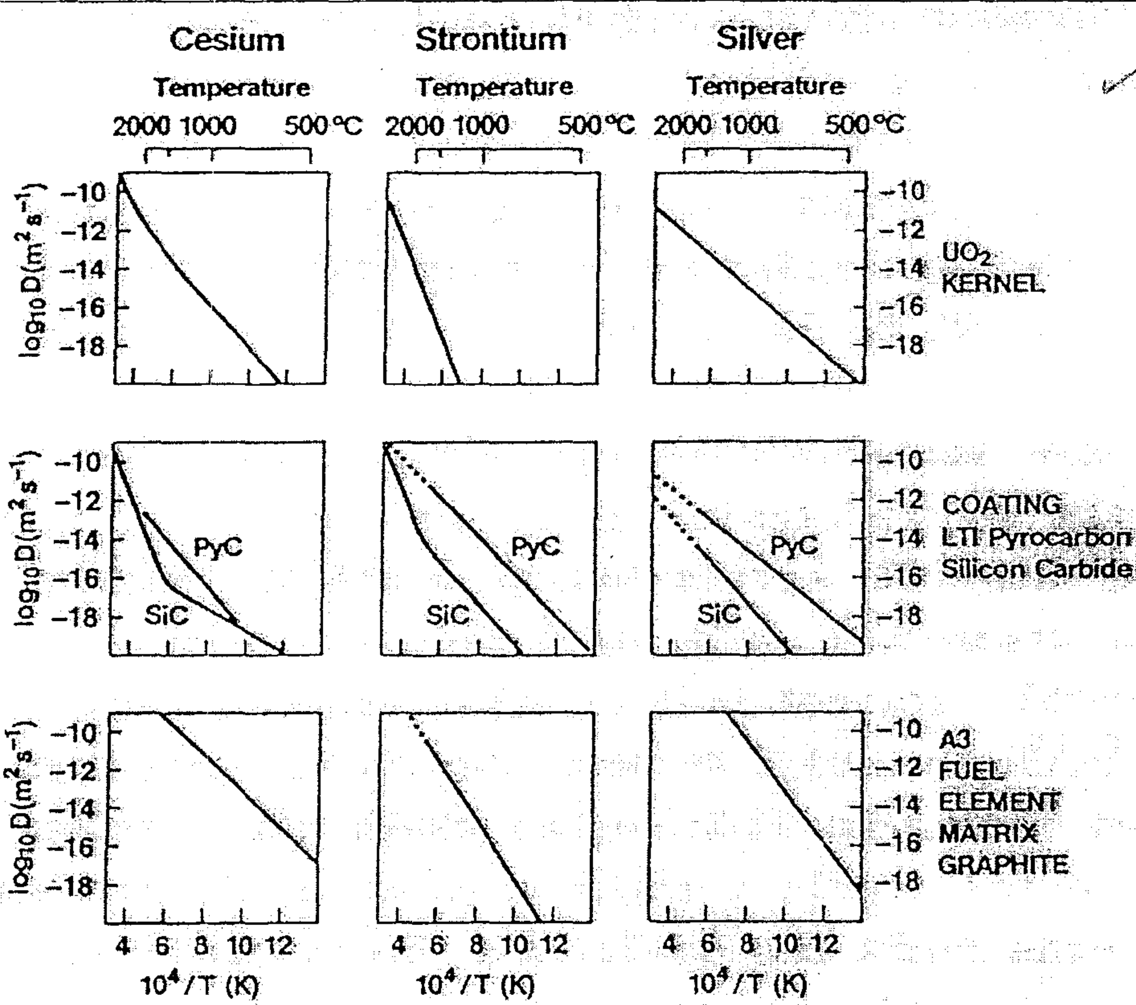

Fig. 6) Diffusion coefficients of key fission products (Cs-137, Sr-90 and Ag-110m) in components of HTR fuel elements.

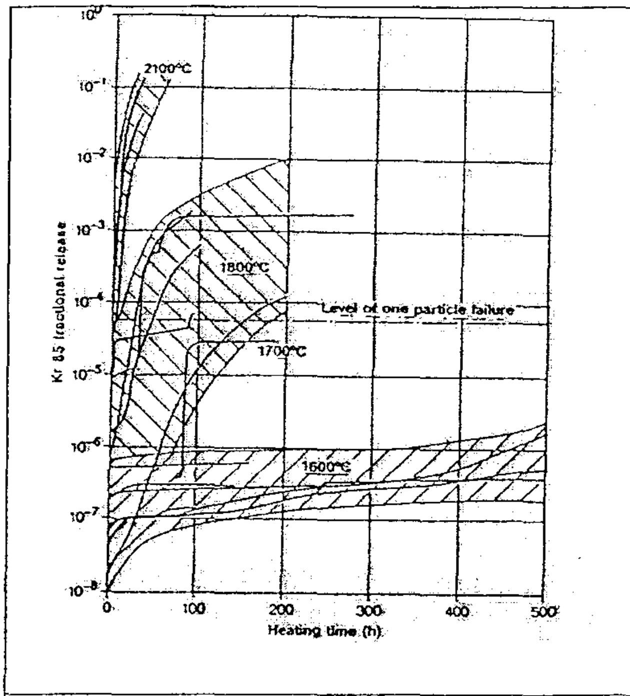

Fig. 7) Krypton release during tests with irradiated spherical fuel elements at 1600 to 2100°C.

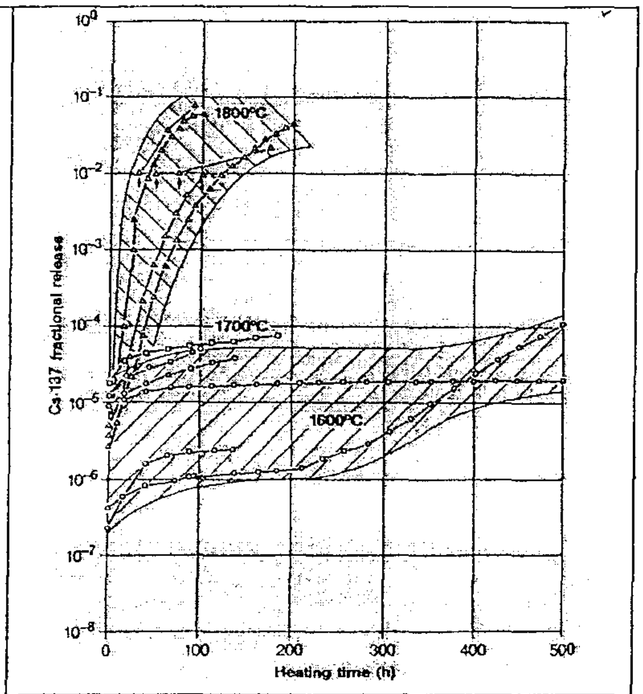

Fig 8) Caesium release from heated spheres as a function of heating times up to 500 hours.

Table 1: Design Data for $(Th,U)O_2$ TRISO and UO_2 TRISO Fuel Elements

Design Parameter		HEU	LEU
Coated Particles			
Kernel Composition		$(Th,U)O_2$	UO_2
Kernel Diameter	µm	500	500
Coating Layer Thickness	µm	95/40/35/35	95/40/35/35
Coating Layer Sequence		Buffer/PyC/SiC/PyC	Buffer/PyC/SiC/PyC
Fuel Element			
Heavy Metal Loading	g/Element	11	8–12
U 235 Enrichment		93 %	7–13 %
No. Particles per Element		19,000	13,000–20,000
Volume Loading of Particles		13 %	10–15 %
Operating Requirements			
Mean Operating Time	d	1100–1500	700
Max. Burnup	MWd/t_{HM}	120,000	90,000
Max. Fast Dose (E > 0.1 MeV)	10^{25} m^{-2}	4.5	3.3
Max. Fuel Temperature	°C	1020	1030
Max. Power/Element	kW	2.7	4.1

CERAMIC COATINGS FOR HTR GRAPHITIC STRUCTURES - TESTS AND EXPERIMENTS WITH SIC-COATED GRAPHITIC SPECIMENS

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Abstract

Graphite materials are used in High-Temperature Reactors for fuel elements and core structures. In the AVR and in the THTR it was successfully demonstrated that especially the spherical fuel elements showed excellent behaviour during normal operation and accident conditions. Improvements are possible as part of efforts to achieve catastrophe-free nuclear technology. In case of a massive ingress of air or steam into the primary circuit of an HTR, it is possible, if no active steps are taken, that serious corrosion of graphitic structures can happen.

For corrosion protection it is appropriate to provide these structures with ceramic (SiC) coatings. These coatings were produced by chemical vapour deposition and slip coating method. The coated graphitic specimens, spheres (without nuclear material) and other samples, were tested in many experiments, such as corrosion, mechanical and irradiation tests. The results of these tests show that SiC coatings applied to many graphites are corrosion-resistant and mechanically safe. The post-irradiation experiments showed for some coated graphitic spheres good corrosion properties at temperatures in the region of 750°C. For one material the corrosion resistance was even good for temperatures up to 1400°C(1600°C).

Furthermore, alternative forms of coated spheres, consisting of screwed half-shells, have already been tested successfully in corrosion and irradiation experiments.

1.Introduction

In HTR plants graphite materials are used for fuel elements and reflector structures. During the operation of HTR plants (AVR,THTR) these graphitic structures were extensively tested. Operating experience was very positive. It was proved that all operating demands (like high temperatures, neutron doses, mechanical loading) were fulfilled by the graphitic structures. It should be emphasized that the primary circuit of HTRs is very clean due to the good retention properties of the fuel elements, especially of the coated particles. Extensive tests proved that these good retention properties for fission products are valid even for temperatures up to 1600°C for several hundred hours. The development and testing of the spherical fuel elements can therefore be regarded as a complete success.

Although the above-mentioned properties of the graphitic structures are very good, research into improvements should continue. One of the HTR-specific accidents (analysed extensively in safety studies) is the entry of foreign media like air and water into the primary circuit. In design-basis accidents the corrosion of fuel elements is not critical for reactor safety. In certain cases of hypothetical accidents like massive air ingress it is possible that the matrix graphite of the fuel elements will be destroyed by corrosion in such a manner that a release of fission products may be possible. Up to now a large number of solutions have been considered to prevent such possibilities. For future nuclear plants new techniques have to be developed which will lead to a transparency of the technology applied. Therefore it is suggested that the outer surfaces of the graphite fuel elements should be provided with ceramic coatings.

The application of a ceramic corrosion protection coating is a very attractive solution not only for fuel elements but for graphitic structures like the bottom and side reflector too.

2.Graphite oxidation

Analyses in the field of graphite oxidation have been made in a large number of experimental tests and theoretical studies. To give an impression of how graphite spheres will be corroded by air, the results of experimental work in different facilities of the University of Duisburg /Epp 90/ , /Roe 94/ are presented here.

In these test plants uncoated graphite spheres (without coated particles) of the original size (6cm diameter) were placed in silos and streamed by air from bottom to top. These tests were made in a temperature region from 600°C to 1200°C and for 3 different velocities. In Fig.1 the measured corrosion rates of these uncoated graphite spheres are presented. To give an impression of how large the corrosion protection by SiC coating on graphite spheres can be, the corrosion rate of a coated sphere is marked by the lower line in Fig.1.

3.Graphite materials

Different graphite qualities were coated with silicon carbide. To obtain a wide spectrum of information, nuclear and commercial graphites were used.

The nuclear graphite A3-3 is used as the matrix material for fuel elements in high-temperature reactors. This material is a composition of 64 wt% natural graphite, 16wt% petroleum coke graphite and 20 wt% resin binder. The fuel elements with this material are processed and heat-treated at temperatures below 2000°C. Therefore this matrix material is the only one which is not fully graphitized. /Jül 82/

The nuclear graphite material IG 110 is used for fuel element blocks in the Japanese HTTR plant. This IG 110 is a fine-grained isotropic graphite, which is fabricated from coke filler with binder on the basis of coal tar pitch. This graphite is isostatically pressed and fully graphitized (at 2800°C). /Jae 91/

The nuclear graphite ASR-1RS was used for reflector blocks in HTR plants. This material is a pitch coke graphite manufactured according to secondary coke technology; three impregnations with coal tar pitch; vibrationally moulded.

The nuclear graphite V 483 was employed for core-supporting columns in HTR plants. This graphite is a pitch coke graphite; fine grain, high binder content, isostatically moulded./Jül 87/

The commercial graphites are: FU 9512 (FP 379) from Schunk, EK98 and EK 432 from Ringsdorf and other graphites. These graphites are electrographites with filler material on the basis of pitch coke and binder on the basis of coal tar pitch./Sch_1 89/, /Rin 91/

4.Production methods for ceramic coatings on graphite surfaces

The SiC coatings on graphite specimens (mainly spheres) were produced by two different methods: Chemical vapour deposition (CVD) and slip coating.

4.1.Coating methods based on CVD

Chemical vapour deposition methods are well known in chemistry and chemical engineering, so only a brief description will be provided here.

Chemical vapour deposition produces coatings from the gas phase. For SiC coatings, the SiC is deposited from the gas mixture CH_3SiCl_3/H_2 in a temperature range of 1150-1450°C at different pressure ranges. To obtain a continuous transition of the substrate and the coating material, an appropriate ratio of coating gas (CH_3SiCl_3) and reductant (H_2) is necessary as well as a suitable reaction temperature (approx. 1300°C).

Schunk /Sch_1 89/ mainly supplied coated graphitic spheres of FU 9512 and IG-110 quality. In the present case a system with temperatures in the range of 1200°C is used. The SiC layers produced show a coating-thickness of 50-150 µm.

In addition, Schunk /Sch_2 89/ provided coated spheres of A3-3 quality. The layers are produced by the following steps: 1) direct siliconizing of the graphitic spheres in a silicon steam atmosphere at temperatures below 1600°C. 2) immersion of the spheres in a SiC melt at temperatures up to 1500°C.

A great deal of interest has been shown also in CVD-SiC in functional gradient materials, whose composition changes continuously from SiC to C. In this case dense CVD-SiC with a thickness of 200 µm was formed on the graphite substrate and a CVD-SiC/C phase with a thickness of 1-2 mm containing voids was then formed on top of the first film. In this region, the ratio of SiC to C changes continuously and by adjusting the orientation of SiC it is possible to control the number of voids. Finally, a CVD-C film of a thickness of 100-200 µm is formed on top of the other films.

4.2. The slip-coating process

A new method for coating graphitic materials by SiC was developed at the Institute of Reactor Safety and Technology of the Technical University of Aachen (RWTH). This process consists in coating graphitic substrates by a slip and after that in a Si infiltration by a high-temperature process.

In a first processing step the uncoated substrate is wetted with a ceramic suspension consisting of α-SiC of different grain sizes, ultra-fine-grained graphite powder, organic binder and a solvent. After complete wetting of the substrate with this SiC slip the solvent evaporates out of the slip during a drying phase. A solid and a porous layer arise by getting out the solvent. The shares of binder polymerize and form a strong fixation between the solid phase of the slip and the surface of the substrate.

In a next step this specimen is put into a Si-contribution paste inside a furnace at a temperature of 1700°C.The melted Si (~1410°C) infiltrates the layer and the substrate of the specimen. During infiltration, the graphite of the outer surface and in the open porosity area at the inner surface, reacts with silicon to form β-SiC. At the same time , the silicon component reacts with the carbon components of the slip coat of β-SiC until the reaction is terminated. This new formed β-SiC grows on the α-SiC grains. The coating thus formed is a good connecting scaffold with the surface of the substrate. The layer penetrates into the graphite into a depth of nearly 1 mm. The open pores near the surface are filled with β-SiC. By this a strong adhesion is obtained.

The SiSiC coatings are produced with thicknesses of 75 to 150μm . Successful coating was performed for nuclear graphites IG 110, V 483 and ASR 1 RS, as well as for electrographites IG 430, EK 98 and EK 432 by optimizing coating and siliconizing process performance.

5.Experimental results with SiC-coated graphites

To characterize the SiC-coatings produced, corrosion and thermal shock tests as well as ceramographic analyses are carried out. Some of these results have already been presented in former reports /Nea 97/, /Alk 98/. The new results of corrosion and post-irradiation experiments are additionally reported here.

The standard tests to check the corrosion resistance of the coated spheres were mainly performed at a temperature of T=750°C for an experimental period of 24 hours in a natural convection air stream. The aim of the development is to reach a corrosion rate of R<0.1[mg/cm^{2}h] in the air stream for 200 hours in the temperature range from 400-1600°C. Corrosion tests in steam were performed in a temperature range from 600°C-1000°C.Additionally heated specimen were also shock-tested by dropping them into cold water. The analyses of these tests give information on the resistance to sudden changes of temperature of C/SiC structures. The following sections present experimental results for SiC coatings produced by different methods.

5.1.Corrosion results for coatings produced by CVD methods

5.1.1Results for SiC-coated full graphite spheres

SiC coatings on graphite spheres by the CVD method have been mainly produced by Schunk /Sch$_2$89/. We started with coating of spheres made of the graphite material FU 9512. This material has a thermal expansion factor which is similar to the SiC expansion factor. The results of the corrosion tests show very low corrosion rates (R≤0.01mg/cm^{2}h) for long test periods (≤200h).

Mechanical tests (dropped 10 times from a height of 2m on graphite spheres) were also performed on these FU 9512 coated spheres. No cracks or damage could be found. Corrosion experiments were then performed again on these spheres. Once again no damage could be found.

The aim of the development is to reach a corrosion rate of R<0.1[mg/cm^{2}h] in air stream for 200 hours in the temperature range from 400-1600°C. This must be achieved for nuclear graphite materials.

SiC-coated spheres made from A3-3 graphite material showed an inadequate corrosion rate. Therefore we looked for other nuclear graphites to coat.

A very good corrosion-resistant quality was achieved with the IG 110 material, a graphite material used in Japan for HTR nuclear graphite. Corrosion experiments were performed with coated spheres of 60 mm diameter. The results were very good (Tab.1). Even short heat-up times of 3 hours had no influence on the results.

5.1.2.Results for SiC-coated screwed (hollow)graphite spheres

The main aim of our work is to obtain SiC coatings on A3-3 graphite spheres in the quality achieved on IG-110 graphite spheres. Until now the corrosion rate of SiC coating on A3-3-graphite spheres has not been sufficient. Therefore we tested alternative forms of coated fuel spheres. Five hollow spheres (d_a=60mm, d_i>50mm) made of IG-110 were produced consisting of two hemispheres screwed together (Fig.2). A3-3 spheres (d<50mm) without coated particles are installed inside two hollow spheres. The other three hollow spheres are without inner spheres. These screwed spheres were coated on the outer surface (d_a=60mm) with SiC and then tested for corrosion resistance see Tab.2. The corrosion-resistant screwed spheres were subsequently irradiated in the HFR Petten.

5.2.Corrosion results for coatings produced by the slip coating process

Tests for the corrosion resistance of the SiSiC-coated graphite qualities were carried out in an air atmosphere with an air flow velocity of 0.2 m/s at temperatures between 700°C and 1400°C for 24 hours.

The slip-coated samples of different graphite qualities showed no measurable corrosion due to graphite burn-up. The corrosion examinations were carried out at a temperature of 750°C-800°C for times ranging from 24 hours to 200 hours. At this temperature the graphite corrosion process is due to pore diffusion of oxygen through possible defects in the protective coatings. None of the samples examined at this temperature and times exhibited any measurable mass decrease, so that outstanding protection of the graphite samples can be concluded.

The reaction rates for slip-coated samples in air are very low for all electrographites and nuclear graphites (without A3-3). For instance the reaction rates for SiC-coated IG110 by slip coating have values in the region of $R \leq 0.01 mg/cm^2h$. These results have already been presented in former reports /Alk 98/, /Nea 97/, /Mei 96/.

6.Irrradiation and post-irradiation tests

To examine the integrity of different coatings after neutron irradiation, SiC-coated graphite spheres ($d=60mm+\delta_{SiC}$) were irradiated in HFR Petten. The temperatures and the neutron fluences for these irradiation tests were chosen in such way that the average operating conditions of an HTR-Modul were covered. The average surface temperature of the coated spheres ranged between 540 and 680°C, the neutron fluence ranged between 1.4 and 1.95×10^{25} m^{-2} (E>0.1 MeV) Tab.3. Graphite spheres coated in different ways were supplied for this irradiation test. Three graphite materials were used for these spheres: IG 110, V 483 and A3-3. The SiC coatings were produced by the slip-coating process and CVD. Before irradiation, these coated spheres were tested for corrosion resistance for 50 hours at 750°C. In these corrosion experiments 5 coated spheres showed no mass loss. Only the coated A3-3 sphere had a small mass loss and was therefore not investigated further.

6.1.Irradiation rig D 247-01

Six coated spheres and one uncoated graphite sphere were loaded in an irradiation capsule in HFR Petten. This capsule consisted of a stainless containment housing the seven spheres in a graphite structure see Fig3. The sample holder was instrumented with twelve thermocouples, three fluence detector sets and two gamma-scan wires /Con 96/

The irradiation test was performed in the HFR Petten for a period of four cycles or 93.89 full-power days.

6.2.Post-irradiation inspections

The weights of the spheres before and after irradiation are given in Table 4.Only sphere No. 6 is not taken into consideration, because the coating showed insufficient adhesion. The weight loss of the uncoated sphere (No. 1) is reasonable, because a borehole for temperature measurement was drilled into the sphere. The visual inspection in the HOT CELLS in Jülich shows that sphere No. 2 has two small points with peel-off. Spheres No. 3 and No. 7 have one place which looks different from the surrounding surface. The other coated spheres have no changes on the surface /Con 96/, /Der 97/.

To answer the question whether the coating of the irradiated spheres is damaged or not, we performed corrosion experiments in the HOT CELLS at Jülich .

6.3.Post-irradiation corrosion experiments

The corrosion experiments with these irradiated spheres were performed in the KORA apparatus in the HOT CELLS at Jülich.

6.3.1.Description of the KORA apparatus

The KORA furnace /Sch 99/, in which specimens up to spherical fuel element size can be heated, is installed in the gas-tight box of a HOT CELL. The resistance-heated furnace contains two concentric tubes placed inside each other, which may be made of fused silica, alumina or SiC, depending on the test temperature. The air first flows into the annular gap between the inner and outer tube where it is heated before reaching the specimen through an opening at the end of the inner tube (Fig. 4).

6.3.2.Results of KORA experiments

The KORA apparatus was used to perform temperatures from reactor operation (750°C) to accident conditions (1600°C). The heat up followed 200°C/h. During the test an air pressure of about 110 kPa and a flow of 30 ltr/h was reached. In contrast to higher temperatures, there is during the 750°C step no formation of a complete protecting oxide layer hindering SiC corrosion. Five SiC-coated spheres and also one uncoated graphite sphere were tested in the KORA apparatus. The results of these corrosion tests are summarized in Tab.5. Two of the five SiC-coated spheres (No.4 and 5) had no damage before and after the first post-irradiation corrosion tests at 750°C. Even the defective SiC spheres– No.2 and 7- had a significantly smaller corrosion rate at 750°C than the uncoated graphite sphere. One of the two spheres with no visible defect (No.5) had no weight loss after all three corrosion tests in air up to 1400°C. During the 1600°C-test the inner SiC furnace tube melted. Therefore the sphere was damaged by this occurrence. There are no indications that this sphere (if not damaged by the tube) would not pass the 1600°C corrosion test successfully.

6.4.Irradiation rig D 247-02

The performance of the irradiation for the second project was successfully completed in 1998 /CON 99/. The irradiation targets of the second project consisted of five screwed SiC-coated graphite spheres (see 5.1.2 and Tab.2) and three SiC samples each 15 mm in diameter and 16 mm in length. Three of the five screwed spheres were hollow without an inner sphere. The other two spheres had an inner sphere of d<50mm. The samples were irradiated in the HFR Petten under typical HTR-Module conditions between 600 and 770°C up to fast neutron fluence of 1.92×10^{25} m^{-2} (E>0.1 MeV).The main operating data were similar to the first experiment (rig D 247-01). A neutron radiograph image, taken shortly after completion of the irradiation, and the visual inspection after recovery of the samples showed no damage to the SiC coating.

The irradiated screwed spheres and SiC samples are now stored in the HOT Cells at Jülich. First measurements showed that the differences between the weights before and after irradiation are very small /Pot 99/ Tab.6 .

These irradiated coated hollow spheres will also be tested in the KORA facility in the next few months.

7.New SiC-coated IG-110 spheres

In order to improve the irradiation behaviour of the IG-110 coated spheres the coating techniques were slightly modified for CVD and for the slip-coating process. New coated full IG-110 spheres were produced and corrosion-tested (see Tab.7). These corrosion tested spheres will be prepared for the next irradiation rig.

8.Summary

In HTR plants graphite materials are used for fuel elements and reflector structures. As part of the efforts to achieve catastrophe-free nuclear technology it is appropiate to provide these structures with SiC coatings. These coatings were produced by chemical vapour deposition and slip coating method. The coated graphitic specimens, spheres (without nuclear material) and other samples, were tested in many experiments, such as corrosion, mechanical and irradiation tests. The results of these tests show that SiC coatings applied to many graphite materials (as electrographites and nuclear graphite IG110) are corrosion-resistant and can withstand the required mechanical loads. The post-irradiation experiments showed for some coated graphitic spheres good corrosion properties at temperatures in the region of 750°C. For one material the corrosion resistance was even good for temperatures up to 1400°C(1600°C).

The main aim of our work is to obtain SiC coatings on A3-3 graphite spheres in the quality achieved on IG-110 graphite spheres. Up to now the corrosion rate of SiC coating on A3-3 graphite spheres has not been sufficient.

Therefore we tested alternative forms of coated fuel spheres. One modification of the present fuel element concept is such that the fuel-free graphite zone of the fuel sphere consists of two screwed half-shells of IG 110 graphite instead of A3-3. Several experiments have been carried out for the coating and joining of such parts. A strong joint of the shells and corrosion resistance of the two parts can also be ensured for this case. Irradiation-damage of the SiC coatings was not observed.

The coating of full A3-3 spheres will be continued with different coating methods.

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Figures

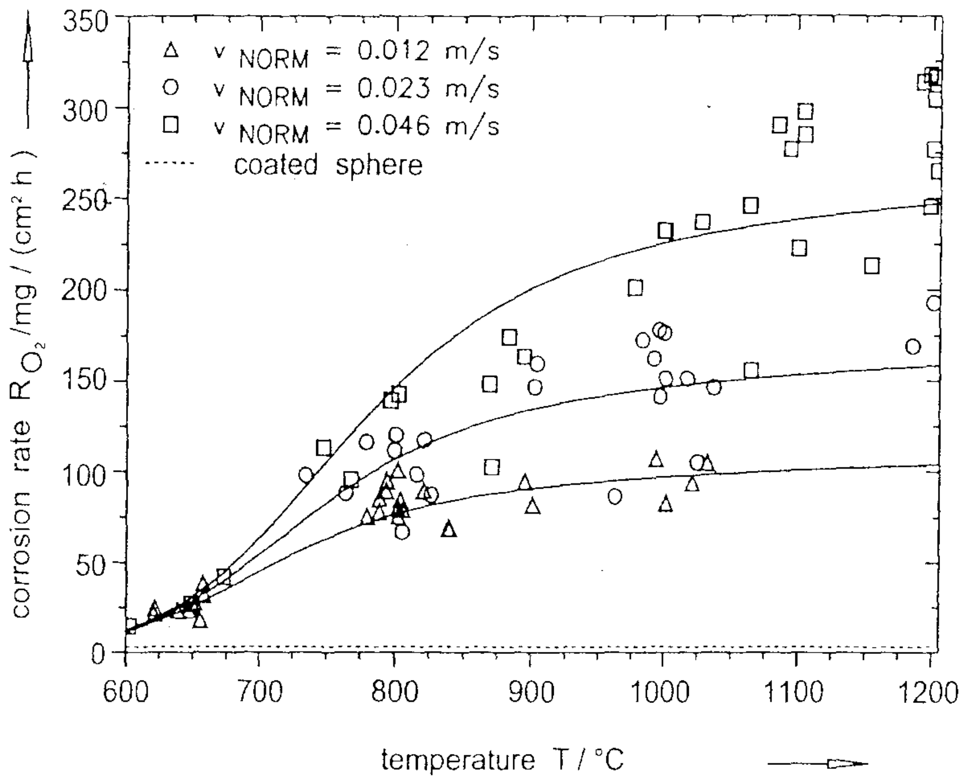

Fig.1: Corrosion rates of graphite spheres /Epp 90/, /Roe 94/

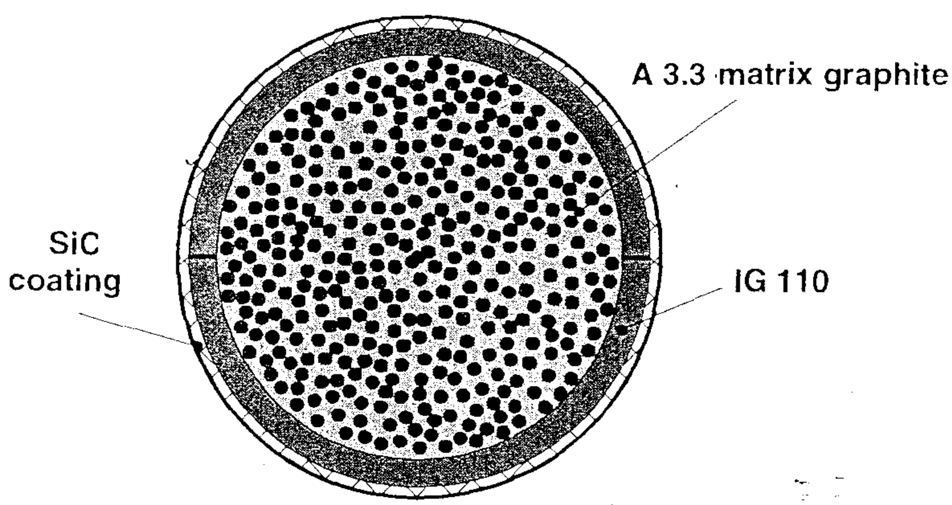

Fig.2: Proposal for a possible modification of the present HTR fuel element concept

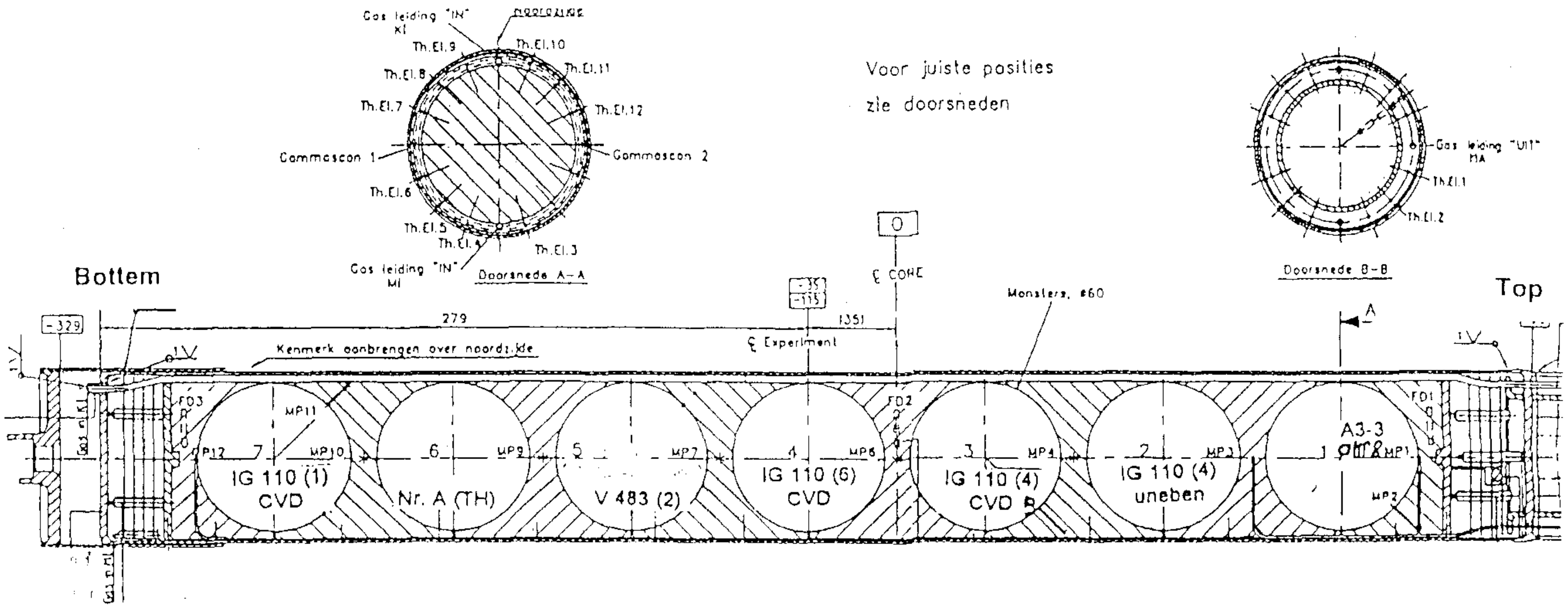

Fig.3: Schematic drawing of RIG D247-01 /Con,/

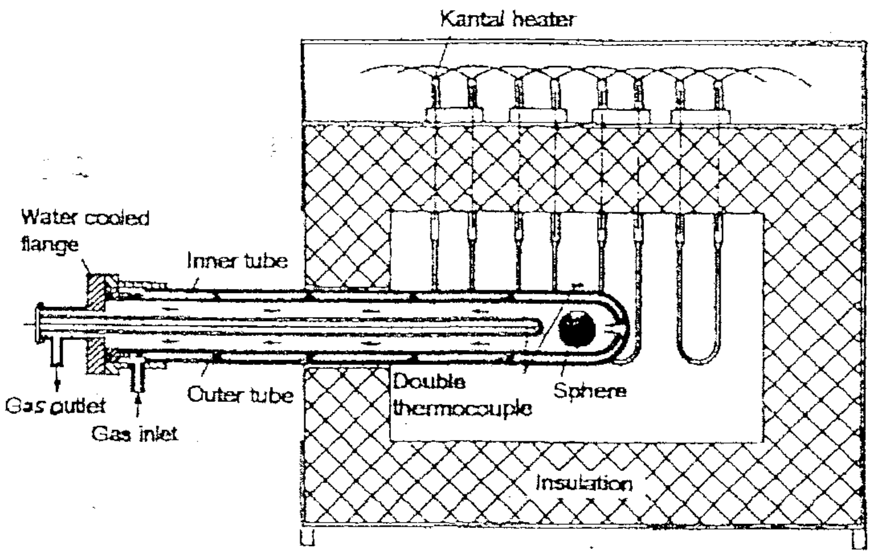

Fig.4: KORA resistance furnace

Tables

Tab.1: Corrosion tests of SiC-coated IG-110 graphite spheres(60mm) at 750°C

air streaming, 50 h total time			
graphite: IG 110		diameter: 6 cm surface: 113cm^2	
sphere no.	heat-up time h	weight change mg	corrosion rate mg/cm^{2}h
1	3	+1	~0
2	3	-261	0.046
3	3	-13	0.0023
4	3	+1	~0
5	3	+10	+0.002
6	3	-0,7	~0

Tab.2: IG-110 screwed spheres, CVD-coated with SiC (2 with inner sphere, 3 without inner sphere)

Corrosion tests at 750°C, 50h, air stream results before irradiation			
hollow sphere	mass [g] before corrosion test	mass [g] after corrosion test	diameter [mm]
HK M1	200.089	200.088	60.3-60.4
HK M2	198.714	198.707 *	60.3
HK O1	86.730	86.729	60.3-60.4
HK O2	86.582	86.584	60.3
HK O3	87.338	87.341	60.3-60.4

*different balance after test

Tab.3: Cumulative neutron fluence data and full-power days, and averaged cycle temperature

Id no.	Item	Sphere no. 1	2	3	4	5	6	7
1	Irradiation duration Full-power days	93.89						
2	Neutron fluence E > 0.1 MeV 10^{25} m^{-2}	1.0	1.39	1.69	1.9	1.95	1.83	1.47
3	Temperatures in °C, measured by thermocouples:							
	cycle 95.08	553	617	650	684	659	670	660
	cycle 95.09	523	580	638	674	670	691	672
	cycle 95.10	467	534	609	646	644	664	668
	cycle 95.11	538	596	645	672	665	678	652

Tab.4: Weight changes of the irradiated spheres

No.	Spheres	Weight before irradiation [g]	after irradiation [g]	difference [g]
1	A3-3 reference	198.21	197.68	-0.53
2	IG 110 (4)	209.85	209.38	-0.47
3	IG 110 (4) CVD	204.04	203.91	-0.14
4	IG 110 (6) CVD	203.99	203.86	-0.13
5	V483 (2)	207.26	206.91	-0.35
6	No. A	not investigated		
7	IG 110 (1) CVD	205.04	204.84	-0.20

Tab.5: Results of KORA corrosion tests

Sphere	1	2	3	4	5	7
Graphite	A3-3	IG 110	IG 110	IG 110	V 483	IG 110
Manufacturing	no SiC		CVD	CVD		CVD
Visual inspection after irradiation		surface defect	small hole	intact	intact	small hole
Weight loss during test (g)						
750 °C, 20 h	53.1	2.6	34.3	0	0	9.5
1000 °C, 20 h	-	5.9	-	21.1	0	-
1400 °C, 20 h	-	14.0	-	-	0	-
1600 °C, 20 h	-	-	-	-	*	-

*damage of furnace tube and sphere

Tab.6: Weight changes of the irradiated screwed spheres

sphere	weight before irradiation [g]	weight after irradiation [g]
HK-M1	200.09	200.15
HK-M2	198.72	198.77
HK-01	86.78	86.78
HK-02	86.66	86.67
HK-03	87.34	87.43

Tab.7: Results of corrosion experiments at 750°C,50h resp. 800°C, 12h in air streaming

ISR SiC-coated IG 110 graphite spheres (d=60mm) Si-infiltrated + CVD (Schunk) / 750°C, 50 h				
Sphere	Weight [g] of SiC-coated sphere	Weight [g] after 50h/750 °C in air	Δm [mg]	Diameter [mm] approx.
VK Si 1	212. 356	212. 353	-3	60.5-60.6
VK Si 2	211. 682	211. 683	+1	60.4-60.5
VK Si 4	212. 647	212. 655	+8	60.50
VK Si 6	211. 173	211. 186	+13	60.4-60.5
RWTH-Aachen SiC-coated IG 110 graphite spheres (d=60mm) slip-coated process / 800°C, 12 h				
VK-Trim14	206.30	206.30	0	60.35-60.45
VK-Trim18	206.85	206.85	0	60.40-60.42

Safety Aspects of HTR Technology

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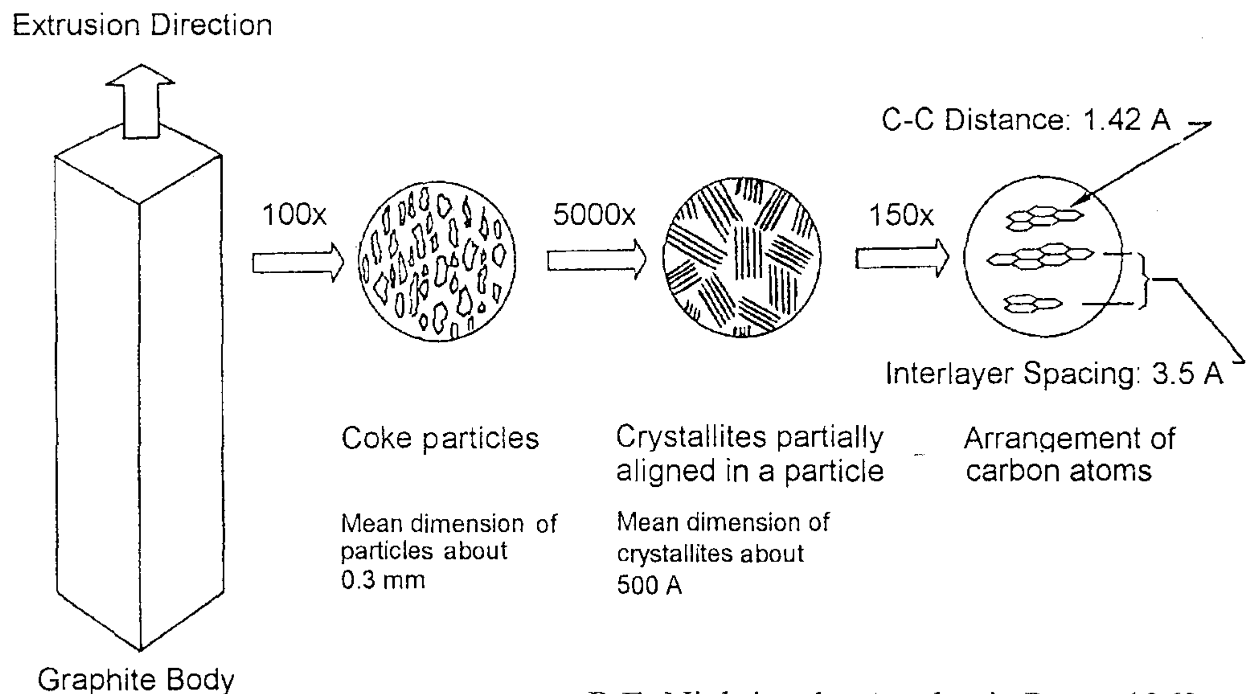
The substructures of polycrystalline graphite



R.E. Nightingale, Academic Press, 1962

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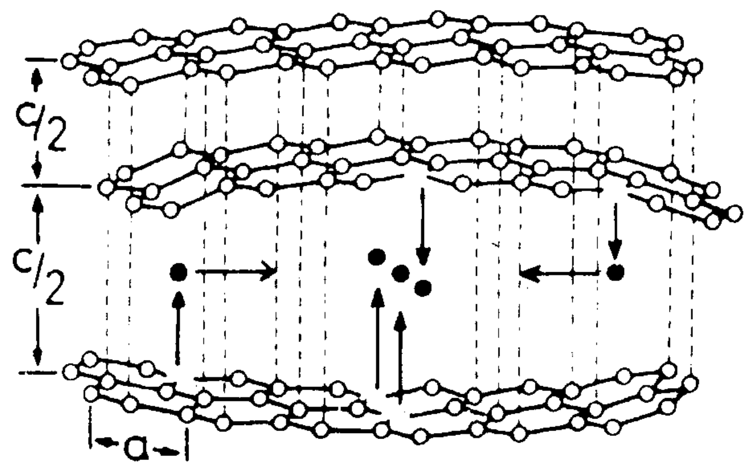
Radiation Damage in Graphite



○ C atoms within the graphite lattice

● Interstitial C atoms after neutron irradiation

Aus E. Fitzer: Graphit als Reaktorwerkstoff, Haus der Technik, 1967 (?)

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The number of C-atoms in graphite crystallites displaced by a single 1 to 2 MeV neutron is of the order of 20 000.

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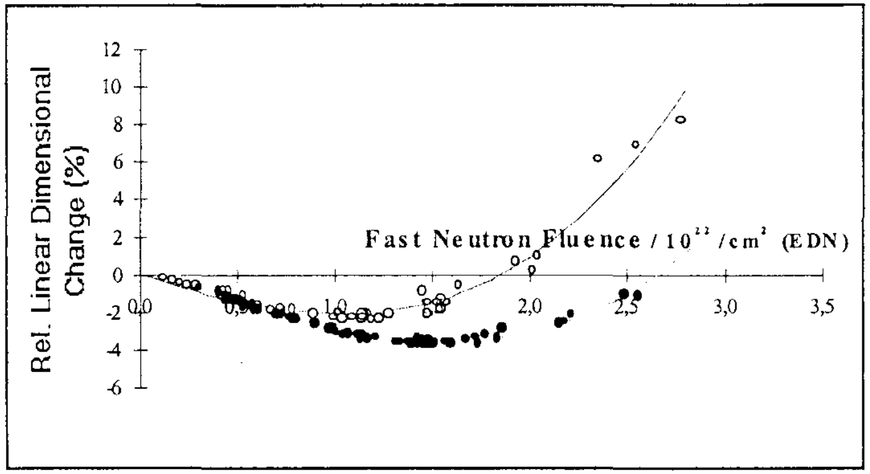
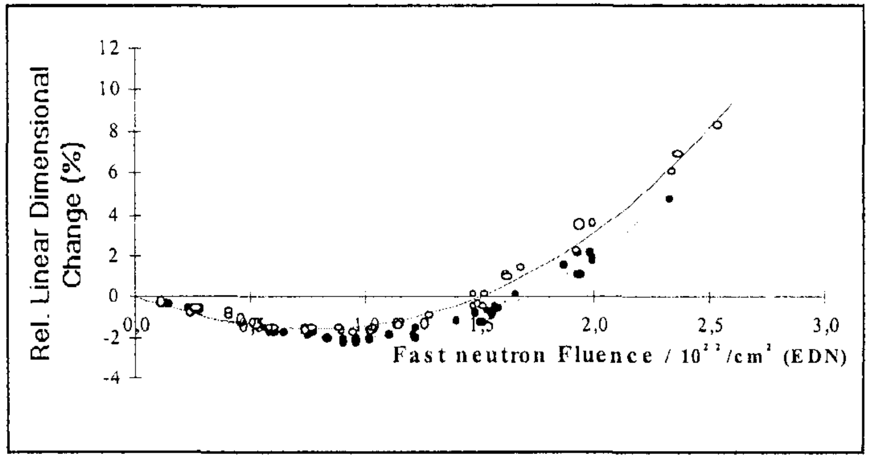
Irradiation Induced Dimensional Changes





Safety Aspects of HTR Technology

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<u>Influence of Coke and Forming Technique on Important Properties of Reactor Graphites</u>

Grade		**ATR-2E**	**ASR-1RS**	**ASR-2RS**	**ASR-1RG**
Coke		Special Pitch Coke	Ordinary Pitch Coke Sec. Coke T.	Ordinary Pitch Coke Sec. Coke T.	Ordinary Pitch Coke
Forming		Extrusion	Vibration	Vibration	Vibration
App. Density (g/cm^3)		1.80	1.82	1.87	1.78
Tens.Strength (N/mm^2)	*par.*	12.6	18.3	19.5	13.0
	perp.	12.4	18.3	18.5	11.6
Anisotropy		1.12	1.05	1.02	1.15

Safety Aspects of HTR Technology

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Statements I

- Nuclear graphite for non exchangeable core components must be nearly isotropic - but not isostatically moulded.
- Special coke processing and careful vibrational moulding yield the best graphite grades with respect to isotropy, strength, and homogeneity.
- The expected lifetime of graphitic core components has to be verified by stress analysis using reliable irradiation data.
- Today, none of the former widely tested graphites is still available.

Safety Aspects of HTR Technology

G. Haag: *Nuclear Graphite for the HTR - Research, Development, and Industrial Production*

Statements II

• Graphite for the PBMR reflector should be produced on a best guess basis using still existing procedures and experience.

• Data for stress analysis calculations should be deduced from similar materials tested in former irradiation programmes.

• Therefore, an international database with data from former nuclear graphite test programmes (US, UK, Japan, Germany, France) should be supported by possible users.

• For future HTR projects, development and irradiation testing of new graphites should be resumed as soon as possible.

critical stresses occur in the across grain direction when the extruded ATR-2E material is used, the vibrated ASR-1RS graphite can be used in the with grain direction, and from fig. 4, it can be seen that the dimensional changes of ATR-2E across grain and of ASR-1RS with grain are almost the same.

In HFR Petten, ASR-1RS (1975) graphite has accumulated a maximum neutron fluence of about 3×10^{22} neutrons/cm^2 (EDN) in about 10 years. Its changes of linear dimensions, Young's modulus, thermal conductivity, and coefficient of thermal expansion (CTE) by irradiation are known.

In the meantime, other grades (see table 1) have been developed and the question was raised whether their more or less different physical properties would lead to a different irradiation behaviour. However, the irradiation testing of all these later produced grades is not yet complete. Nevertheless, it looks as if their behaviour is not too much different from ASR-1RS (1975). This result is somewhat unexpected; before irradiation, ASR-1RS (1979) exhibits higher density, smaller CTE, and higher strength compared to the reference graphite but shows almost the same relative changes during irradiation (fig. 5).

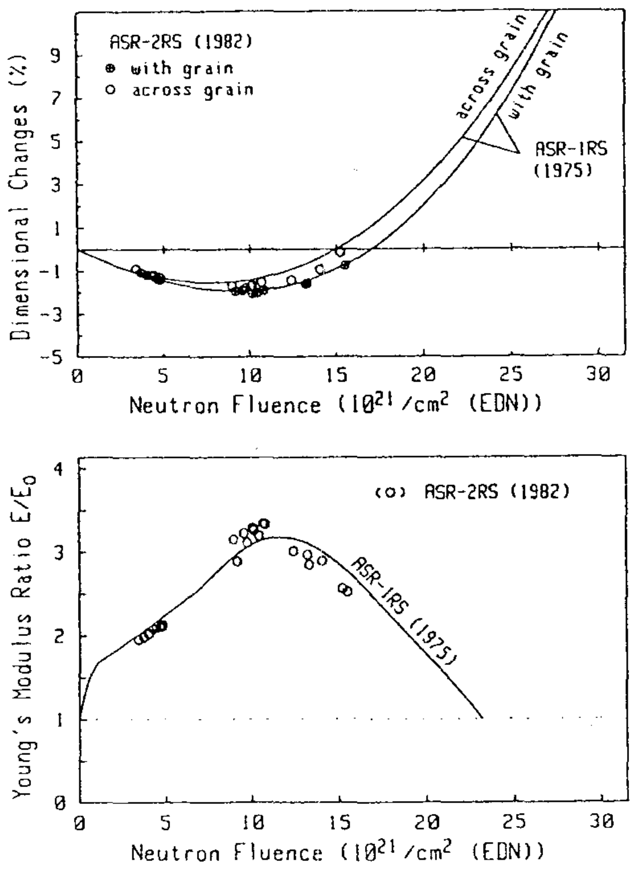

Fig. 6. Irradiation behaviour of ASR-2RS (1981) graphite at 500 °C in comparison with the reference grade ASR-1RS (1975).

The same impression is given by the ASR-2RS data in fig. 6. Even a second impregnation, yielding a bulk density as high as 1.87 g/cm^3, does not affect either the dimensional or the Young's modulus behaviour significantly.

If these results can be confirmed and are also valid at temperatures different from 500 °C, it could be concluded that the irradiation behaviour of graphite can be easily reproduced from batch to batch if only the same raw materials are used. Moreover, the irradiation testing of the graphite ASR-1RG (manufactured with regular pitch coke) will show if the secondary coke technique is essential for good irradiation behaviour.

For reactor designers these results may be reassuring. However, the question "How can the irradiation behaviour of reactor graphite be predicted from its physical properties?" remains as open as it was almost twenty years ago when G.B. Engle and W.P. Eatherly emphasized [5]: "The mechanisms of irradiation damage and crystallite changes and the relationships between crystallite and bulk dimensional changes have not been developed to the point where dimensional and volumetric changes of reactor graphites can be predicted accurately from pre-irradiation properties or structural features."

Acknowledgement

This work has been carried out in the frame of the project Hochtemperaturreaktor-Brennstoff-Kreislauf financed by the State of Northrhine-Westphalia.

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[1] R.E. Nightingale, Nuclear Graphite (Academic Press, New York, 1962).
[2] H. Rosswurm, G. Pietzka and W. Ulsamer, in: Preprints Int. Conf. Carbon '72, Baden-Baden (1972) p. 403.
[3] S. Wilkening, Extended Abstracts 13th Biennial Conf. on Carbon, Irvine (1977) p. 330.
[4] M.H. Wagner and W. Hammer, in: Preprints Int. Conf. Carbon '80, Baden-Baden (1980) p. 502.
[5] G.B. Engle and W.P. Eatherly, High Temp. – High Press. 4 (1972) 143.

As the irradiation temperature in HFIR experiments is calculated before and verified after the irradiation (monitoring and controlling equipment are not available during irradiation), in the German research programme HFIR data are only used to compare the irradiation behaviour of different graphite grades, whereas design data are created in the High Flux Reactor (HFR) Petten at neutron fluences about 4 times smaller, but at well controlled temperatures. Capsules equipped with thermocouples and containing about 90 to 120 specimens are kept at constant temperature by gas mixture and by shifting the capsule vertically as fuel is burned and control rods are raised.

The first batch of ASR-1RS graphite was produced in 1975 when the graphite grade ATR-2E (made from special pitch coke) was still under consideration and had already proven its good irradiation behaviour. To determine whether using the secondary coke technique and vibrational moulding is a better way to produce reactor graphite, and to focus the development programme on the most promising materials, it was necessary to get to know the irradiation behaviour of ASR-1RS as soon as possible. Therefore, all candidate graphites have been irradiated in three steps in HFIR. After accumulation of 2×10^{22} neutrons/cm^2 (EDN) (which is equivalent to about one year effective irradiation time) significant differences between the candidate graphites became visible. With regard to its very small anisotropy and good strength, ASR-1RS graphite was defined as a reference material for the highly loaded parts of the reflector.

Later, when the irradiation results from HFR Petten experiments became available, this decision was confirmed. Fig. 4 shows that ATR-2E graphite exhibits higher maximum shrinkage and higher anisotropy than ASR-1RS (both leading to higher irradiation induced stresses), but assumes the original length at a higher fluence in both directions. However, considerably larger blocks are manufactured by the vibrational moulding technique compared to extruding. In all cases where the

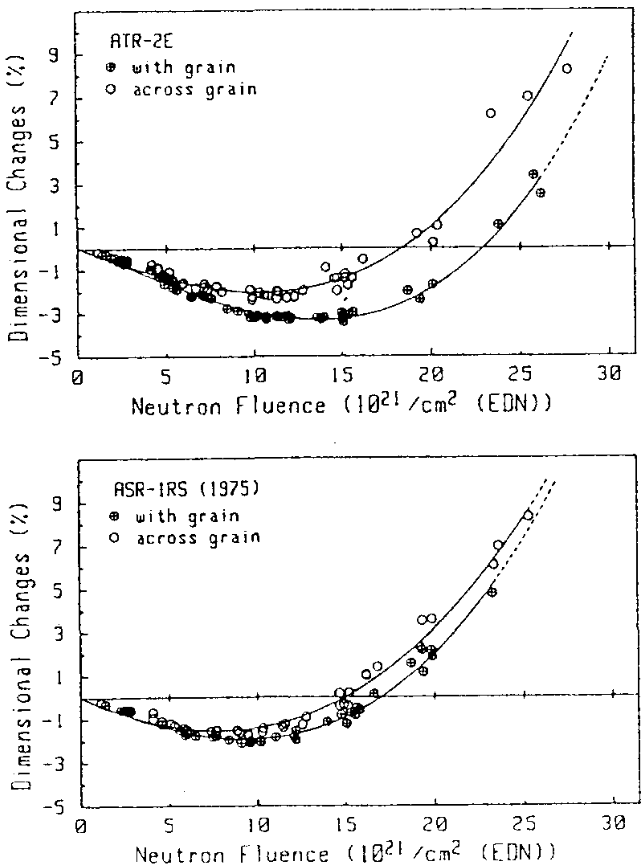

Fig. 4. Irradiation behaviour of extruded ATR-2E graphite at 500 °C in comparison with the reference grade ASR-1RS (1975).

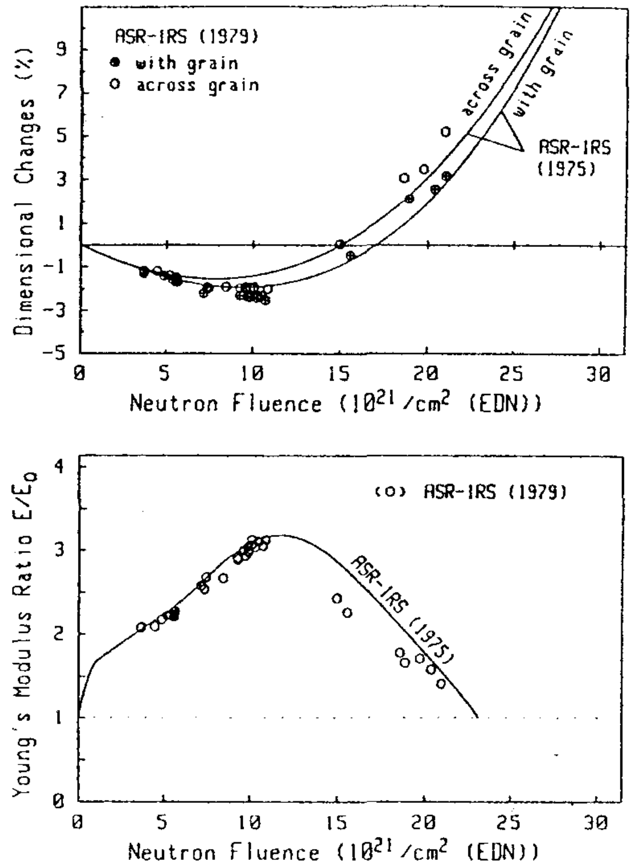

Fig. 5. Irradiation behaviour of ASR-1RS (1979) graphite at 500 °C in comparison with the reference grade ASR-1RS (1975).

Thus, effort has been concentrated on graphite grade V483 and several changes of the production technique have been investigated to improve the baking behaviour of blocks as large as Ø370 × 900 and 370 × 300 × 900 mm^3.

The irradiation testing has been done primarily in the HFIR reactor at Oak Ridge, directly comparing all the concurrent graphites. Therefore, neither temperature regulation was necessary nor flux effects (if existing) had to be considered. It turned out that V483 graphite exhibited faster and higher initial shrinkage due to its relatively high binder content. After the dimensional changes reversed, rapid expansion occurred, which obviously would lead to higher stresses in graphitic reactor components. This was the reason for giving up the V483 material. Being a fine grain graphite its fracture behaviour is expected to be worse than that of coarse grain graphites and the relatively high strength was not high enough to compensate this disadvantage.

3.6. Developing graphite for core support columns

Originally, graphite for core support columns had been produced with the same mixture V483 as was used for developing graphite for reflector components. However, the required block dimensions and physical properties were different, and other steps had to be taken to realize the desired product. With the available press, it was not easy to achieve the block dimensions of Ø155 × 1980 mm^3 after machining and, at the same time, to get high strength and high corrosion resistance.

Table 2
Properties of graphite V483 for core support columns

Property [a]		Grade T	T2	T5	T6
Ash content (ppm)		240	70	13	32
Density (g/cm^3)		1.76	1.78	1.78	1.77
CTE (20–500 °C)	(∥)	3.57	3.59	3.83	3.93
(10^{-6}/K)	(⊥)	4.01	3.87	4.18	4.48
Anisotropy factor		1.12	1.08	1.09	1.14
Youngs' modulus	(∥)	9.5	9.5	11.1	11.3
(kN/mm^2)	(⊥)	8.4	9.1	10.7	10.9
Bending strength	(∥)	28.6	38.9	51.6	55.3
(N/mm^2)	(⊥)	25.0	37.7	47.3	51.3
Compressive strength	(∥)	59.9	77.8	103	111
(N/mm^2)	(⊥)	62.1	79.9		
Tensile strength	(∥)	17.2	25.0		
(N/mm^2)	(⊥)	15.5	24.5		
Thermal conductivity	(∥)	143	111	94	86
(W/m K)	(⊥)	134	106	87	79

[a] (∥) = parallel, (⊥) = perpendicular to grain orientation.

The first reasonable result was the graphite grade V483T; its properties are listed in table 2. However, it turned out that using still smaller filler grain significantly improved values for strength could be obtained and simultaneously, an important reduction in ash content was achieved. This improved grade is referred to as V483T2 in table 2.

This stage of development was reached in 1982. In the meantime, no further pebble-bed reactor has been built. However, the best way to conserve this knowledge is to continue developing graphites for future applications and to guarantee long-term availability. This led the Ringsdorff-Werke to the decision to use other raw materials, and to find out how to reproduce or even to improve the properties of graphite for core support columns.

The data of grades V483T5 and V483T6 in table 2 show that these efforts have been very successful. The good bending and compressive strength values of V483T2 have again been increased by about 30% (T5) and 40% (T6) and the ash content has been decreased by more than a factor of 2 in both grades.

4. Irradiation testing

Irradiation induced stresses resulting from the Wigner effect are most important for the selection of graphite for application at high fast neutron fluences. The graphite lattice is fundamentally damaged in a reactor environment by collision of high-energy neutrons with carbon atoms in the lattice. The carbon atoms are displaced to interstitial positions, leaving behind vacant sites in the layer planes. Some of the vacancies and interstitial atoms are immediately annealed by recombination, but those remaining may concentrate, depending upon the conditions of neutron fluence and irradiation temperature, and form larger clusters.

Since the German HTR concepts are based on unexchangeable reflector components, lifetime fluences are particularly high ranging from 3 to 4 × 10^{22} neutrons/cm^2 on the so-called Equivalent-Dido-Nickel (EDN) scale. Therefore, irradiation testing of HTR core materials is done in material test reactors such as the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory or the High Flux Reactor (HFR) at Petten Joint Research Center, where significant neutron fluences are accumulated at high neutron flux.

Table 1
Properties of German graphites for the reflector of High Temperature Reactors

Property [a]		Graphite grade and year of production			Improved ASR-1RS 1979	Demonstration grades		
		Reference grades				ASR-1RS 1986	ASR-2RS 1986	ASR-1RG 1986
		ASR-1RS 1975	ASR-2RS 1981	ASR-1RG 1981				
Ash content (ppm)		390	40	130	490	640	620	570
Density (g/cm^3)		1.78	1.87	1.79	1.81	1.84	1.88	1.79
Lin. therm. expansion	(‖)	4.70	3.92	3.50	4.22	3.62	3.62	3.15
coefficient (10^{-6}/K)	(⊥)	4.87	4.12	3.95	4.42	3.85	3.84	3.72
Anisotropy factor		1.04	1.05	1.13	1.05	1.06	1.06	1.18
Young's modulus	(‖)	9.9	10.5	8.7	10.2	10.1	11.1	8.7
(kN/mm^2)	(⊥)	9.2	10.1	7.7	9.8	9.9	10.7	7.6
Bending strength	(‖)	26.0	28.5	19.4	26.9	24.7	27.8	16.1
(N/mm^2)	(⊥)	23.0	28.2	17.1	26.4	25.8	28.6	15.5
Compress. strength	(‖)	67.1	79.8	47.0	66.5	62.5	73.4	39.8
(N/mm^2)	(⊥)	63.1	80.0	47.8	66.5	64.8	75.7	41.5
Tensile strength	(‖)	14.9	19.1	12.0	17.9	17.2	18.8	10.6
(N/mm^2)	(⊥)	13.5	18.1	10.9	18.1	17.5	19.6	10.1
Thermal conductivity	(‖)	125	146	154	134	155	162	162
(W/mK)	(⊥)	125	142	136	130	149	157	141

[a] (‖) = parallel, (⊥) = perpendicular to grain orientation.

highest possible strength is required.

Some parts of the core structure of a pebble-bed HTR (especially in the so-called HTR-Modul reactor) are exposed to only small or medium neutron fluences. For this purpose it has been suggested to use a cheaper graphite made directly from ordinary pitch coke by vibrational moulding with only one impregnation. This material exhibiting higher anisotropy and smaller strength is called ASR-1RG. Some physical properties of all reference grades are listed in table 1.

The development of isotropic reactor graphite with optimized properties raises the problem of how the properties change from batch to batch or when the production technique is scaled up. As an example, table 1 shows that the transition from ASR-1RS(1975) produced in laboratory scale, to ASR-1RS(1979) produced in preproduction scale, resulted in remarkable property changes. This led to the decision that, at the end of the development programme, so-called demonstration batches of all reference grades had to be produced in production scale, i.e. in the order of 30 to 40 full size blocks. The question, if the observed variations in some of the physical properties are significant to the design of graphitic reactor components, remains upon until the irradiation testing is done.

3.5. Developing isostatically moulded graphites

Based on the experience that reactor graphites for core components must be isotropic, the isostatic moulding technique has to be taken into account when, in the early 1970s, graphites made from Gilsonite coke had to be replaced. At that time, the Ringsdorff-Werke company had a lot of experience with that moulding process. However, the block dimensions had to be increased by a factor of 2 to 3 for reactor purposes and at that time there was only slight experience of the neutron irradiation behaviour of isostatically moulded graphite.

These conditions seemed to be somehow contradictory since isostatic moulding requires small grain size of the filler but high binder content. Consequently, the risk of graphite blocks cracking during the baking process increases with increasing block size and also a higher binder content, in general, leads to a decrease of dimensional stability under irradiation.

In the first stage, different cokes, for example regular petroleum coke and pitch coke, had been taken into consideration which showed that there might be some preference for petroleum coke. However, this development has also been influenced by the decision to use domestic raw materials exclusively.

inert mass of the covering weight. Evacuating the mould before and during densification improves the final density and the homogeneity of the product.

The optimization of the vibrational moulding process had to take several parameters into account: The best *vibration frequency* is dependent on the composition of the green mixture and of the height of the block; the optimum frequency ranges from 20 to 35 cps. The optimum *amplitude* increases from 1.5 to 4 mm with increasing height of the block. Amplitudes below the optimum lead to smaller densities, higher amplitudes favour crack formation. The *covering weight* depends on the composition of the green mixture, the height and the height to cross section ratio of the block. The optimum pressure ranges from 150 to 500 g/cm^2. The *vibrating time* is less than 1 min.

By carefully tuning the frequency and amplitude, a stable counterphase oscillation of the vibrating table and the covering weight is adjusted, which provides steady progress of compression and good homogeneity of the blocks.

3.3. Improving the graphitization behaviour

In the initial stage of the development of isotropic reactor graphites based on coal tar pitch coke, many graphite blocks cracked during graphitization. At that time, experiences concerning the expansion/shrinkage behaviour of carbon materials were limited to anisotropic grades (e.g. from the examination of puffing effect), whereas the behaviour during graphitization of isotropic carbons based on pitch cokes had not yet been investigated in detail.

After observing the dimensional behaviour of different baked carbon grades in a high temperature dilatometer from 100 to 2400 °C, it became clear that carbons based on petroleum-needle coke behave different from those based on secondary coke. Fig. 3 shows that these materials behave similarly at temperatures below the baking temperature of 900 °C where they both expand linearly, but at higher temperatures the secondary coke carbon reverses its dimensional behaviour, whereas the needle-coke material continues expanding – slowly from 900 to 1700 °C and again faster above 1700 °C. The pronounced shrinking at temperatures above the baking temperature was found to be characteristic for carbon materials based on isometric coke and, when combined with the higher Young's modulus of isotropic graphite, is believed to cause the poor graphitization behaviour.

Numerous test series have been performed to investigate the influence of various production parameters. As can be seen from fig. 3, adding graphite to the

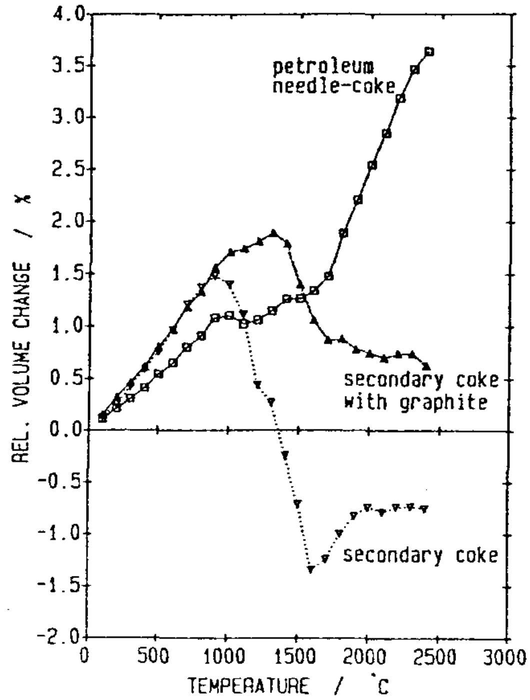

Fig. 3. Dimensional behaviour of baked carbon based on different fillers in the temperature range from 100 to 2400 °C.

secondary coke filler markedly reduced the shrinkage during the graphitization process. Another positive influence on the dimensional behaviour has been found from impregnating and rebaking at higher temperature [4]. Therefore, to prevent isotropic carbon artifacts from cracking the following steps have been established:

- adding about 15 to 20% ground graphite of the same grade to the final mixture (which does not change the properties of the product),
- increasing the maximum baking temperature from 900 to 1100 °C (which improves the properties of the product).

3.4. Reference graphites

A wide range of different graphite grades have been irradiation tested (see section 4). Finally, the decision was made in favour of the graphite grade ASR-1RS. Its smaller anisotropy, larger block sizes and appreciably higher strength had been decisive.

As there is only one impregnation necessary to achieve such good properties, it is possible to improve this graphite again by a second impregnation. This leads to the grade ASR-2RS to be used in all cases where the large block size areas in fig. 1) are almost entirely protected against fast neutrons by the bottom reflector, the top reflector and the inner side reflector. Therefore, the design of these components does not depend on any irradiation behaviour. The strength requirements depend on the special function of the components but normally are of minor importance compared to the inner reflector components. However, development and production of core support blocks with dimensions of $500 \times 500 \times 2000$ mm^3 was a major problem. Because the temperatures of these components under operation are particularly high, good chemical purity was necessary to achieve low corrosion rates.

The core support columns are also part of the support construction (see fig. 1). They must have very high strength, in particular as they are exposed to cooling gas streams of different temperatures leading to thermal stresses. Core support columns are also in danger of corrosion and therefore are made from graphite with small ash content.

3. Developing reflector graphite for High Temperature Reactors

The HTR graphite development programme was originally based on two main aspects:

Development of new cokes with isotropic structure and isometric grain made from coal tar pitch to produce reactor graphite using conventional methods (e.g. forming by extrusion).

Development of new methods for the production of isotropic graphites from commercially available anisometric pitch coke.

However, the demand for raw materials for nuclear purposes has been estimated to be too small to make every effort to develop new cokes. Actually, there was only one special pitch coke available to produce the ATR-2E graphite which exhibited good physical properties and excellent irradiation behaviour without modifying the conventional production process. This experience led to the conclusion that the development of new production techniques using ordinary pitch coke might be successful.

3.1. Secondary coke technique

Normally, the industry produces graphite from well graphitizing cokes with a more or less pronounced layer structure. These cokes fracture parallel to the layer planes when they are ground, leaving anisometric grains which may be needle-like in the extreme. Depending on the pressing technique, anisometric grains can show preferred orientation, which makes the final product anisotropic.

The preferred orientation of the grains can be avoided using a particular preproduction technique [2]: The precurser coke with 0.12 mm maximum grain size is mixed with standard coal tar pitch binder using a fast mixer. Then, the hot mixture is compressed by vibration moulding (see section 3.2.) yielding large blocks to be baked at temperatures above 1100 °C and then ground to a maximum grain size of 1 mm. Thus, an isotropic coke aggregate is obtained with high bulk density, excellent mechanical properties, low contents of ash and volatiles, and nearly spherical grains.

This secondary coke is used exactly like conventional filler; it is mixed with coal tar pitch binder in a fast mixer, with 20% ground isotropic nuclear graphite added (see section 3.3.) to improve the baking and graphitization behaviour.

3.2. Vibrational moulding

In principle, the preproduct technique can be used in combination with all forming methods. Nevertheless, reflector blocks for future High Temperature Reactors are so large, that forming by moulding or by extrusion had to be given up for cost reasons and the vibrational moulding technique [3] has been developed. The equipment and the procedure are shown in fig. 2. The vibration moulding machine consists of a vibration table, a mould, a covering weight with a guide rod and a vacuum device.

The hot green mixture is poured into the mould directly from the mixer. Compression of the mixture results from vertical vibrations of the mould against the

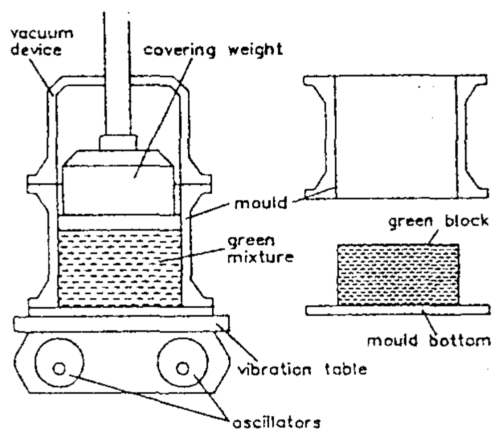

Fig. 2. Schematic diagram of the vibrational moulding process.

Germany. During the early 1970s it became clear that only isotropic graphites might be able to meet the needs of future pebble bed High Temperature Reactors.

The most promising way of manufacturing isotropic graphites had been based on Gilsonite coke as filler, a naturally occurring bitumen mined in Utah, USA, which consists of almost spherical grains. However, for some reasons the longterm availability of Gilsonite coke had to be called into question and the first oil crisis in 1973 led to the decision to use only domestic raw materials for German reactor graphites.

2. HTR requirements

The German development programme for reactor graphite is based on the cooperation of the companies Hochtemperatur-Reaktorbau GmbH (HRB) and Gesellschaft für Hochtemperaturreaktor-Technik GmbH (GHT)/Interatom GmbH as nuclear reactor constructing industry, Sigri GmbH and Ringsdorff-Werke GmbH manufacturing graphite and the Kernforschungsanlage Jülich GmbH (KFA).

As an example for future High Temperature Reactors, fig. 1 shows schematically the core construction of a HTR-500 pebble-bed reactor. The different requirements with respect to neutron fluence, mechanical load, chemical impact or block size led to different goals of the graphite development programme.

2.1. Reflector components

Unexchangeable components for the high- and low-fluence zones are called reflector components. Their main purpose is to reflect and moderate the neutrons escaping from the reactor core.

Because the reflecting power depends on the number of carbon atoms per cm^3, the density of reflector graphite should not be less than 1.70 g/cm^3.

With respect to neutron physics, the overall neutron-capture cross section is required to be smaller than 5 mbarn. Consequently, the content of neutron-absorbing chemical elements such as Gd, B, Sm, and Eu, has to be kept small. With respect to corrosion by impurities from the cooling gas such as O_2, H_2O and CO_2, catalytically acting chemical elements such as Fe, Ca, Sr and Ba are important. Experience shows that all this can be summarized by an upper limit for the ash content of nuclear graphite of about 600 ppm.

For the use of graphite as structural reactor material, the mechanical stability – primarily the tensile strength – has turned out to be most important. As a result of the Wigner effect and of temperature gradients, internal

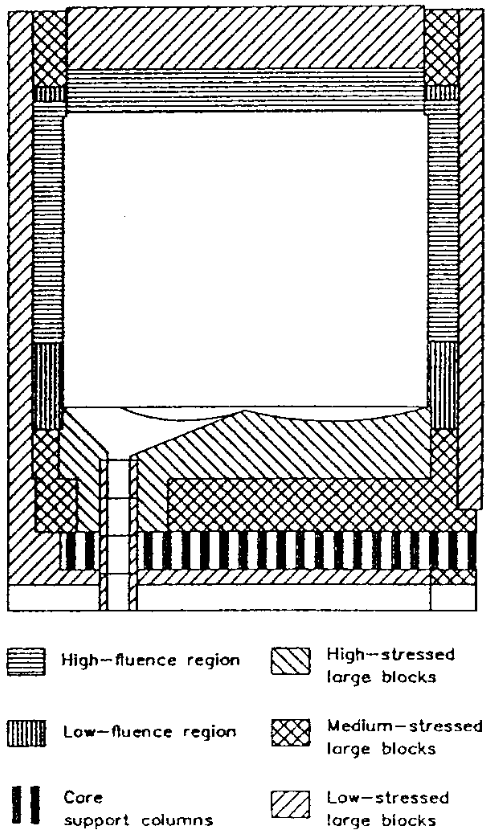

Fig. 1. Vertical section of a pebble-bed HTR (schematic diagram).

stresses are created in the components which might be damaged severely if the stresses exceed the strength. Therefore, some other physical properties are supposed to satisfy limiting conditions such as dynamic Young's modulus not to exceed 12 kN/mm^2, thermal conductivity higher than 90 W/m K at room temperature, and coefficient of linear thermal expansion (CTE) smaller than 6×10^{-6}/K from 20 to 500 °C.

The anisotropy factor of graphite is defined as the ratio of the linear thermal expansion coefficients in the two main crystallographic directions. For the construction of reactor components only isotropic graphites with anisotropy factors from 1 to 1.05 are stable enough against fast neutron irradiation damage. A compromise had to be found between small CTE-values to be achieved using anisotropic coke and isotropic products based on coke with high CTE.

2.2. Core support construction and outer parts of the reflector

Graphite components belonging to the core support construction or to the outer parts of the reflector (see

Journal of Nuclear Materials 171 (1990) 41–48
North-Holland

DEVELOPMENT OF REACTOR GRAPHITE *

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Received 30 June 1989; accepted 15 July 1989

The German graphite development programme for High Temperature Reactors has been based on the assumption that reactor graphite for core components with lifetime fluences of up to 4×10^{22} neutrons per cm^2 (EDN) at 400 °C can be manufactured from regular pitch coke. The use of secondary coke and vibrational moulding techniques have allowed production of materials with very small anisotropy, high strength, and high purity which are the most important properties of reactor graphite. A variety of graphite grades has been tested in fast neutron irradiation experiments. The results show that suitable graphites for modern High Temperature Reactors with spherical fuel elements are available.

1. Introduction

Graphite is an almost perfect reactor material. In 1942, Enrico Fermi already used graphite as moderator in the first nuclear reactor fashioned by the hand of man. In the course of nuclear reactor development, the physical and chemical properties of graphite have been responsible for many important advances in nuclear reactor technology.

It is obvious that on the way to modern High Temperature Reactors (HTR) providing temperatures of about 1000 °C severe technical problems had to be solved, but in no case was graphite with its good mechanical and excellent thermal properties the reason for these problems. Only the future reactor concepts HTR-500, HTR-100 and to some extent the Modular HTR make use of graphite in a way that limiting factors have to be considered. This was the reason for the development of new isotropic graphites and for irradiation programmes to investigate the influence of the Wigner effect (i.e. fast neutron radiation damage in the graphite crystal structure) on the physical properties of graphite at very high neutron fluences.

Most of the nuclear graphites have been produced by the Acheson process which for several decades had been used for the production of furnace electrodes [1]: A coke filler is mixed with thermoplastic hydrocarbon binder, pressed to form a "green" artifact and then in two steps is subjected to a more or less complex heat treatment. The first step consists of the baking process at temperatures of about 1000 °C, converting the binder into almost pure carbon, while in the second step the baked material is graphitized by electric resistance-heating to a final temperature of 2600 °C or higher. In some cases where higher density is required, the porosity resulting from the raw materials and the baking process is decreased by impregnation with pitch or tar prior to graphitization.

The fast neutron irradiation behaviour of polycrystalline graphite had been studied for many years in the frame of the Advanced Gas-cooled Reactor programme and the OECD Dragon Reactor Project in Britain, the Experimental Gas-Cooled Reactor (EGCR) program and the High-Temperature Gas-cooled Reactor (HTGR) program in the USA and the AVR and THTR Pebble Bed Reactor programmes in the Federal Republic of

* Dedicated to Prof. H. Nickel on the occasion of his 60th birthday.

Forschungszentrum Jülich

Institute for Safety Research and Reactor Technology ISR

Heat Transfer, Fluid Flow and Power Feedback in Pebble-Bed Reactors

by

W. Scherer

presented on the occasion of the
Visit of a NRC Delegation to Germany
on the topic

Safety Aspects of HTR Technology

July, 23-26, 2001
Cologne and Jülich, Germany

18. 07. 01

Heat Transfer Mechanisms

- **Conduction in Coated Particles**

  important in extremely fast transients

- **Conduction inside pebbles**

  Conductivity depends on temperature and fast neutron fluence

- **Transfer from pebbles to coolant**

  Nusselt's law according to experimental results

  KTA-rule

- **Transfer from pebble to pebble**

  by radiation and conduction via pebble contacts

  effective heat conductivity from theoretical considerations validated at experimental results

- **Convective Transport in Coolant**

  2-dimensional models applied

Basic Statements on Heat Transfer in Pebble Beds

- **Temperature profile in pebbles**

  leads to temperature differences < 70 °C in normal operation

- **Temperature jump at pebble surface**

  is normally < 30 °C

- **Effective Conduction in Pebble bed**

  determines the max. fuel temperature in depressurization events. Tmax < 1600 °C

- **Convective Transport in Coolant**

  plays also a significant role during LOFC events (natural convection) and determines there the temperature shift upward and the temperature loads on metallic components

Fluid Flow in Pebble Beds

- **Forced Flow maintained by blower (steam-cycle) or turbo-compressor unit (direct cyc.)**

  Pressure drop relations from experiments.
  KTA rule.

- **Natural convection flow driven by buoyancy forces caused by temperature distribution in core**

  Natural convection causes heat transport from hot to cold areas of the core if forced convection is not present.

Basic Statements on Fluid Flow in Pebble Beds

❖ Detailed Flow around pebbles hardly to model

Models assume homogeneus massflow in 2 dimensions. Pressure drop relations from experiments. KTA rule. Pressure drop is about 0.05 to 0.1 MPa

❖ Quasi-Steady-State Flow assumption in normal operation

Inertia forces are small in Helium which justifies the assumption for model calculations

❖ Natural convection flow significant only at elevated pressure

In LOFC situations nat. conv. causes heat transp. mainly in upward direction and heats up top core structures. In DLOFC situations nat. conv. may be neglected

Basic Statements on Heat Removal/Power Feedback

- **Reduced Heat Removal Reduces Nuclear Power**

  Strong negative feedback via fuel and graphite temperature. Easy power control by mass-flow without significant temperature changes

- **Loss of coolant performs Reactor Shut-down**

  Xenon decay leads to recriticality after long time

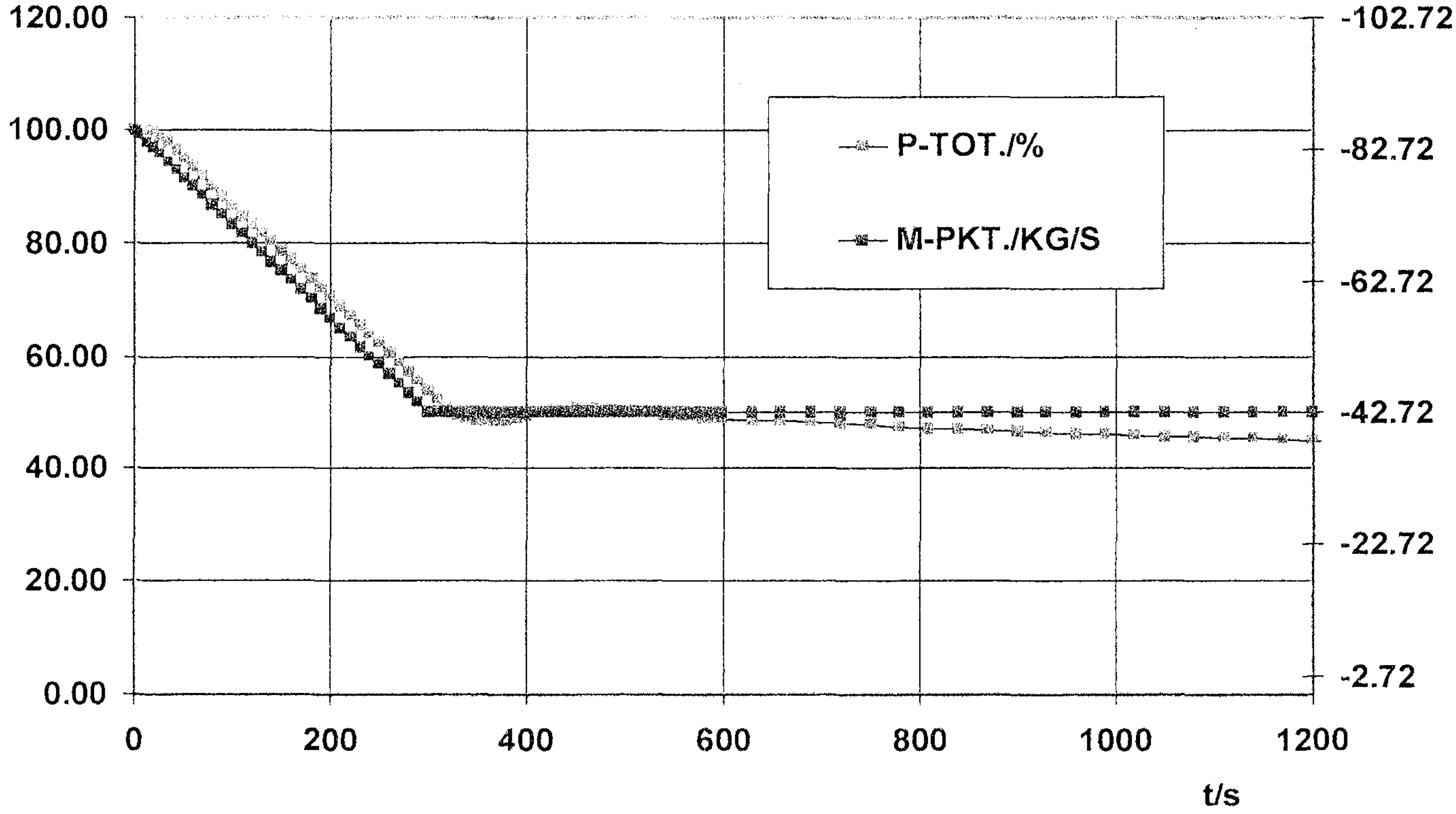
- **Loss of forced cooling initiates Natural convection flow with earlier recriticality**

  Better cooling leads to lower temperature increase with less subcriticality

MODUL-200

(TINTE Transients)

50% Part-Load (massflow; steamgenerator)



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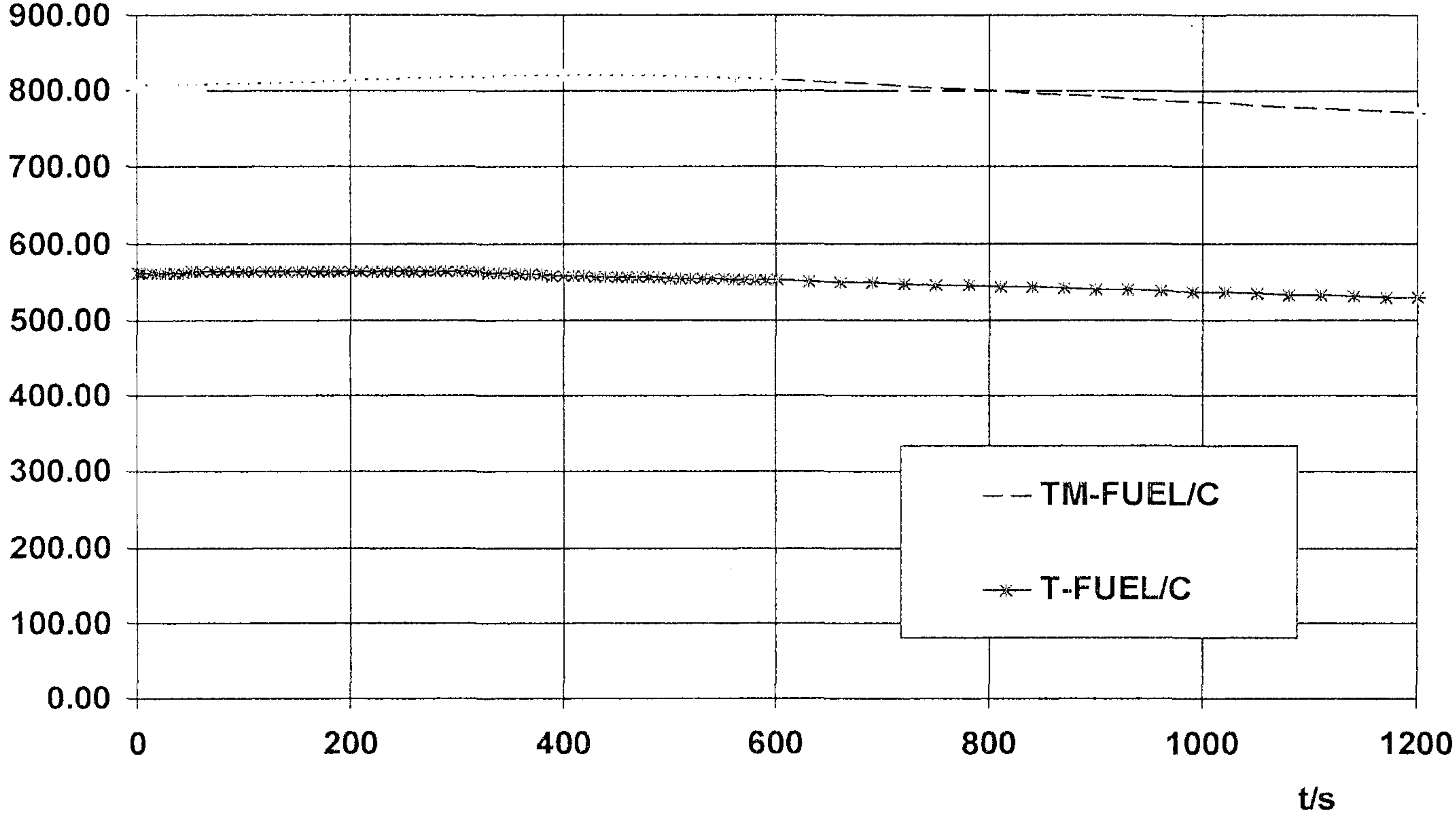
MODUL-200

(TINTE Transients)

50% Part-Load (massflow; steam-generator)



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MODUL-200

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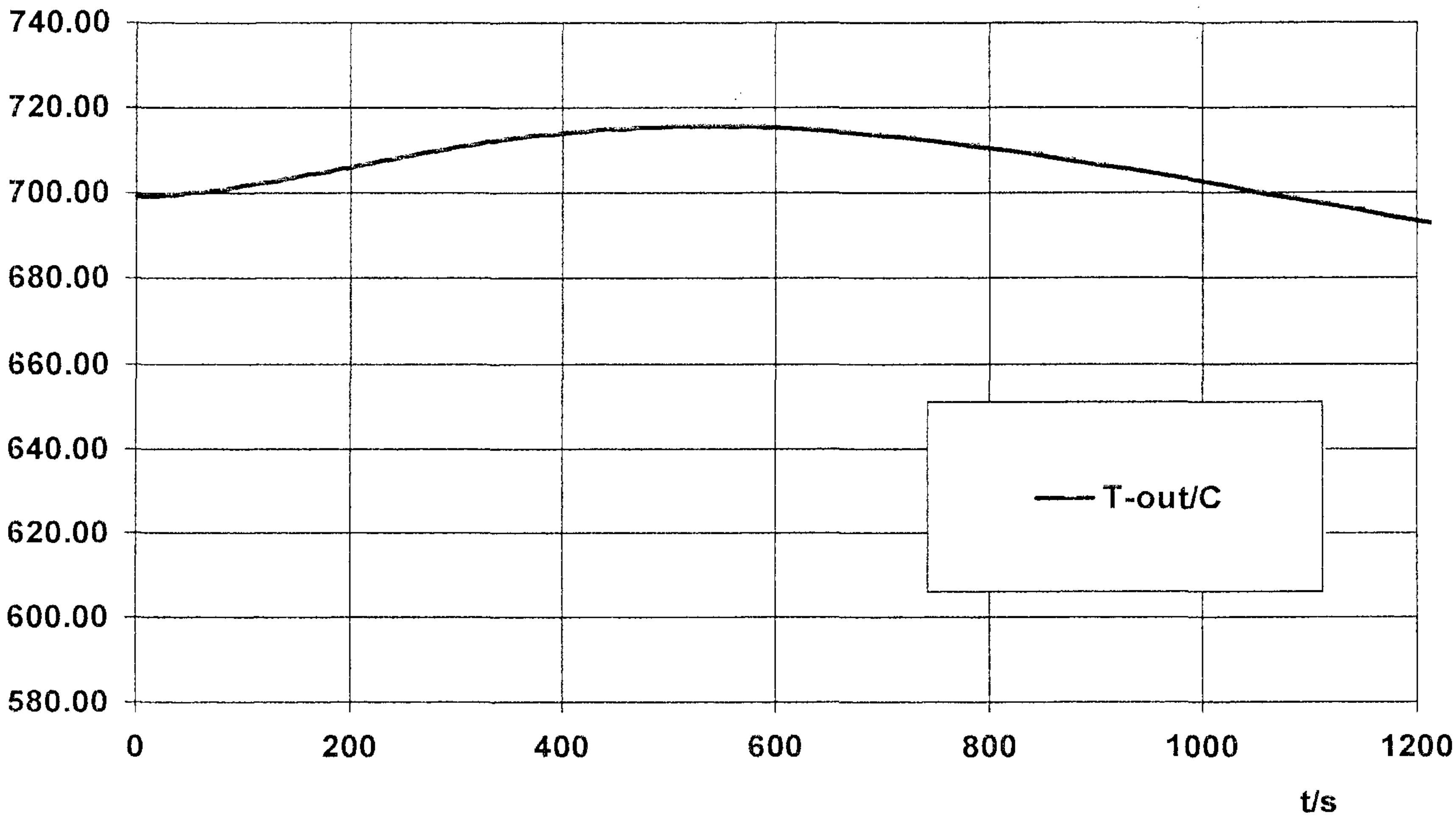

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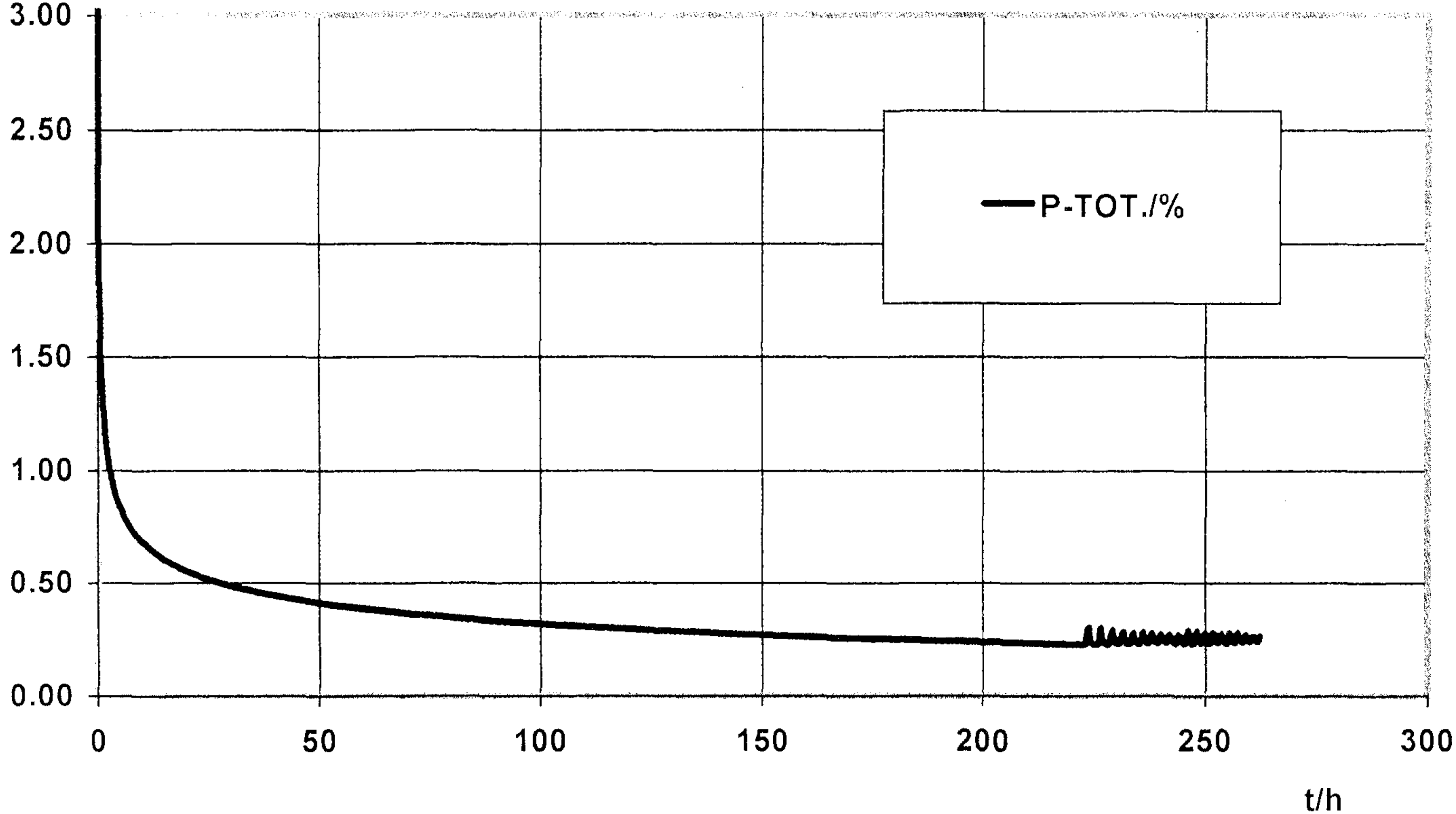
MODUL-200

(TINTE Transients)
LOCA Depressurisation



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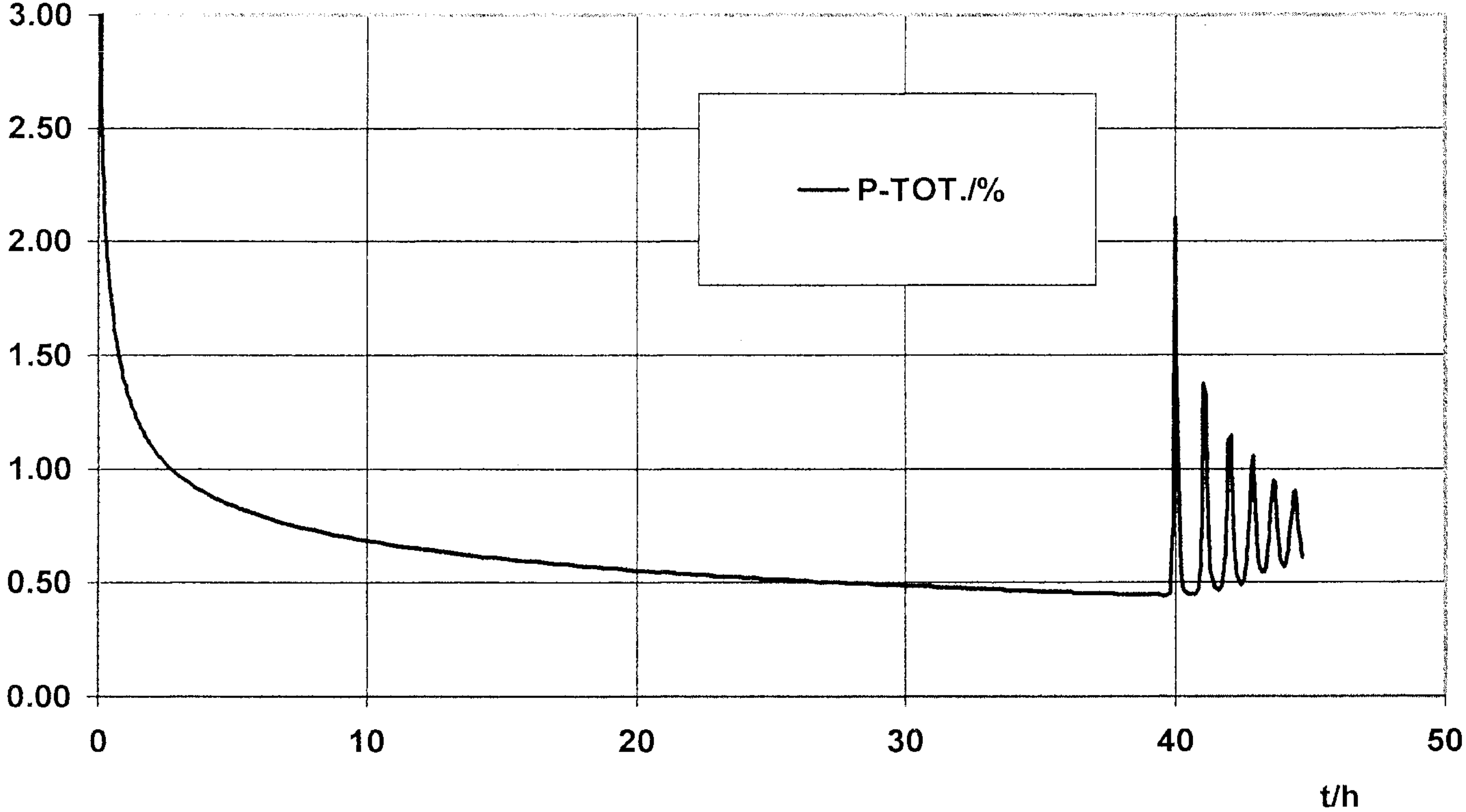
MODUL-200

(TINTE Transients)

LOFC with Shortcut over Circulator



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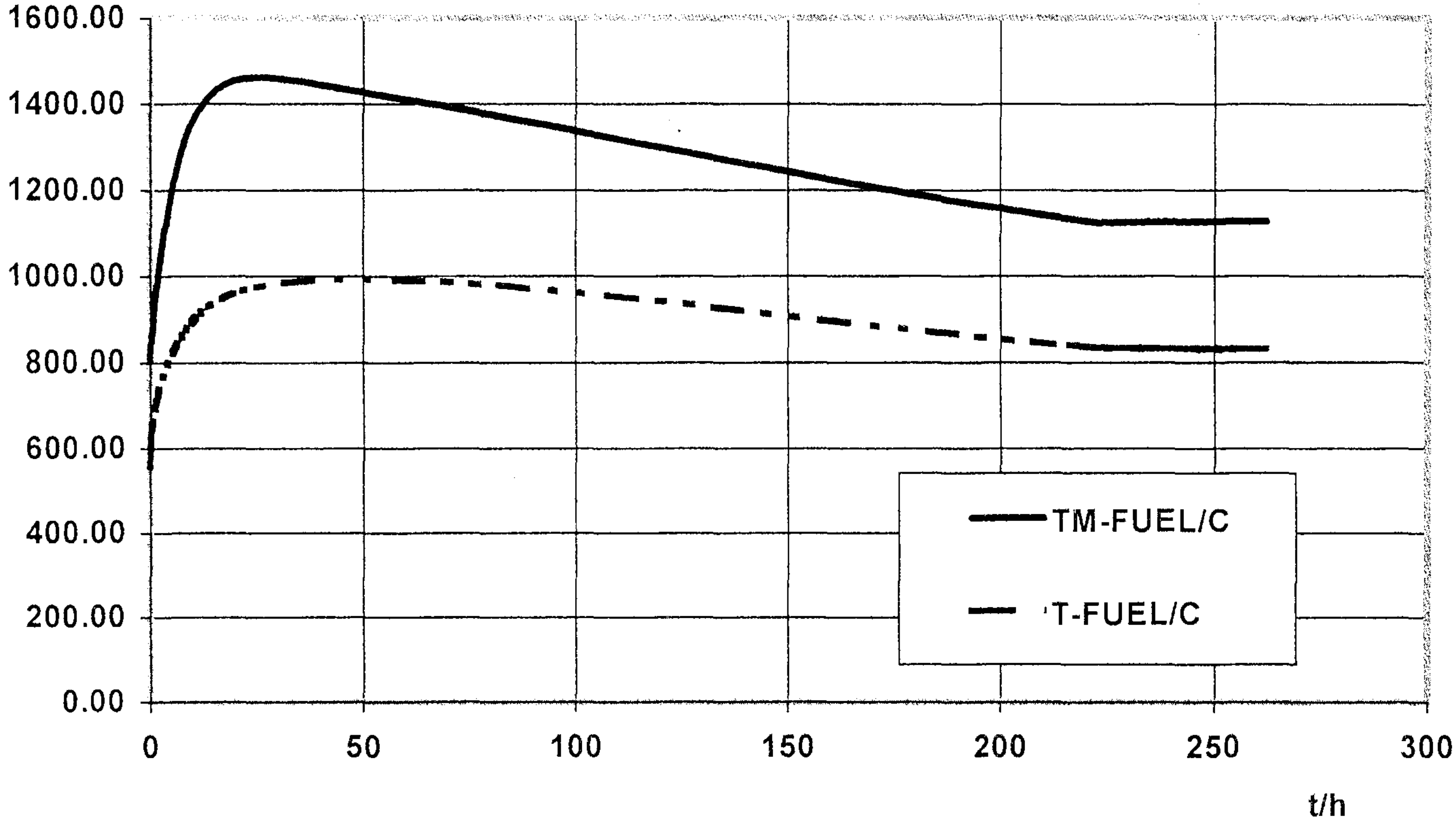
MODUL-200

(TINTE Transients)
LOCA Depressurisation



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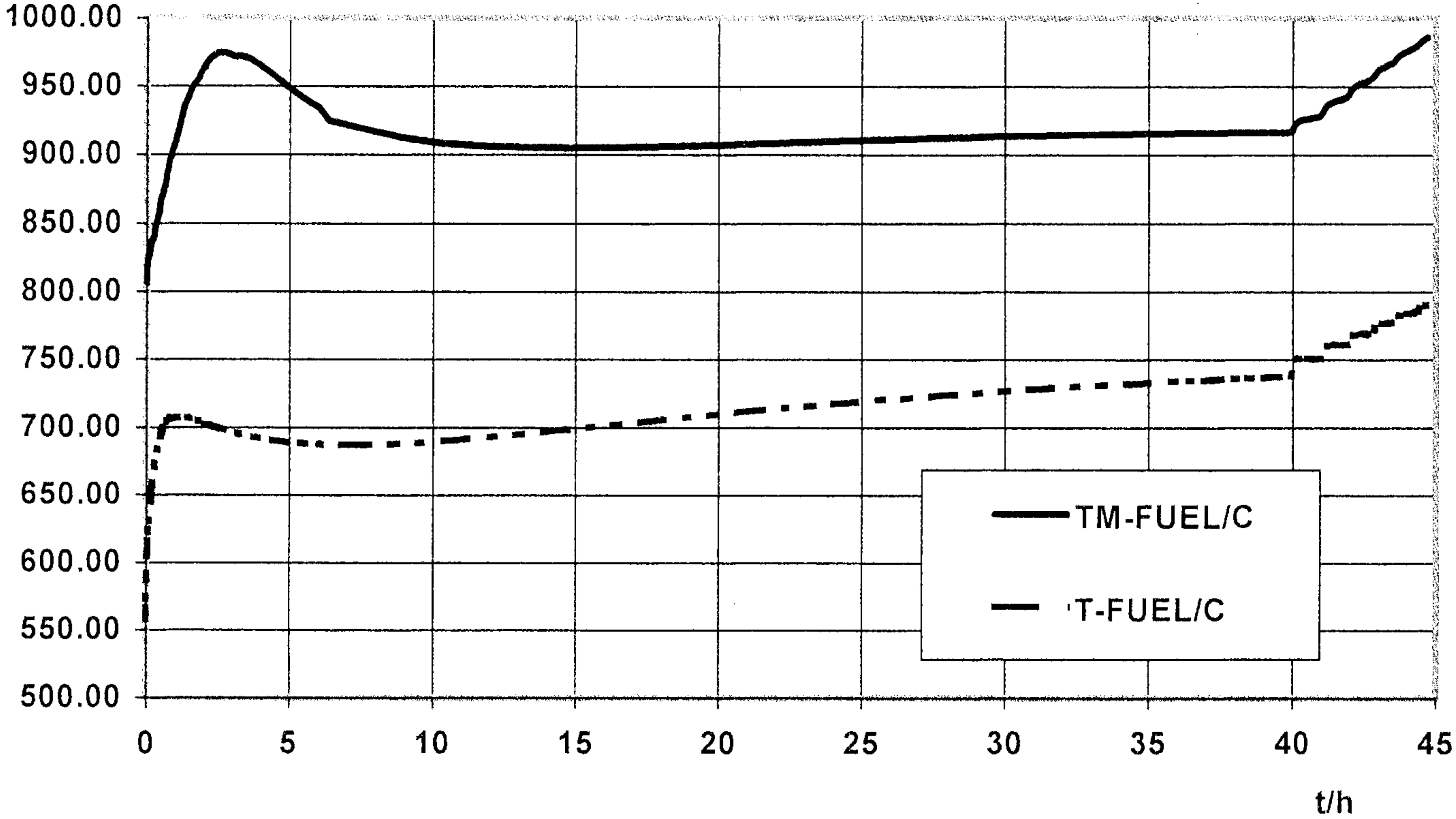
MODUL-200

(TINTE Transients)
LOFC with Shortcut over Circulator



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AVR GmbH
Arbeitsgemeinschaft Versuchsreaktor AVR GmbH
AVR = Joint Working Group Experimental Reactor

5 July 2001

Short Notice

AVR Operational Experience, Overview

Edgar Wahlen, AVR
Peter Pohl, AVR

Executive Statements Summary

1 History

1956 Engagement of AVR in HTGR from the very beginning

1961 Begin plant construction

1964 First core ordered (UCC)

1966 First core delivered, First criticality

2 Overall achievements

2.1 The plant has been operated for 21 years (1967 – 1988). Given the experimental and first-of-its-kind character of the plant, the achieved time availability with a record value of 92 % in 1976 is quite remarkable (Fig. 1).

2.2 Starting in 1974, operation at 950 °C had a share of nearly 30 %.

2.3 The personnel dose uptake records show significant improvements in the course of the years due to better components and procedures (lessons learned, Fig. 2).

2.4 Radioactivity release to the atmosphere remained well below licenced levels (Fig. 3).

3 Fuel

3.1 AVR was the indispensable mass test facility for all development steps of pebble fuel (Table 1).

3.2 Pebbles with oxide fuel, and no matter if HEU or LEU, BISO or TRISO coatings, showed at max. fuel temperatures of > 1300 °C and burnups of partly > 20 % fima excellent fission product retention (Table 2).

3.3 In the modern UO_2, TRISO pebble the fission product release is practically exclusively determined by the little as-manufactured free uranium outside of the coatings.

4 Pebble cycling

During plant operation, 2.4 million pebbles were cycled. The cycling system as a first of its kind needed frequent maintenance but worked well after various improvements, and accounted for only 3 % of the generator non-availability (Fig. 4).

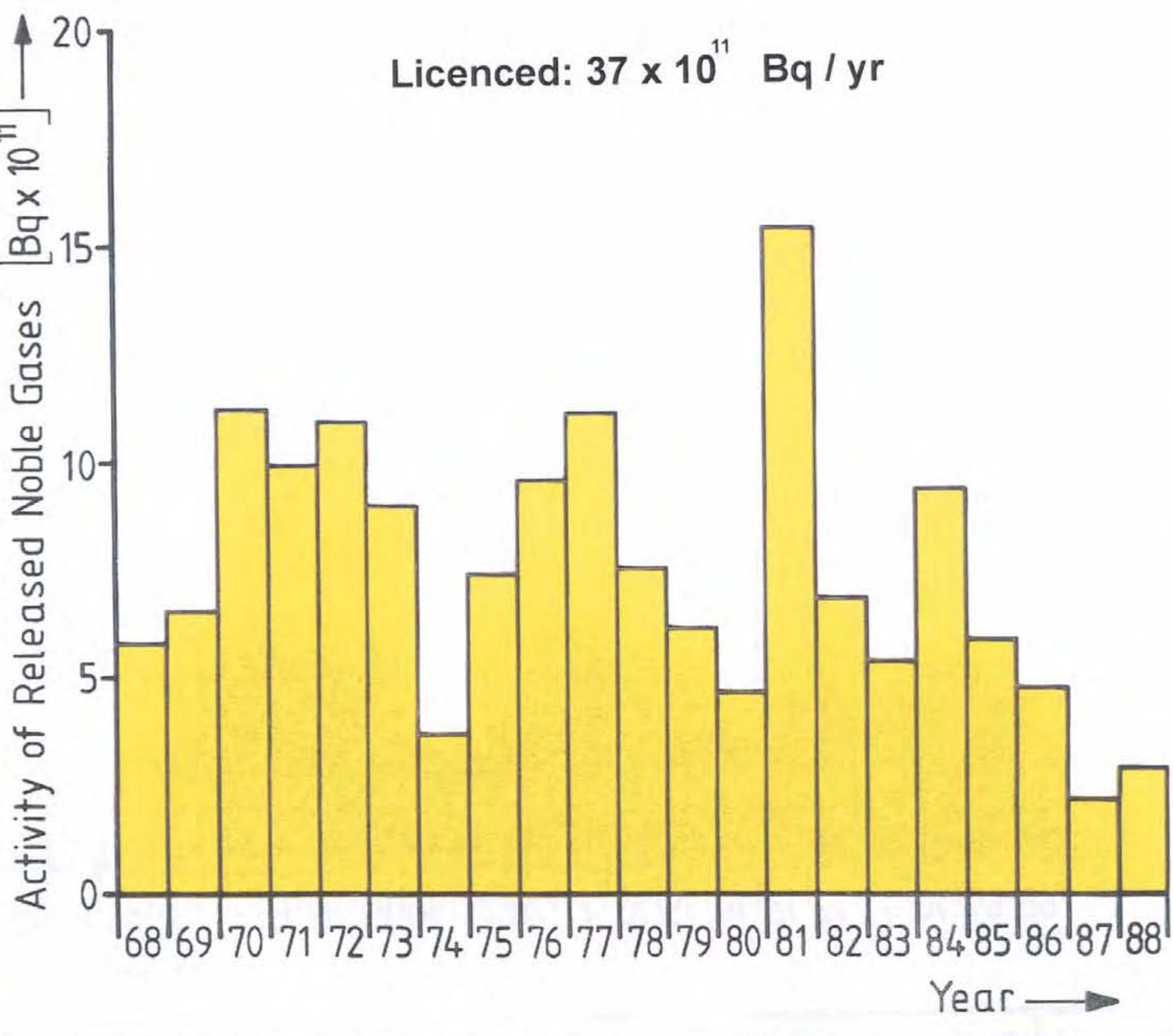
5 Water ingress

A water ingress in 1978 due to a steam generator leak did in the end not affect continued plant operation, and there was no need to replace fuel.

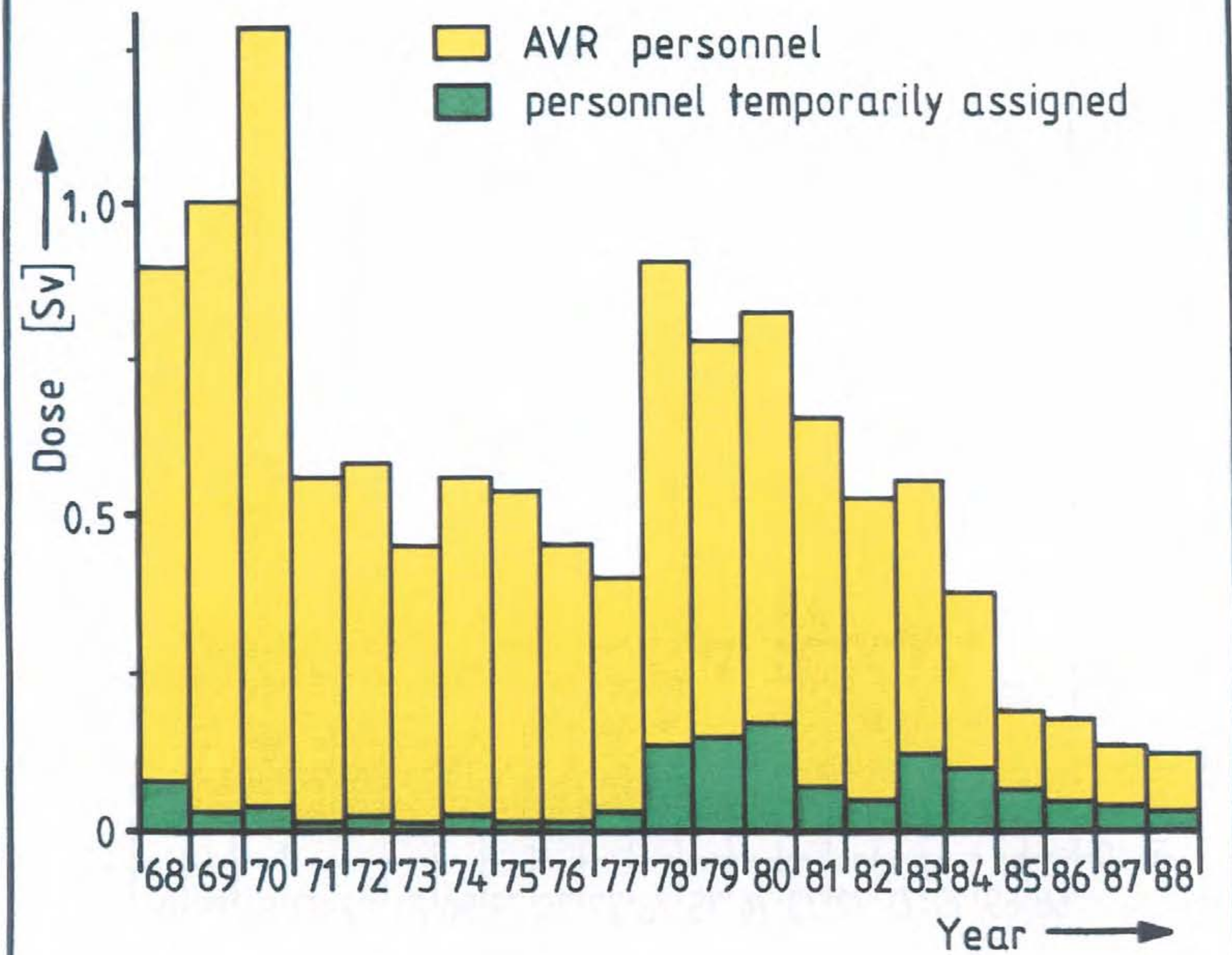
6 Safety demonstrations

6.1 Experiments simultaneously simulating the loss of forced cooling and stuck rods resulted in a simple shut-down and, with the rods kept withdrawn, in a recriticality with the reactor stabilizing at a very low core power (Fig. 5).

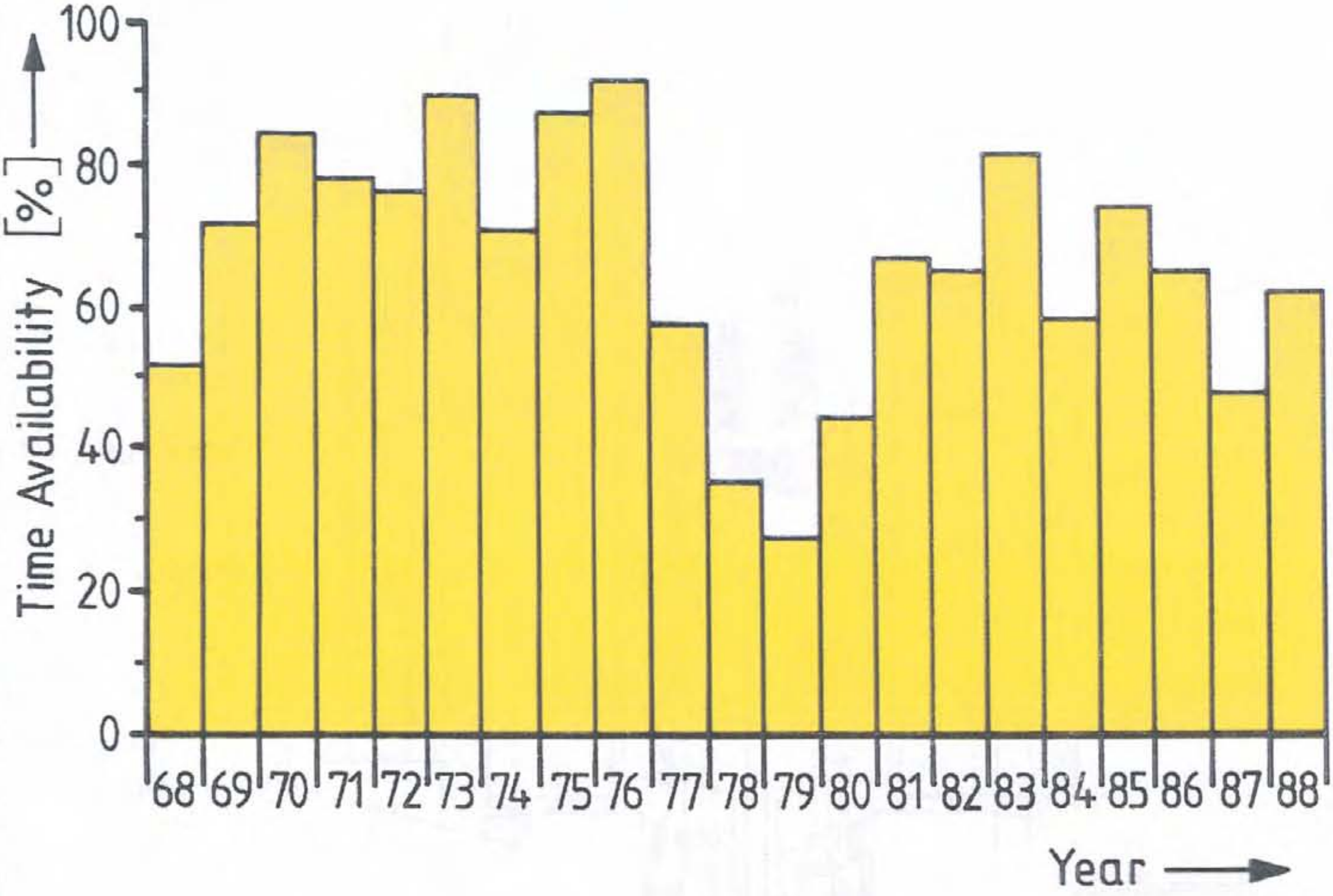
6.2 A complete loss-of-coolant accident was realistically simulated with the AVR at depressurized conditions (Fig. 6). A maximum temp. of 1090 °C occurred in the core center in less than 10 hrs after accident initiation.



AVR | Release of Radioactive Noble Gases to Environment over the Years 1968-'88 | Fig. 3



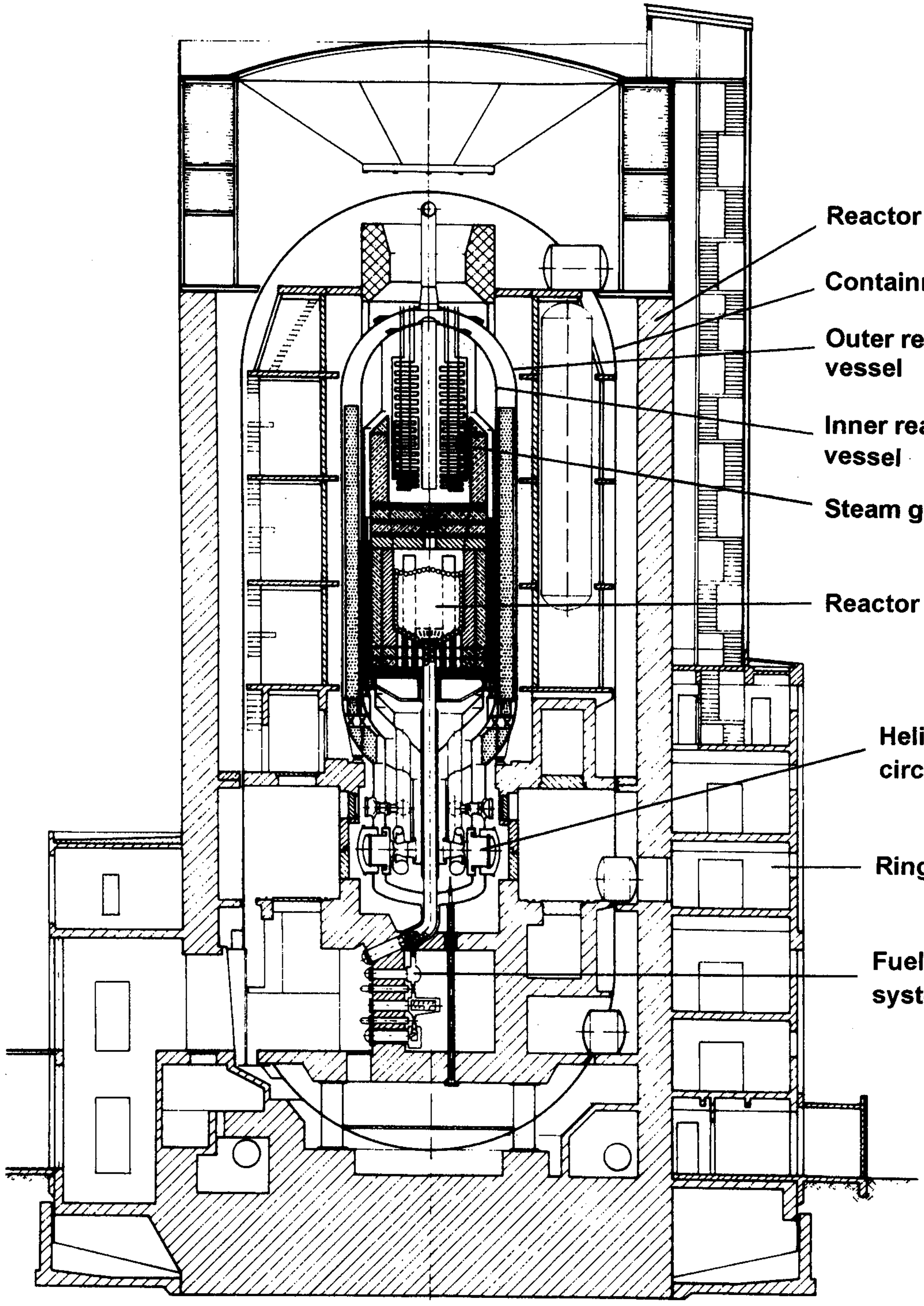
AVR | Personnel Radiation Exposure Data for AVR Annual Collective Doses | Fig. 2



AVR	Time Availabilities of AVR	Fig. 1

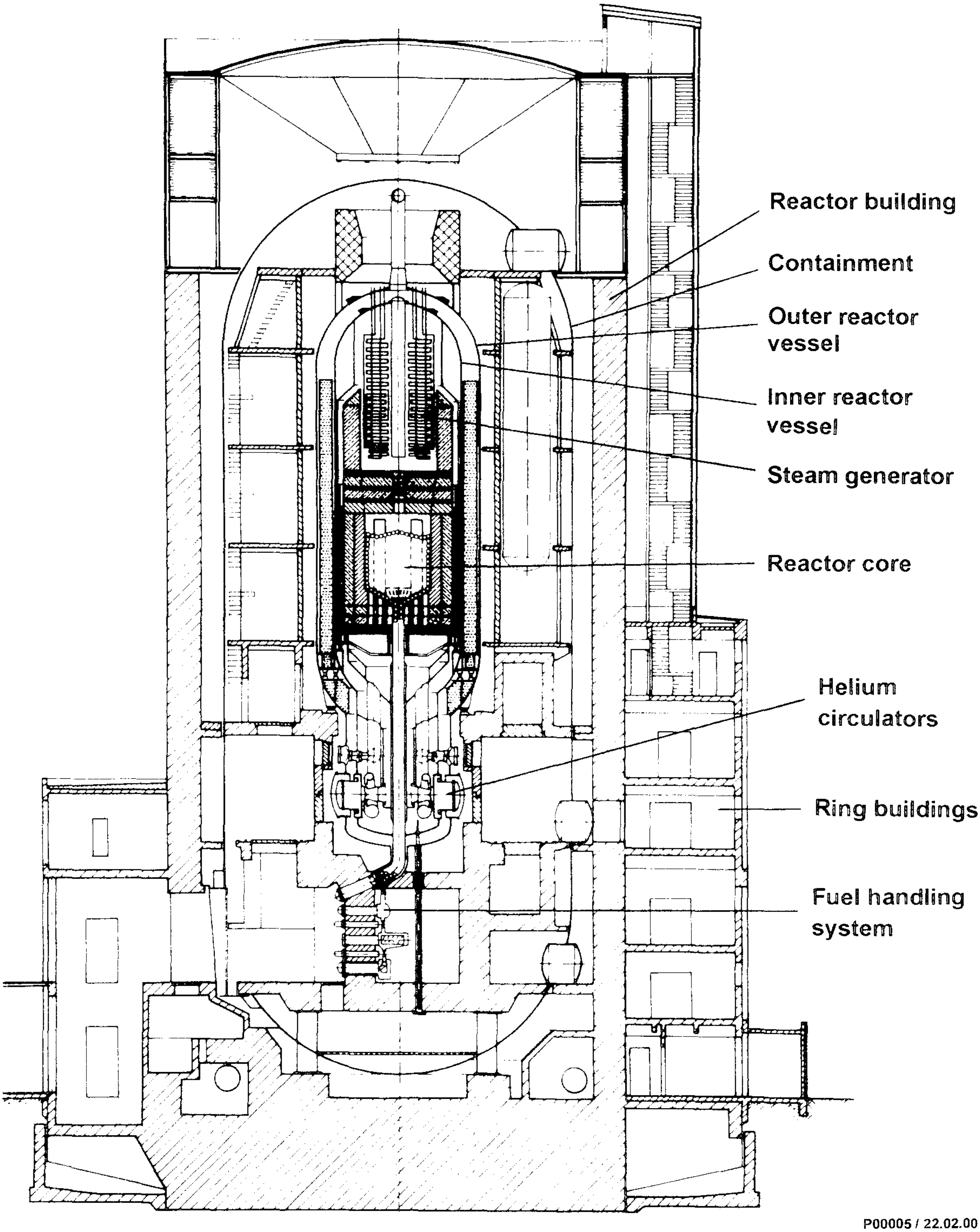
AVR

Reactor building



AVR

Reactor building



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AVR as mass test bed

Pebble structure

▪ Shell type	37,700
▪ Pressed type	253,000

Fuel design

HEU

▪ (U / Th) C_2 with 5 g Th	87,600
▪ (U / Th) O_2 with 5 or 10 g Th	129,400
▪ Feed / Breed, UO_2, UC_2, UCO, ThO_2	20,300

LEU

▪ UO_2, different enrichments	53,400

Coating design

▪ BISO type	202,900
▪ TRISO type	74,300
▪ Feed / Breed (TRISO / BISO mixed)	13,500

Stationary operation at 950 °C

Activity concentrations in Bq / m^3 and descending order

▪ Total fission gases	4.6 E 08
▪ Tritium	3.7 E 07
▪ C 14	1.9 E 07
▪ J 131	5.2 E 02
▪ Cs 137	3.0 E 02
▪ Sr 90	2.0 E 02
▪ Ag 110m	4.9 E 01
▪ Co 60	1.0 E 01

FUEL HANDLING SYSTEM

1 REDUCER

2 SINGULIZER

3 SPHERE VALVE

4 FRAGMENT SEPARATOR

5 FRAGMENT VESSEL

6 DOSING WHEEL

7 CONVEYOR

8 FUEL ELEMENT VESSEL

9 RING ROOM

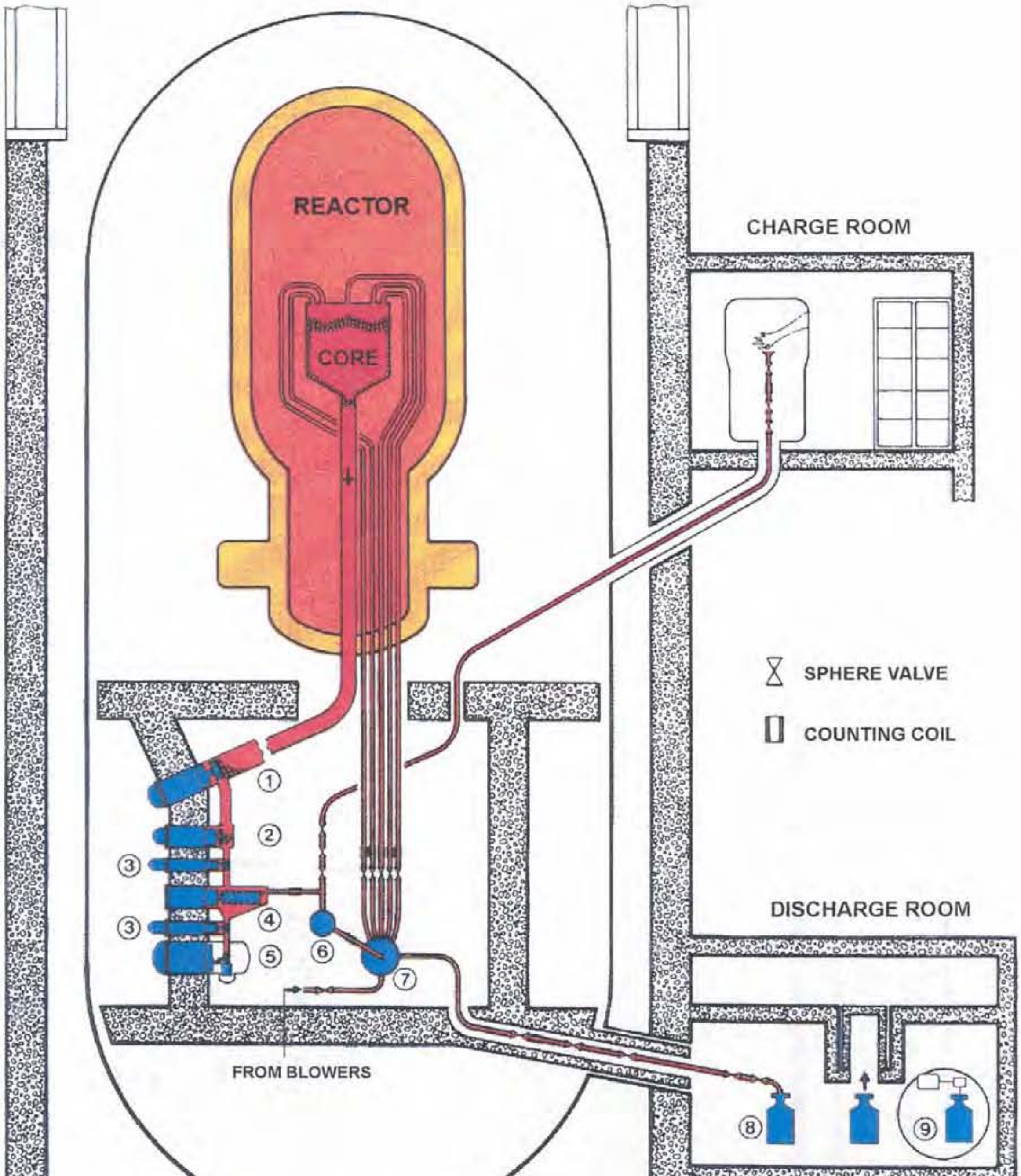
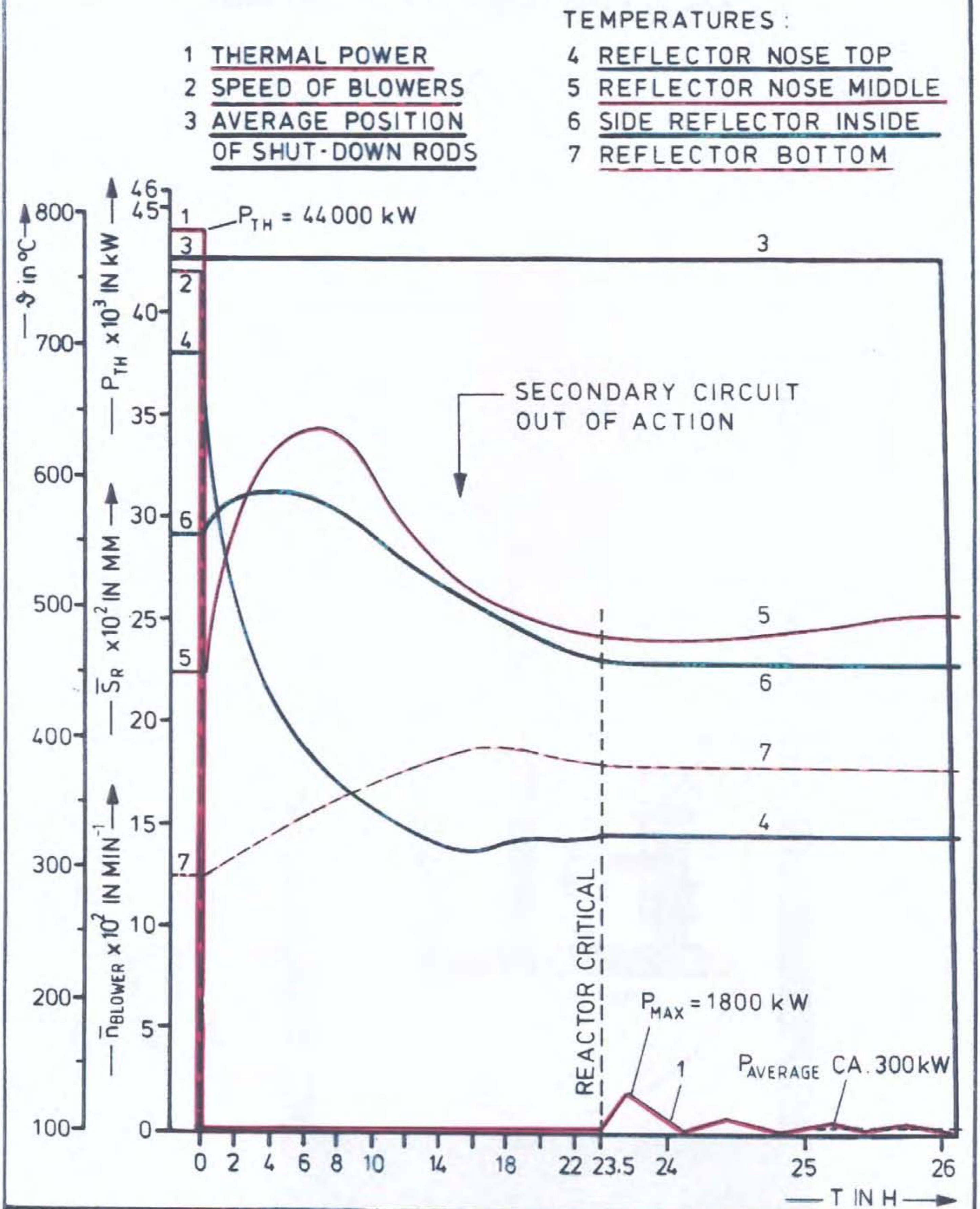


Fig. 4



AVR	SIMULATED FAILURE OF SHUT DOWN EQUIPMENT AND INTERRUPTED DECAY HEAT REMOVAL FOR THAT TIME	Fig.5

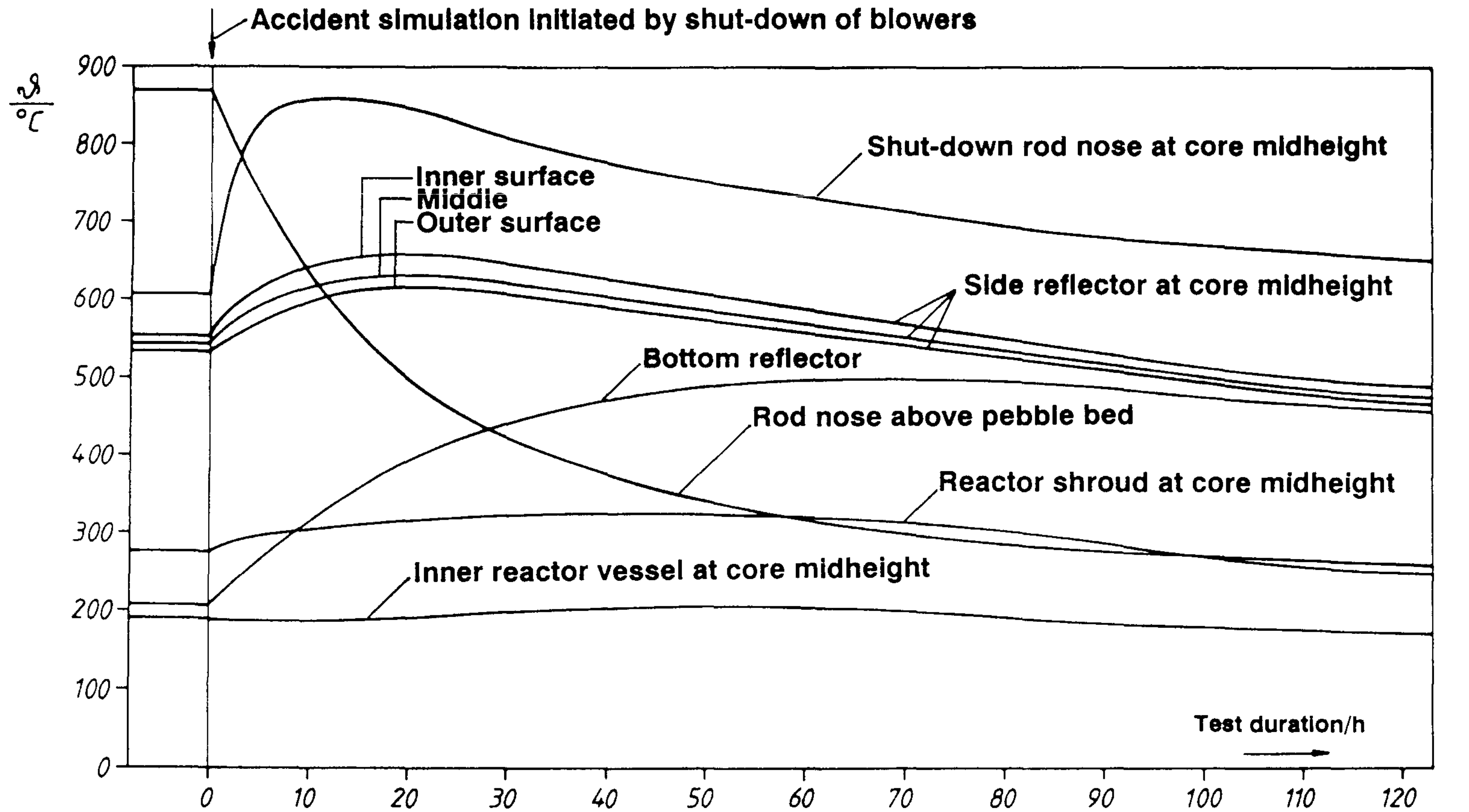


Fig. 6	Measured Temperature Curves during LOCA Simulation of October 14, 1988	AVR GmbH

AVR

Plant history

- 1956 — Engagement of AVR in HTGR, from the very beginning
- 1961 — Begin plant construction
- 1964 — First core ordered from Union Carbide
- 1966 — First criticality
- 1967 to 1988 — In operation as Experimental and Pebble Test Reactor

- Electricity production 1,670 GWh with about 300,000 pebbles.

  That means an average production of 6 MWh from each pebble.

XA9848060

DECOMMISSIONING OF THE AVR REACTOR, CONCEPT FOR THE TOTAL DISMANTLING

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AVR GmbH, Düsseldorf

U. BIRKHOLD
ISE GmbH

Germany

Abstract

After more than 21 years of operation, the 15 MWe AVR experimental nuclear power plant with pebble bed high temperature gas-cooled reactor was shut down in 1988. Safestore decommissioning began in 1994. In order to completely dismantle the plant, a concept for Continued dismantling was developed according to which the plant could be dismantled in a step-wise procedure. After each step, there is the possibility to transform the plant into a new state of safe enclosure.

The continued dismantling comprises three further steps following Safestore decommissioning:

1. Dismantling the reactor vessels with internals
2. Dismantling the containment and the auxiliary units
3. Gauging the buildings to radiation limit, release from the validity range of the AtG (Nuclear Act), and demolition of the buildings

For these steps, various technical prrocedures and concepts were developed, resulting in a reference concept in which the containment will essentially remain intact (in-situ concept). Over the top of the outer reactor vessel a disassembling area for remotely controlled tools will be erected that tightens on that vessel and can move down on the vessel according to the dismantling progress.

1 Introduction

The 15 MWe experimental nuclear power plant with helium cooled pebble-bed high temperature reactor of the Arbeitsgemeinschaft Versuchsreaktor Jülich (AVR) GmbH was one of the first nuclear power plants developed in the Federal Republic of Germany (Fig. 1). In 1987, the dismantling was decided and in 1988, the reactor was definitively shut-down after more than 21 years of operation /1/.

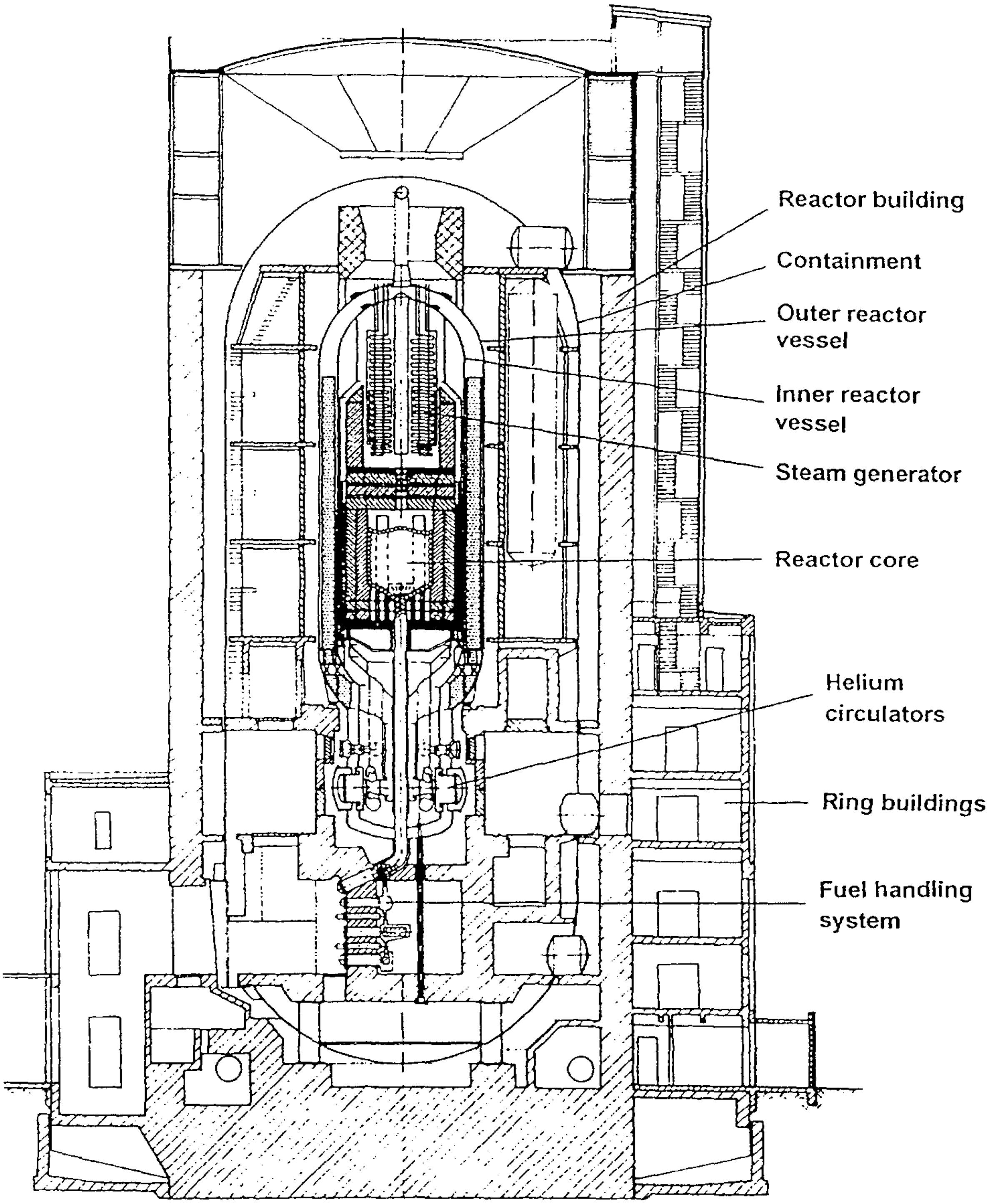

Fig. 2. AVR reactor building

Containment

Diameter	16 m
Wall thickness	12 mm
Height	41.5 m
Design pressure	3 bar

1.2 Existing systems and facilities of the reactor operation

The systems and facilities of the reactor operation, as shown in Table 1, are available to sustain the basic operational functions during the dismantling works, i. e. control, supply and disposal functions. The operation of these systems takes place in accordance with the operating instructions of the existing decommissioning manual (Stillegungshandbuch SHB).

Table 1: Existing systems and facilities for the dismantling of the AVR plant

Vent systems 1 and 2
Vent systems 1 and 2 WW
Exhaust air control systems 1 and 2
Liquid waste disposal systems 1, 2, 3 and 4
Drain pump system of vessels 21 and 22
Compressed air supply system
Power supply system
Fire water supply system
Radiation monitoring laboratory
Clean rooms
Personnel locks +5 m, +11 m and +38 m

1.3 The radiological starting position of the plant

The inventory of radiological activity has been calculated for the year 1992 and is compiled in Table 2 listing the important nuclides.

Table 2: Activity of the principal nuclides of the reactor vessels including internals

Nuclide	Steam generator	Thermal shield	Biological shield I	Reactor vessels	Ceramic internals	Primary loop	Total
Co 60	3.1E+12	1.2E+14	2.0E+09	3.1E+14	2.8E+15		3.2E+15
Sr 90	3.6E+13					1.3E+13	4.9E+13
Cs 137	2.5E+13					6.6E+11	2.6E+13
C 14					1.2E+13		1.2E+13
Tritium					1.5E+15		1.5E+15

Besides the activation products and the activated corrosion products (e.g. Co 60, Fe 55, Ni 63), there exist dust-bound fission products (Sr 90, Cs 137, Cs 134 etc.) and partly nuclear fuel fines caused by abrasion.

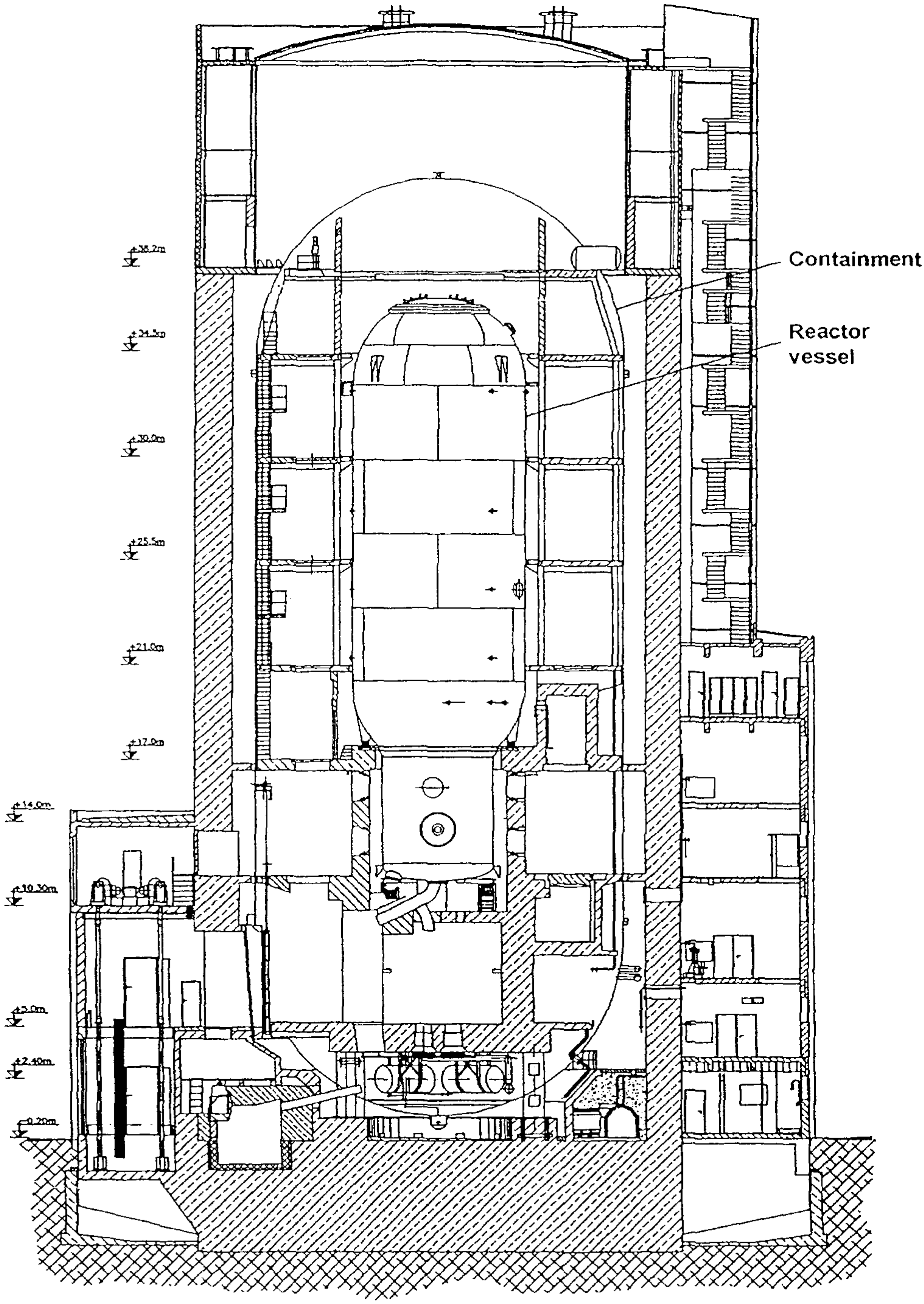

Fig. 4. Status before dismantling the reactor vessels

3 Concept and pre-engineering

For the further decommissioning project steps 1 to 3, various technical procedures and concepts were developed. These studies were aimed at showing feasible ways to dismantle the AVR plant, emphazising the dismantling of the reactor vessels with internals, and assess the costs in order of magnitude. On this basis, the AVR GmbH has requested one consortium to provide the engineering of the disposal and two consortia to provide the pre-engineering of the following concept variants:

- Concept variant 1: Dismantling of the reactor with extension of the containment
- Concept variant 2: Dismantling of the reactor without structural alterations of the containment (In-Situ Concept)

AVR requested that two conditions are to be adhered to during the pre-engineering:

- The Two-Barrier-Concept is to be maintained; i. e. the barriers containment and outer reactor vessel are to be preserved or to be adequately replaced
- The dismantling work in the inner reactor vessel is to be performed under an inert atmosphere.

3.1 Concept variant 1

The concept variant 1 was investigated by ARGE BABCOCK/STEAG-DETEC and is characterized by the following criteria /2/:

- After the dismantling of the roof, the containment will be enlarged (Erweiterter Schutzbehälter, ESB)
- Installation of a disassembling area inside the ESB
- Dismantling of the steam generator and disassembling in the disassembling area
- Dismantling and disassembling of the reactor vessels in parallel to the disassembling of the steam generator
- Use of Master-Slave manipulators

3.2 Concept variant 2

The concept variant 2 was investigated by ARGE NOELL-LENTJES and shows the following criteria:

- Installation of a disassembling area without enlargement of the containment
- Step by step lifting of the steam generator and disassembling of the tube bundle by use of a power manipulator
- Installation of a large manipulator with tools to dismantle the reactor vessels
- The dismantling of the steam generator and the reactor vessel are executed sequentially.

4.2 Technical concept

It is the essential objective to dismantle the AVR plant within the constraints provided by the regulatory body, the budget and the a.m. protective aims. In order to ensure this general requirement, the reference concept is based on the following superimposed engineering requirements which are to be adhered to in any case.

4.2.1 Engineering requirements

In-Situ-Concept

The dismantling of the reactor vessel is to be performed without any - from outside - visible alterations of the reactor building and under keeping the containment. If need be, the containment inside the reactor building may be altered. These alterations, however, may only be of insignificant nature.

Two-Barrier-Concept

The dismantling of the reactor vessels with internals is to be performed under the restraints of the Two-Barrier-Concept. I. e. during the dismantling of the reactor vessels, the two barriers

- containment and
- outer reactor vessel

are constantly to be maintained by appropriate measures in order to warrant a safe activity enclosure.

Emission of radioactive materials with exhaust air

The limiting values for the emission of radioactive materials with exhaust air in safestore decommissioning are also to be adhered to during the dismantling of the reactor.

Waste treatment

Due to the limited space available, the waste treatment and conditioning of dismantled parts are to be performed in the Hot Shop. The dismantled parts are thus to be packed into appropriate containers on location and to be transported to the Hot Shop for further treatment.

Packing

The packing of the radioactive wastes has to abide to the receiving conditions of the possible future disposal site KONRAD. The packing of radioactive materials has to abide to the receiving conditions of the neighbouring Research Centre (REBEKA facility) and external waste disposal companies (e.g. Siempelkamp).

Regulations from the safestore decommissioning licence

The design and licensing for the dismantling of components has to consider the relevant clauses and regulations of the approval for safestore decommissioning /2/.

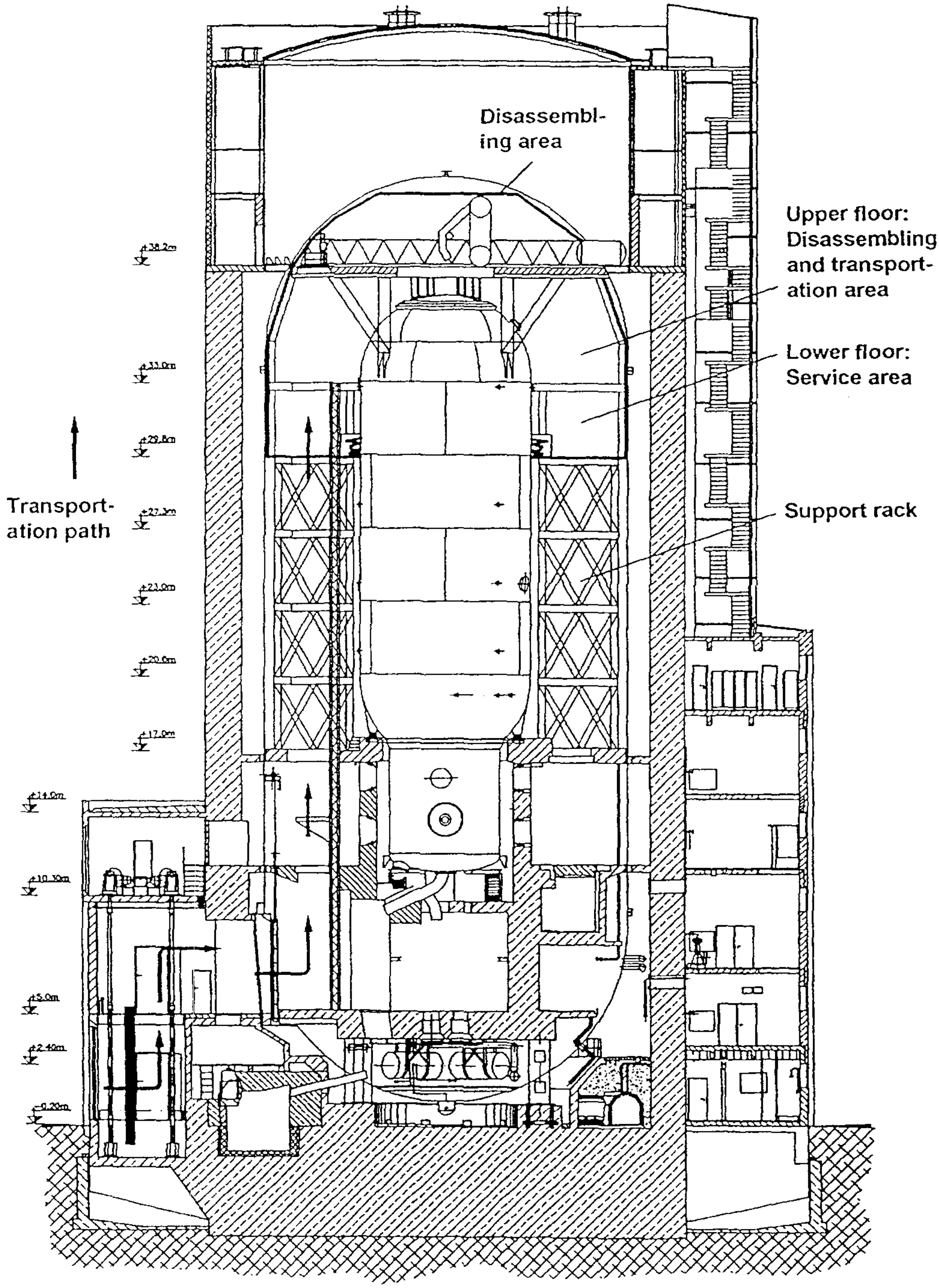

Fig. 5. Installation of disassembling area

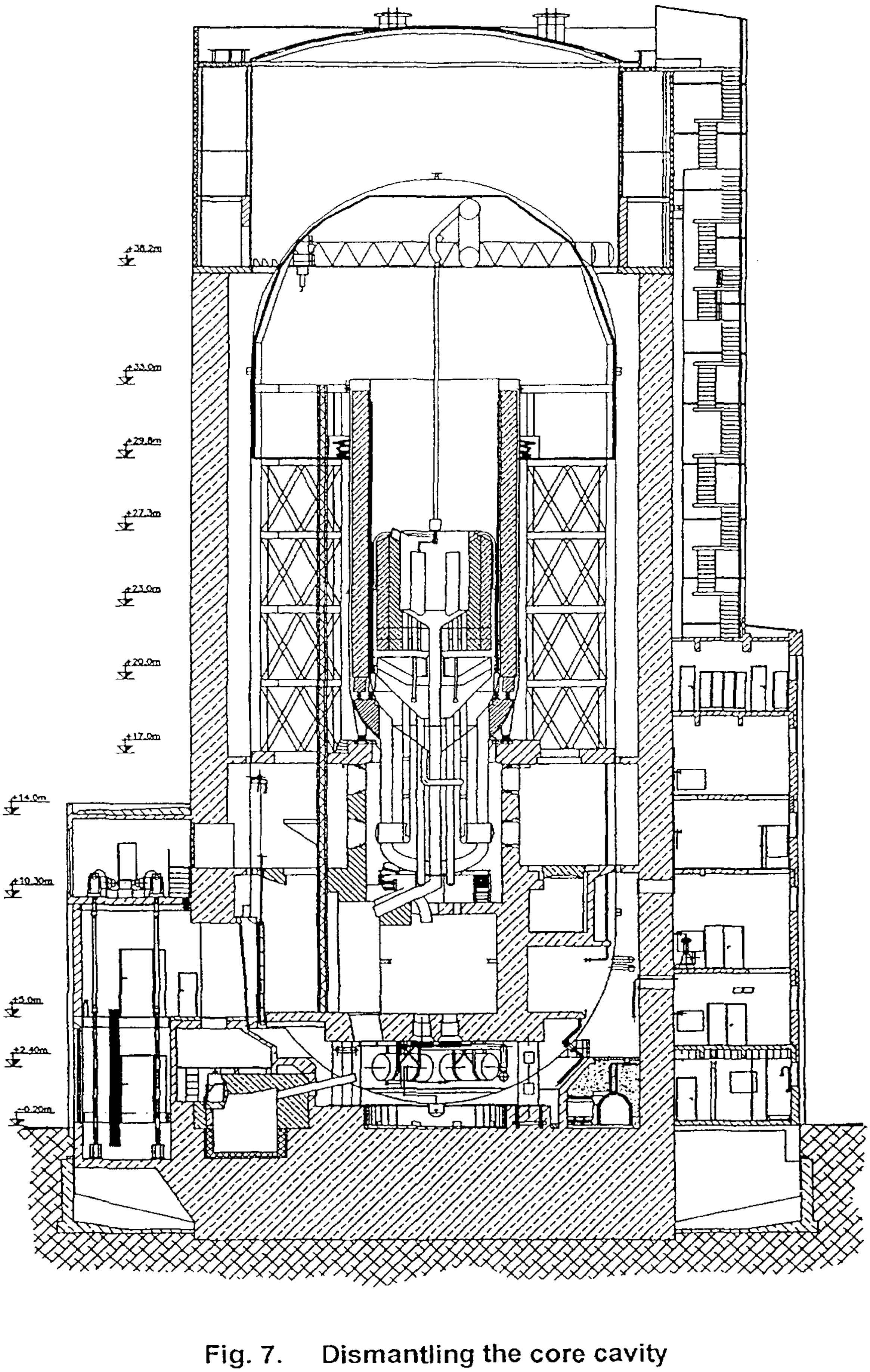

Fig. 7. Dismantling the core cavity

For the dismantling of the steam generator and the reactor vessels, the following facilities shall be installed on the upper level:

- Electric powered Master-Slave-Manipulators (EMSM), carrying capacity approx. 100 kg
- Power manipulator, carrying capacity approx. 100 kg
- Auxiliary manipulator (power manipulator)
- Manipulator support system and bearings for parts up to approx. 1000 kg
- Lifting magnet for dismantled parts
- Disassembling facilities with guiding and mounting devices for milling and sawing tools used to disassemble the vessel walls and the thermal shield.
- Devices to dismantle the steam generator tubes and the fuel discharge tube
- Facility to transport the dismantled parts between manipulator and polar crane
- Auxiliary disassembling area

The lower level contains the service area with a lock area for removed materials, tooling machines, tools, supplementary means and persons during interventions and a measuring area for removed materials. Transportable shielding for interventions is stored at appropriate locations in the disassembling area.

4.3.3 Dismantling of the steam generator (Fig. 6)

The disassembling of the steam generator is to be performed in the mounted stage, whereas the load transfer continues to take place via the bracing tubes and the inner reactor vessel lid. For the disassembling, the central opening for the displacement tube of the steam generator will be enlarged only to the size necessary to bring in the EMSM. Cutting tools used shall be a hydraulic cutter for the vertically oriented steam generator tubes and a double disk saw for the horizontally oriented steam generator tubes. The cutting of the steam generator progresses from top to bottom collecting the cut-off tube pieces in a transportable bin.

The disassembly of the tightly coiled steam generator tubes, equipped with spacers and bracing tube fixations will have to be demonstrated on a model during the 'design and licensing phase'.

4.3.4 Dismantling of the reactor vessels with internals (Fig. 7, 8)

The dismantling concept provides to dismantle the reactor vessels with internals successively from top to bottom. In order to dismantle the ceramic internals the upper vessel domes will only be opened as much as necessary to bring in the manipulators.

The four time lowering of the disassembling area by approx. 3 m each time permits the disassembling of the cylindrical reactor vessel walls including the biological shield 1 moving from the outside to the inside simultaneously to the disassembling and dismounting work inside the inner reactor vessel. This permits a flexible way to proceed.

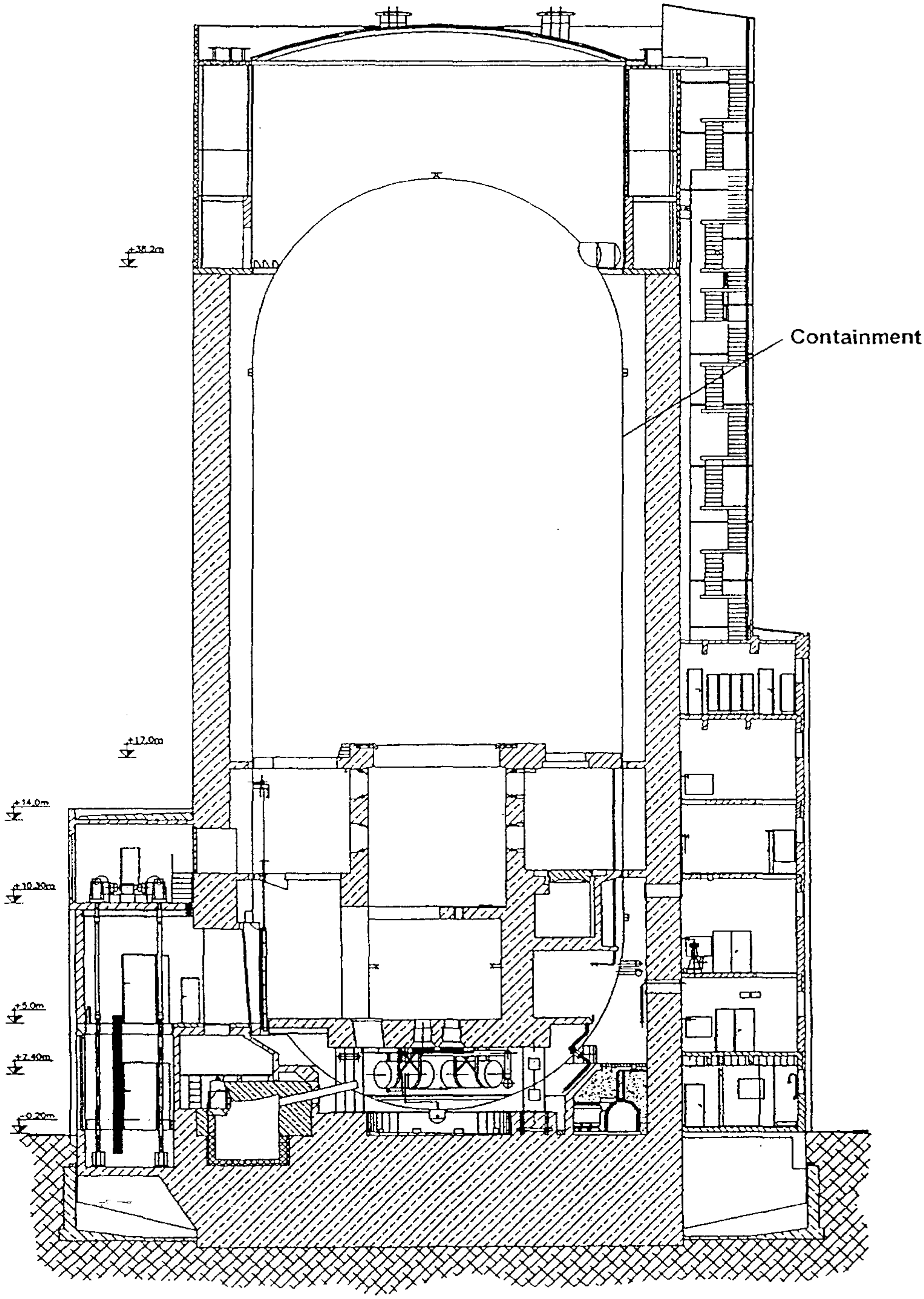

Fig. 9. Status before containment dismantling

After the plant is released from the AtG the demolition of the buildings and the disposal of recyclable materials takes place according to conventional procedures under the conventional regulatory body, e.g. Kreislaufwirtschaftsgesetz, BImSchG.

The dismantling of the AVR will end with the compilation and archiving of the safety documentation and the return of the site to green field.

4.6 Disposal concept

In order to dispose of the waste, different packing variants have been investigated in view of the final repositories ERAM and KONRAD. The calculation of the optimised packing volume resulted in a storage volume of approx. 4 700 m^3 at the KONRAD facility. Part of the drums and bins still need type rating for this purpose. Difficulties arise in particular from the restrictive KONRAD reception conditions for tritium (H-3) and carbon (C-14).

4.7 Time schedule

The time schedule is shown in Fig. 10 from a today's view point. Under the assumptions that the application for the dismantling of the reactor vessels can still take place in 1998 and the permit be granted until the end of 2000, the state of 'Green field' for the AVR plant may be accomplished in 2011.

4.8 Costs

The cost estimates which have been performed during the two pre-engineering phases resulted in approx. 250 Mio DM for the dismantling of the AVR plant. The disposal effort is supposed to be in the same order of magnitude. Thus, for the dismantling of the AVR plant a total cost of approx. 500 Mio DM is expected (without safestore decommissioning).

5 Further proceedings

In July 1997, selected bidders have been invited to tender for the service package of dismantling the AVR plant containing the continued dismantling steps 1 to 3 and partial services of the safestore decommissioning phase 2. Objective of the invitation to tender is to find a qualified general contractor for the engineering and the realisation of the total dismantling project. The closing date for bid acceptance is set for October 1997. AVR is confident that the award to perform the engineering services will take place by beginning of 1998.

During the first project phase, the final concept will be fixed by the general contractor in a modified reference concept. This will be detailed in the subsequent design phase and the documents for the licence application will be generated. The application for the dismantling of the AVR plant along with the safety analysis report (Sicherheitsbericht), the final hazards summary report (Sicherheitsbetrachtung), and the environmental compatibility report (Bericht zur Umweltverträglichkeit) shall be filed in autumn 1998. The supplementary documents (Erläuterungsberichte) shall be filed in 1999 in order to expect the granting of the licence no later than by the end of 2000.

The second project phase contains essentially the accompaniment of the licensing process and the detailing of the design including preparation of the specifications for the facilities needed to dismantle the reactor vessels.

The third project phase starts with the granting of the approval to dismantle the reactor vessels of the AVR plant and encompasses the production engineering, the preliminary inspection, the production and the procurement of facilities, the testing of the remote controlled devices and the execution of the dismantling measures. The dismantling of the reactor vessels shall in today's view be completed in 2009.

Up to 2001 the already approved dismantling tasks as well as the supplements of the 2nd safestore decommissioning phase, still subject to approval, will be executed. Based on today's time situation, the prerequisites for the dismantling of the reactor vessels with internals will be in line by approx. 2001.

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1. Introduction

The 15 MWe AVR experimental nuclear power plant is one of Germany's oldest nuclear installations; construction began in 1959. Its reactor belongs to the first generation of high temperature gas-cooled reactors (HTGRs) and was among these with its 21 years of operation certainly the most successful. For design and achievements, former publications like /1/ should be referred to. In this report on decommissioning only some key items shall shortly be recalled:

- Core of about 100,000 ball shaped fuel elements (pebble bed) cycled during reactor operation,
- Highest ever reached coolant temperature of 950 °C,
- Indispensable mass test facility for HTGR fuel development,
- First-ever-done experimental simulation of a loss-of-coolant accident /2/.

An overview of the reactor design and the site structure is given in Figures 1 and 2.

The plant was finally shut down end of 1988. A licence for Safestore decommissioning, first applied for in 1986, was granted in March 1994. Since a pebble bed reactor is never defuelled during reactor operation, defuelling is the major concern in Safestore decommissioning, and the whole task was separated in a first phase with defuelling and dismantling outside of the reactor building and a second phase with dismantling and preparations for the later dormancy period inside the reactor building.

The paper looks at the achievements obtained in now three and a half years of decommissioning activities, the future programme of Safestore decommissioning , and gives an outlook on the possible continuation of decommissioning towards the green field. The latter is presented in more detail in an own presentation within this TCM.

2. Overall Progress, Achievements, Highlights

Although defuelling is still not terminated, and the second phase of Safestore decommissioning with major dismantling in the containment could not yet start, the project has not been lacking considerable progress, summarised in the following.

- Since all obstacles and limitations concerning the transfer of the low-enriched part of the AVR fuel to the neighbouring Jülich Research Center could be finally lifted in July 1996 a major progress in defuelling has been achieved. Beginning of August 1997, only 19 % of the fuel was still left in the reactor.

- The dismantling in the turbine hall is nearly and that outside of the buildings is fully terminated.

- The cooling towers are demolished.

- The helium bottle-battery storage and helium compressors were removed from the ring buildings in Dec. 1996. This was the first dismantling inside the reactor building and belonged to the projects that AVR was allowed to advance from the second into the first (defuelling) phase of Safestore decommissioning because of the delays in defuelling.

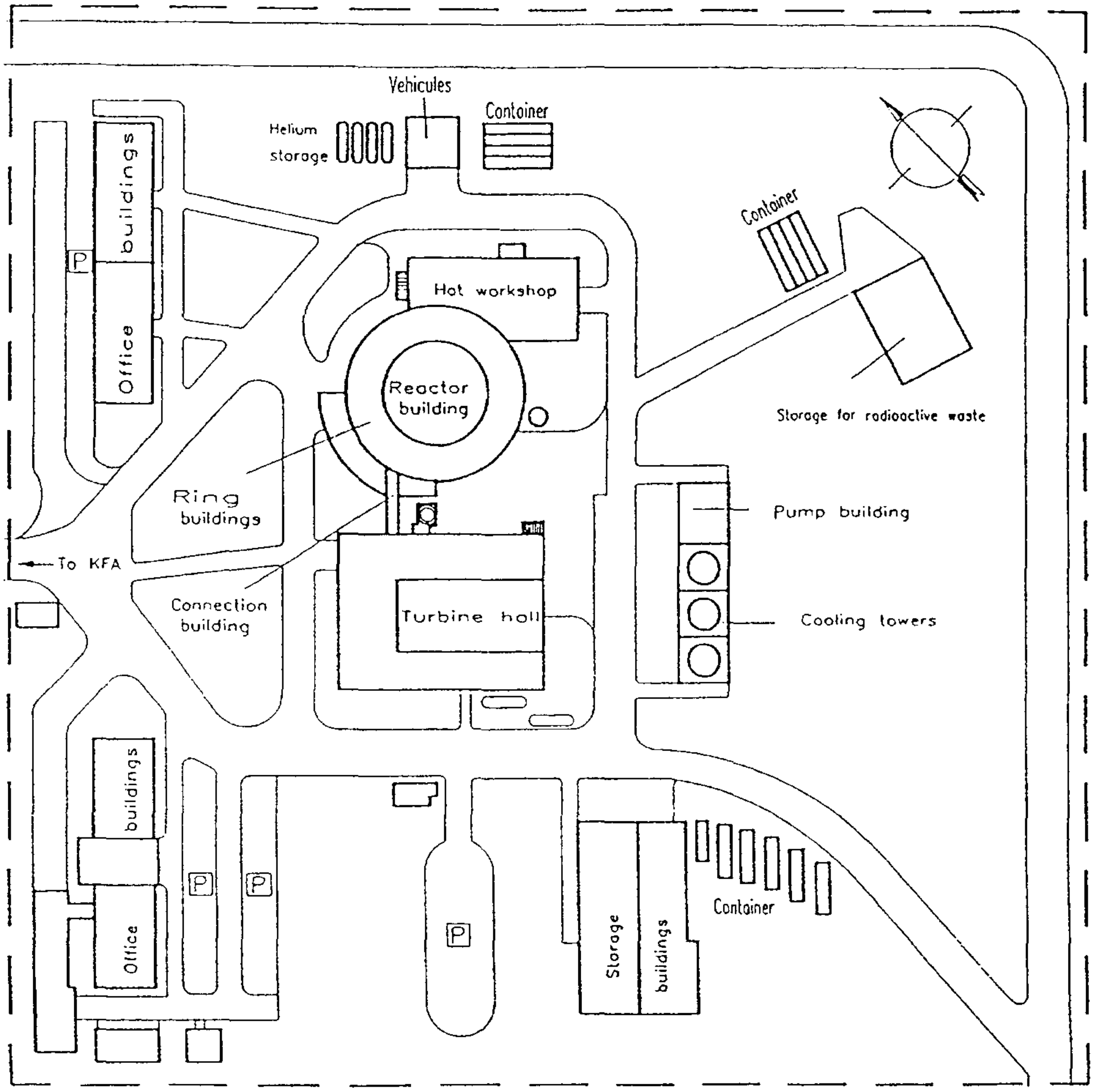

Fig. 2 Plan of AVR site

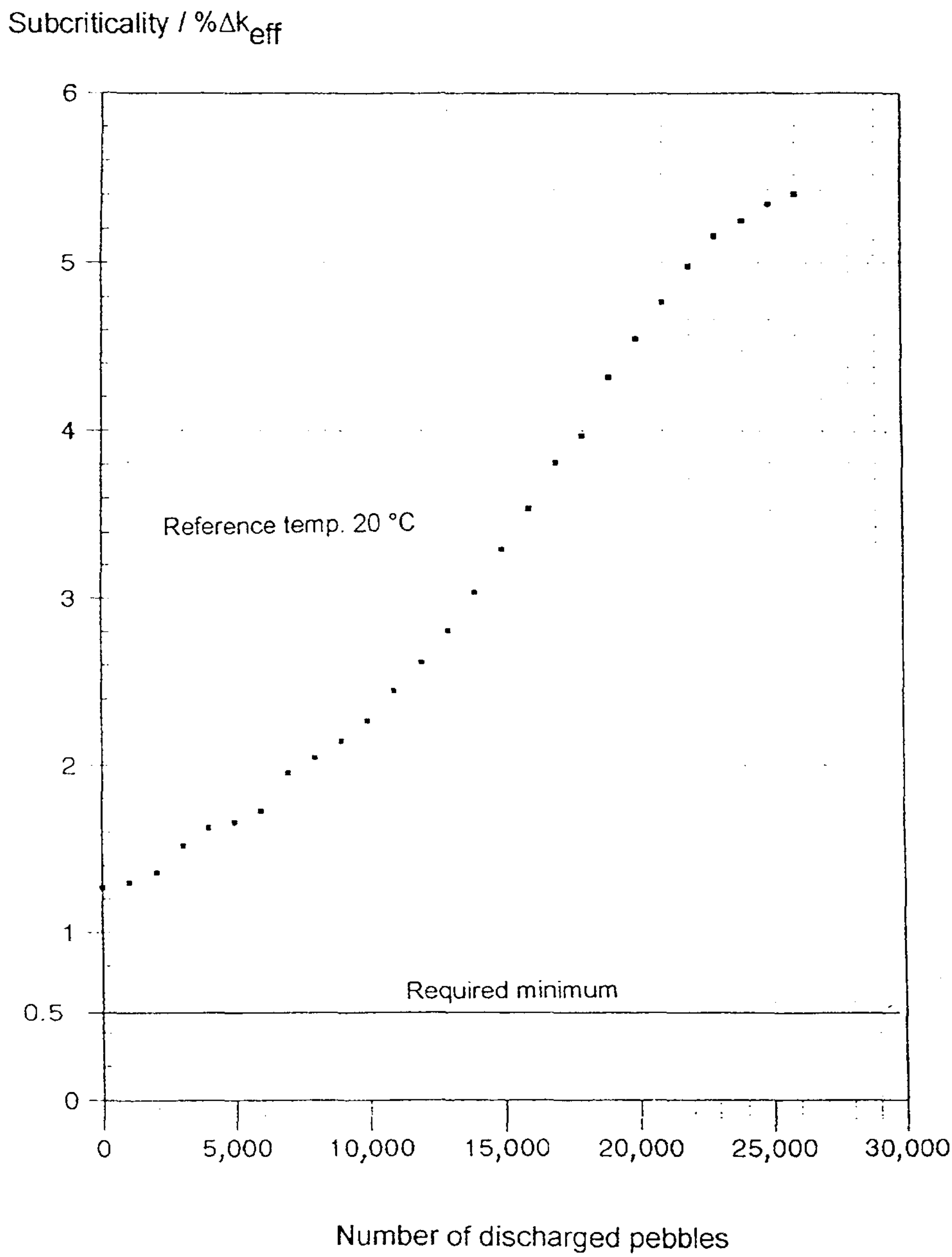

Fig. 3 Measurement of subcriticality during AVR defuelling

AVR DECOMMISSIONING
Safestore decommissioning incl. supplements and Continued dismantling

AVR GmbH

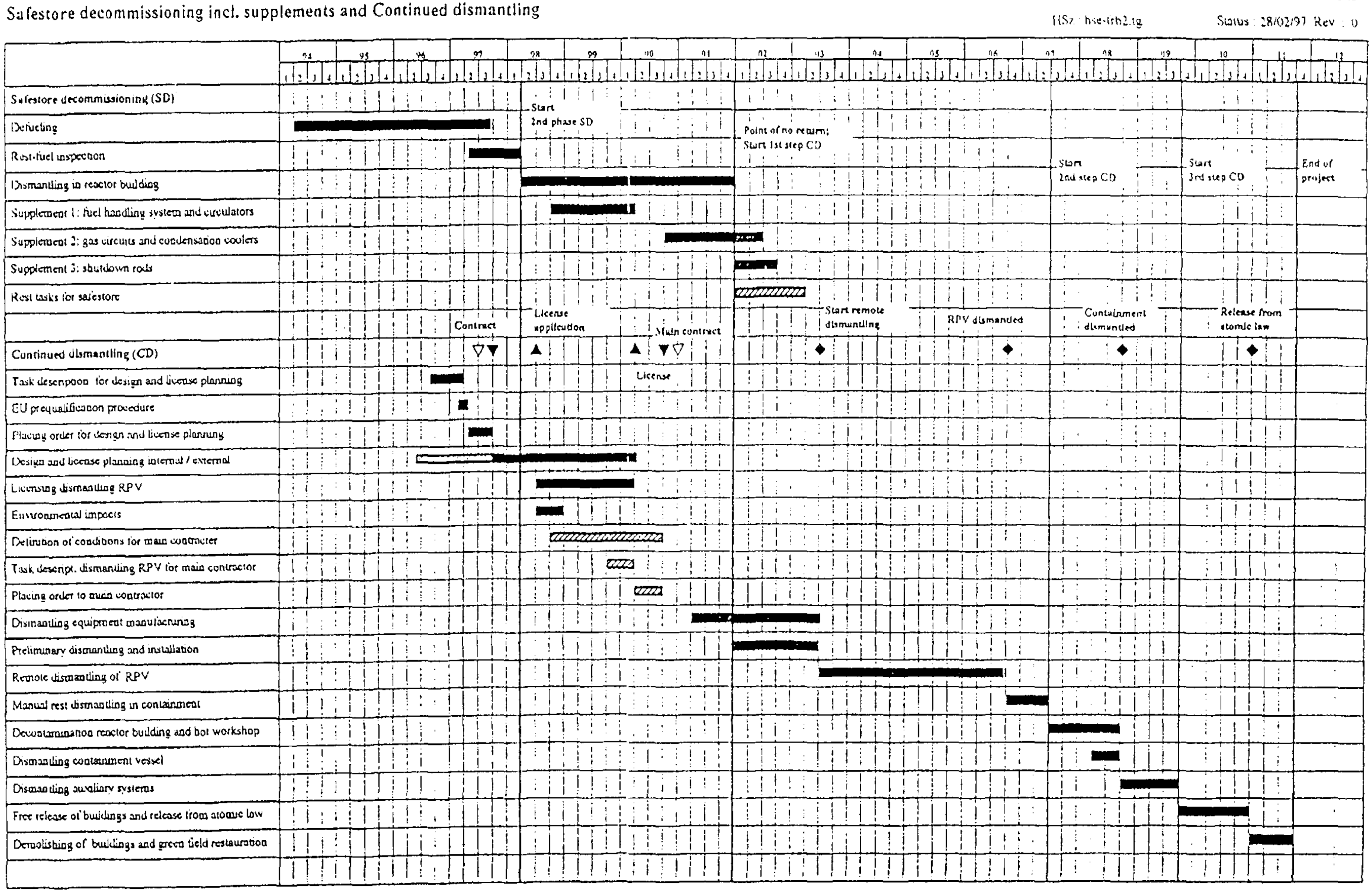

Fig. 4 Project schedule

4.2.1 First supplement

The first supplement to Safestore decommissioning, licensed in March 1997, comprises the dismantling of the

- fuel handling system,
- coolant circulators, and
- interspace convection pipe.

The **fuel handling system** can be divided into 4 main sections:

- First-components' wall: It contains vertical-in-line all the components below the fuel discharge pipe, from the reducer-wheel that closes that pipe down to the scrap collection bottle.

- Pebble distribution wall: It contains the switch wheel as well as pebble valves in all incoming and off-going pebble pipes. The 5 feeding pipes to the core will be cut and sealed below their penetration through the outer reactor vessel.

- Fresh fuel feed system: It comprises components in- and outside of the containment.

- Spent fuel discharge system: It also comprises components in - and outside of the containment, including the installations in the ring channel for buffering empty and filled fuel cans.

The **coolant circulators**, integrated into the bottom part of the 2 reactor vessels, will be dismantled including their oil lubrication system. The circulators will be removed using existing equipment for removal, shielding and transportation.

The **interspace convection pipe** with the in-built water-operated interspace cooler enabled the natural convection and cooling of the helium in the interspace between the reactor vessels at power operation. The pipe extends from the top of the outer reactor vessel, all the way down the containment wall, spreading up at the bottom, and entering the 4 shut-down rod casings at their lower ends.

4.2.2 Second supplement

The second supplement to Safestore decommissioning is in an advanced planning stadium and the licensing process is about to start. It will address the dismantling of the helium purification system and the condensation coolers.

The term **'helium purification system'** is to understand here in a wider sense since the central part of the system, the adsorption-material-containing vessels (partly deep temperature adsorption), are already covered by the original licence under the task item: removal of operational material. The wider sense comprises here all of the helium systems inside which, in a way, the purification system is central, comprising all pipework, valves (including their control systems), various compressors, a vacuum pump, vessels and filter units. The multitude of valves and many of the smaller components are grouped in a number of steel racks on nearly all floors in the containment. The goal is to remove these racks as

Reference Concept

For the remote reactor dismantling the 'in situ' concept has been chosen in which the containment vessel remains intact and the steam generator has to be cut in situ. The concept was given preference to the competing 'extension concept' in which the containment vessel would be opened at the top and largely extended to a veritable dismantling house offering enough space to pull out the steam generator as a whole unit. The key advantage of the in situ concept is that it relies on an existing and accepted boundary which should facilitate and shorten the licensing procedure to a large extend.

The in situ concept has been further detailed and fixed to a reference concept considering on top of the reactor vessels a ventilation tight dismantling area that tightens at its bottom on the cylindrical part of the outer reactor vessel. Furthermore, it has to be designed in a way that it can be moved down on the cylindrical part of the outer vessel according to the dismantling progress. Thus, a certain pre-determination of the overall method to be employed has already been made, and any detailed solution has to be based on this concept.

Design and Licence Planning

A first important step towards Continued dismantling was the decision in 1996 to award a contract for the design and licence planning. The budget for this task has been secured (about 8 mill. DEM). An EU prequalification for bidding was evaluated in July 1997 and the actual bidding process is in an advanced stage. The task list for the contractor has been divided into the following items:

(1) Planning of Continued dismantling
(2) Accompaniment of the licensing process
(3) Execution of Continued dismantling
(4) Execution of a distinguished task from Safestore decommissioning (dismantling the fuel handling system)
(5) Maintenance of the remaining plant

Items (4) and (5) are bound to a transition of AVR personnel to the contractor.

5. Costs

At present, the costs situation of the AVR decommissioning project for both Safestore decommissioning and Continued dismantling can be summarised as follows:

Spent for waiting period 1989 till 1993	c. 120 mill. DEM
Spent for decommissioning from the beginning in March 1994 till the end of 1996	c. 105 mill. DEM
Estimated total for Safestore decommissioning (incl. supplements)	c. 270 mill. DEM
Estimated costs for Continued dismantling	c. 230 mill. DEM
Public funding for AVR project, as of 31 Dec. 1996	c. 670 mill. DEM

XA9848069

UNLOADING OF THE REACTOR CORE AND SPENT FUEL MANAGEMENT OF THTR 300

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Abstract

Following granting of License 7/12a on October 22, 1993 and preparatory work, unloading of the THTR pebble bed reactor core was initiated on December 7, 1993.

Achieving the state 'plant free of nuclear material' was one prerequisite for implementation of further preparatory activities to establish safe enclosure. To reach this target, it was necessary to remove approx. 670,000 operating elements (approx. 84% of which were fuel elements).

Basically, unloading of the core was implemented in the same way as removal of the operating elements during duty operation, however, process engineering modifications to the charging system were required due to replacement of the primary gas helium with nitrogen and air and reduced temperature and pressure as compared to duty operation.
During unloading operation, the operating elements were sorted by means of the burn-up measuring system and were transferred into operating element containers (steel cans), 2,100 elements per container.

Insertion of absorber rods and addition of unirradiated absorber elements ensured clearly subcritical conditions at any moment during unloading of the core, which was confirmed by the measured values of neutron flux density.

The residual inventory of fissile material remaining in the reactor pressure vessel after completion of core unloading activities by December 1994 is 0.976 kg and is thus significantly lower than the required value of 2.5 kg.

Due to the limited storage capacities of the plant, it was necessary to ship the fuel element containers simultaneously with core unloading. In a remote-controlled process, the fuel element containers were transferred from the spent fuel store to a shielded loading station, loaded into one transport and storage cask of the CASTOR THTR/AVR-type each, which was then sealed with the primary lid. Following leak testing and definitive sealing by staff working on a working platform outside of the loading station, the transport and storage casks were transferred to six-axle purpose-designed railway wagons and shipped to the Ahaus fuel element interim storage facility (BZA). By April 1995, a total number of approx. 620,000 fuel elements had been transported from THTR to BZA in 57 shipments, on general 6 transport and storage casks on 2 railway wagons per shipment.

Due to actual burn-up of the THTR fuel elements falling below the design values (mean burn-up per fuel element container max. 85,000 MWd/t HM) and the long cooling-down period, dose rates on the casks were very low. Neutron dose rate measurements taken on a loaded transport and storage cask showed results of < 1 µSv/h at the cask surface.

After loading the cask on the transport wagon a gamma dose rate of 1 - 2 µSv/h at the closed transport hood and of 0.5 µSv/h in a distance of 2 m from the transport wagon was measured.

2. UNLOADING OF THE REACTOR CORE

The reactor core of the THTR 300 consists of a loose bed of spherical elements. At the beginning of unloading operation, the core contained approx. 563,000 fuel elements, 76,000 graphite elements and 31,000 absorber elements. These so-called 'operating elements' are spherical elements with a diameter of 60 mm and consist exclusively or in the main of graphite. Unirradiated fuel elements of the THTR contain approx. 1 g of highly enriched uranium (93% U 235) and approx. 10 g of thorium; the absorber elements and graphite elements used do not contain fuel.

Figure 2 shows diagrammatically the charging system. During duty operation of the plant (September 1985 to September 1988), it was used for continuous charging of the reactor with fuel elements. During this period, the fuel elements were recirculated several times and damaged elements sorted out by the damaged spheres separator.

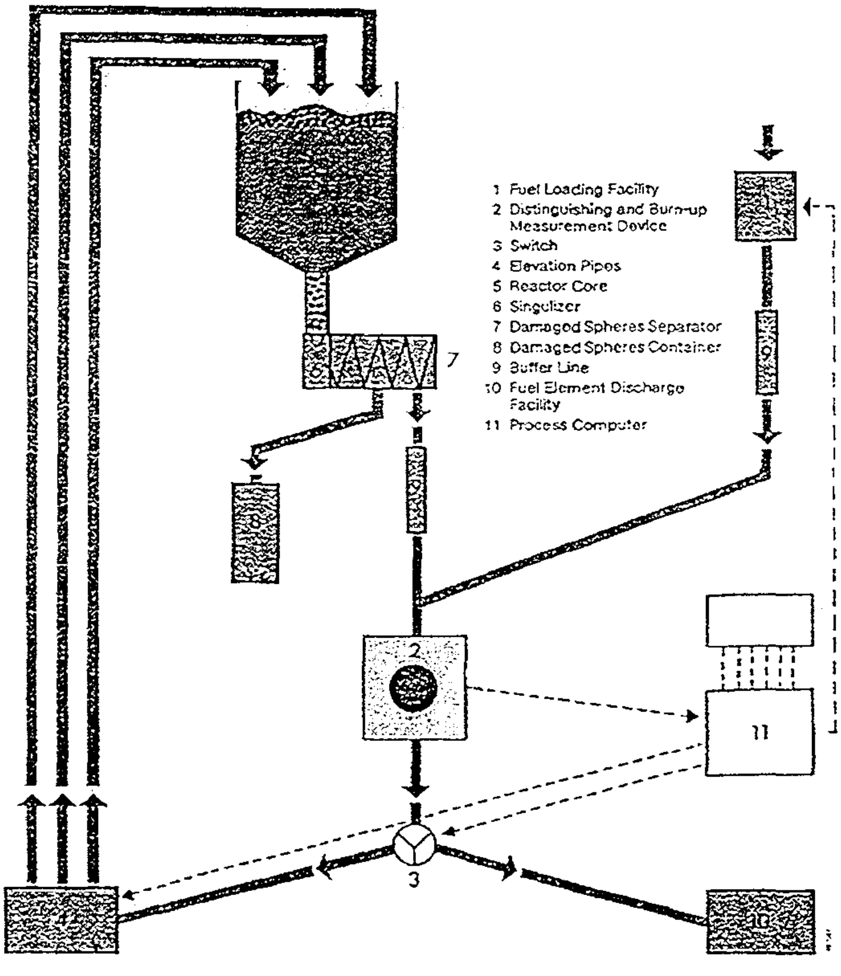


FIG. 2. THTR fuel circulating system

Basically, unloading of the core was implemented in the same way as removal of the operating elements during duty operation; however, process engineering modifications to the charging system were required due to replacement of the primary gas helium with nitrogen and air and reduced temperature and pressure as compared to duty operation.

Fully inserted absorber rods and addition of a total of approx. 4,200 unirradiated absorber elements at certain unloading steps ensured clearly subcritical conditions at any moment during unloading of the core, which was confirmed by the measured values of neutron flux density.

The development of neutron flux densities during the unloading period is shown in Figure 4. The decrease corresponds to the radioactive decay of the neutron source (Cf 252-source). When the core surface comes closer to the position of the neutron source, the decrease accelerates due to influences of geometry. Finally, only the neutron flux density caused directly by the source remains.

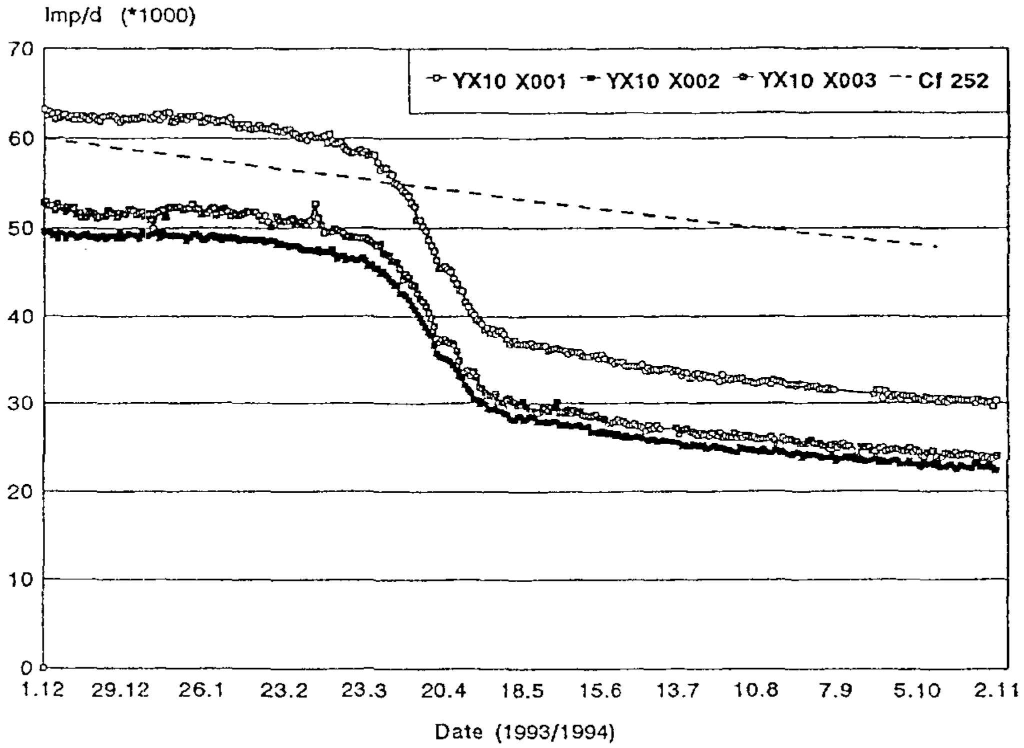


FIG. 4. Neutron flux measurement with three detectors of the YX10 system. (as reference: decay of Cf 252, normalized by arbitration)

Images supplied by a video camera that had been brought into the reactor core from time to time showed that the gradient of the funnel during reactor core discharge was within expectations.

During the final inspection, some operating elements were removed from the lower part of the operating element discharge tube and pushed into the containers provided for damaged fuel elements. Altogether 14 containers for damaged operating elements were filled during the period from the start of operation of the plant until the end of unloading operation.

After the first amendment to license 7/12a had been issued on February 2, 1995, fuel elements that might have been filled into 20 containers during the first year of reactor operation (1985/1986), containing possibly a mix of different types of operating elements, were sorted out and filled into the containers with damaged fuel elements.

Due to HKG's very tight schedule for decommissioning, processing of the transport and storage casks was implemented from the beginning in a multiple-shift operation. Through introduction of 3-shift operation and 6 days working week and through additional optimizing measures during transport and cask handling, a weekly processing rate of max. 11 CASTOR casks was reached.

By April 1995, a total number of approx. 620,000 spent fuel elements had been transported in 305 CASTOR casks from THTR to BZA in 57 shipments, usually six transport and storage casks on 2 railway wagons per shipment.

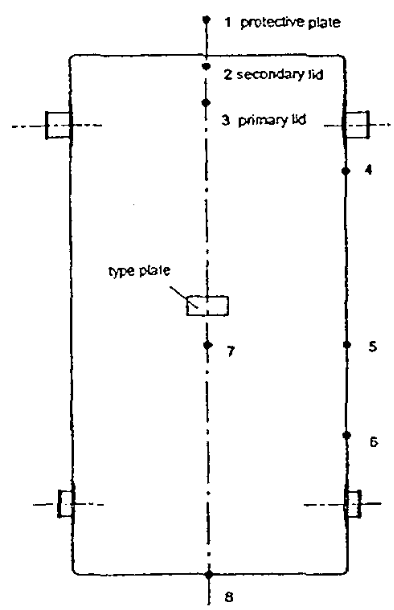
3.2 Exposure of the operating personnel to radiation during cask processing

According to the originally planned burnup and cooling time of the irradiated THTR fuel elements to be stored in the casks (mean/max. burnup 11.4% / 15% fima, 200 days minimum cooling time) a surface dose rate of max. 100 μSv/h (from gamma and neutron radiation) at a 37 cm shielding thickness of the cask material GGG-40 cm had been established in the supply specification.

Due to the real burnup history of the irradiated THTR fuel elements (reduced burnup and longer cooling time prior to storage in the transport and storage casks; max. burnup per fuel element container was approx. 8.8 % fima or 85,000 MW·d/t HM), the dose rate was reduced by about one decimal exponent to below 10 μSv/h. At a measured maximum surface dose rate of a loaded unshielded fuel element container of 10,000 mSv/h, this results in a weakening of the radioactive radiation by a factor of approx. 10^6.

With max. 100 W, the decay heat of the charged fuel element containers also was significantly lower than the design parameters.

Figure 6 shows the typical gamma dose rates measured on the loaded transport and storage cask at various points of a CASTOR cask filled with high-burnup fuel elements.



Measuring Point	Dose Rate
1	0.3 μSv/h
2	1.5 μSv/h
3*	24.0 μSv/h
(open screw holes)	(300 μSv/h)
4	0.4 μSv/h
5	1.8 μSv/h
6	0.6 μSv/h
7	2.0 μSv/h
8	3.0 μ Sv/h

* only accessible during preparation on the working platform

FIG. 6. Ambient dose rates on transport and storage casks

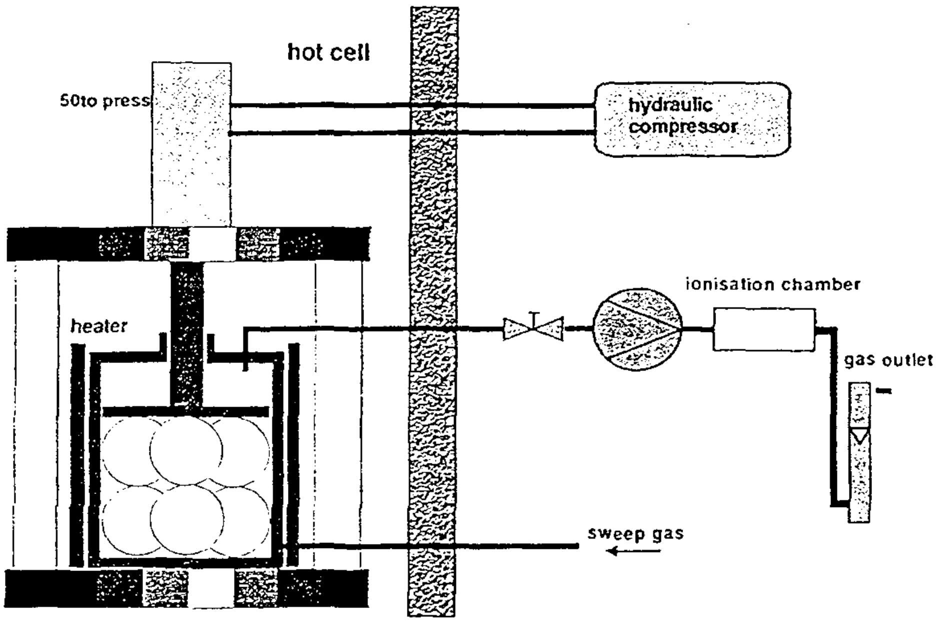


FIG. 10: 50to press to crush up to 9 spherical fuel elements (simulation of a mechanical pressure of 30 MPa)

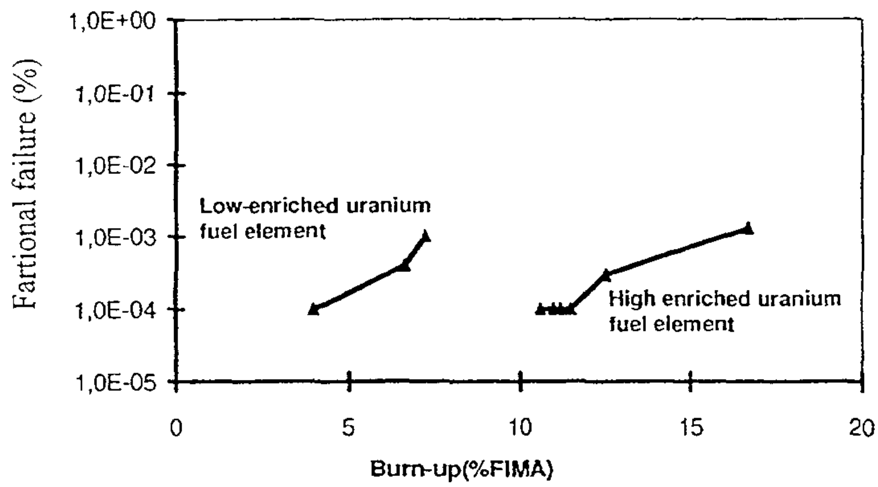


FIG. 11: Fractional failure of coated particles after crushing the matrix

failures may largely influence the long-term safety of a repository as far as the containment of long-lived radionuclides is concerned.

3.2 Behaviour under accidental conditions under the aspect of long-term safety

No radionuclide release can take place in a dry repository. Only if the radioactive waste gets into contact with the ground water radionuclides can be transported to the environment. Salt domes are geologically stable formations, which have been sealed from the ground water for more than 10^6 years. Therefore these formations are considered as ideal for final disposal.

However, in different accident scenarios it is assumed, that ground water may penetrate into the storage field through little crevices in the anhydride layers, which may be part of the salt dome. This water will form saturated, high corrosive salt brines and after corrosion of the storage casks the brines will interact with the fuel elements. A large number of experiments to study the behaviour of HTR fuel elements in such salt brines were performed at the FZ Jülich starting in the late 70th /5, 6, 7/. A short review of the obtained results is summarised in this chapter.

The radionuclide release depends not only on the diffusion of the nuclides through the graphite matrix. The basic process is the dissolution of the radionuclides in the fuel kernels, which differs for the fuel matrices UO_2 or $(Th,U)O_2$. A release from intact coated particles didn't occur, but the fuel elements contain between 10^{-4} and 10^{-5} defect particles from the production process. This rate increases by a factor of approximately ten for highly irradiated material. Therefore, the source term for radionuclide release is mainly influenced by the number of broken coated particles. To investigate the behaviour of the irradiated fuel, coated particles were collected from irradiated, electrochemically disintegrated fuel elements. The particles were carefully point-loaded until the coating cracked. The single kernels were leached with Q-brine in air at 20°C and at 90°C, respectively, and 100kPa or 13MPa, respectively. The following Fig. 13 and FIG. 14 show the release rates of different radionuclides from the two fuel matrices /7/.

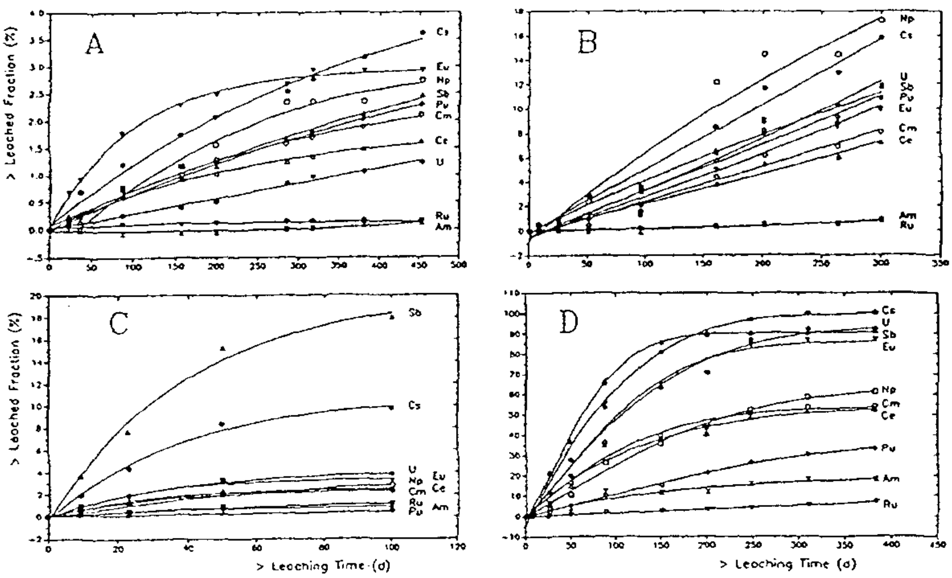


FIG. 13: Radionuclide leaching from UO_2 kernel with Q-Brine
A: 20°C/100 kPa B: 20°C/13 MPa C: 90°C/100 kPa D: 90°C/13 MPa

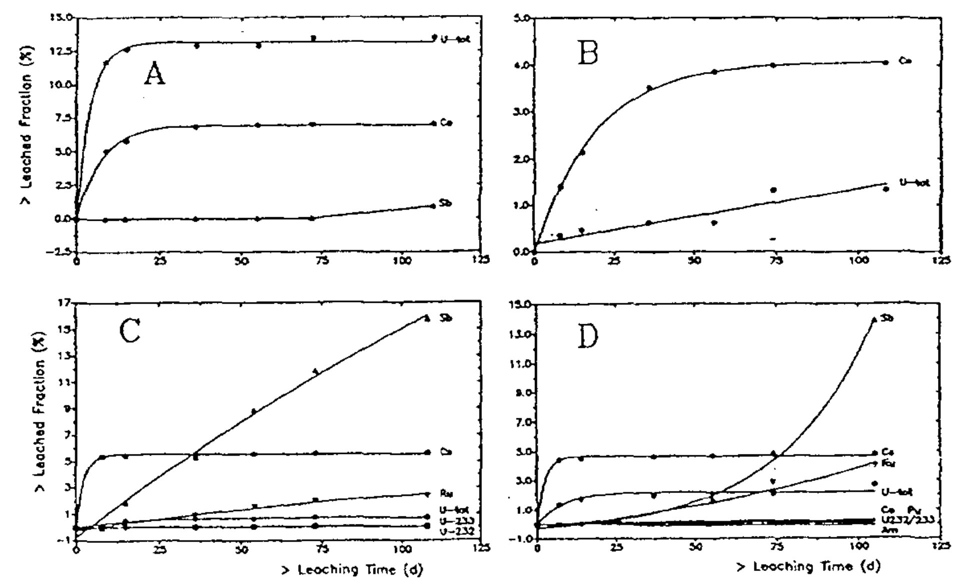


FIG. 14: Radionuclide leaching from $(Th,U)O_2$ kernel with Q-Brine
A: 20°C/100 kPa B: 20°C/13 MPa C: 90°C/100 kPa D: 90°C/13 MPa

An additional problem of final disposal is the production of hydrogen by either radiolysis of water or corrosion of metals. This hydrogen may increase the release of radionuclides to the environment by pressing contaminated brine out of the repository. A set of experiments concerning the formation of hydrogen by radiolysis has recently been finished /11, 12, 13/.

Different irradiated fuel elements were exposed to Q-brine under argon or air atmosphere in a spherical autoclave (FIG. 17). The gap between autoclave wall and fuel element had a thickness of 1 or 2 mm, respectively. The experiments were performed at 22 or 55°C brine temperature. A gas plenum with a pressure gauge for continuos measuring was located above the autoclave. Gas samples were taken with an attachable gas sampling tube to analyse the gas composition by gas chromatography and radio gas chromatography.

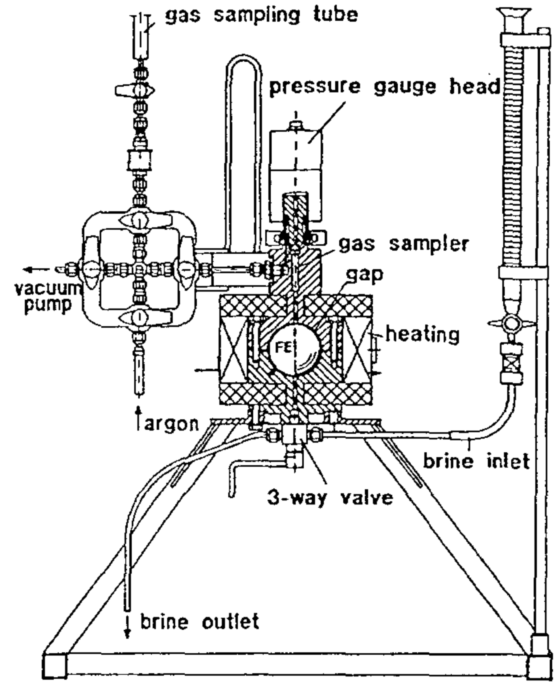


FIG. 17: Spherical autoclave

These experiments had the additional aim to determine the release of ^{14}C in gaseous form of $^{14}CO_2$ and solved in the brine. The ^{14}C is mainly formed by an (n,p) reaction of the nitrogen impurities in the cooling gas helium. The fresh ^{14}C is absorbed at the graphite matrix and due to the high temperature mounted into the crystal structure of the graphite. This radionuclide is important for the long-term safety because of its long half-life together with its different chemical behaviour in comparison to the other, mostly cationic radionuclides. Moreover, it acts as an indicator for the corrosion of the graphite matrix. Therefore brine in- and outlet were attached to the autoclave. The brine was replaced in the same time intervals as the gas samples had been taken.

The following diagram (FIG. 18) shows the pressure build-up for the different experimental conditions.

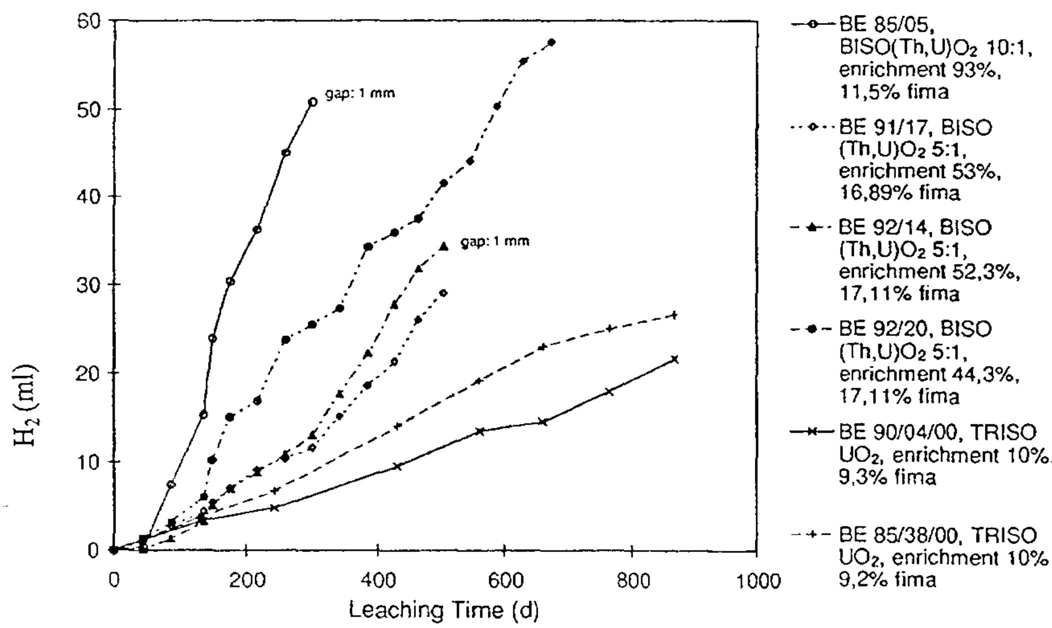


FIG. 18: Hydrogen build-up

4 Conclusions

The back-end of the fuel cycle concept for spent High-Temperature Reactor fuel elements in Germany is based upon intermediate dry storage in shielded casks in a surface facility followed by direct disposal in a deep salt repository. Depending on the disposal technique a simple conditioning may be carried out prior to disposal.

Long-term experiments have proven, that HTR fuel elements can safely be stored in dry casks. Only trace amounts of volatile or aerosol-bound radionuclides were found to be released during storage, which represent no risk for the public or the environment. Two facilities to store spent HTR fuel in dry CASTOR-type casks are being operated in Jülich and Ahaus.

Disposal concepts assume the emplacement of spent HTR fuel elements in thick-walled casks in horizontal drifts, or in thin-walled containers in boreholes. In both cases, the ceramic fuel element itself represents the main technical barrier against the long-term release of radionuclides, if the waste disposed off comes into contact with water at all. Leaching experiments have proven that only extremely low amounts of radionuclides are released from the graphite matrix. The release from the coated particles is extremely low and results mainly from defect coatings. Defects in the coating are known to be very low and depend upon type of coating and burn-up. Hence, the fuel elements are well-suited for disposal in a salt repository.

However, there are some questions left: How stable is the particle coating against mechanical and chemical impacts in the long-term run? Does a changing pressure during disposal have a major influence on the release of radionuclides from the graphite matrix? These questions should be further investigated in order to optimise the concept in terms of safety and economics.

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XA9848070

R&D ON INTERMEDIATE STORAGE AND FINAL DISPOSAL OF SPENT HTR FUEL

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Abstract

The back-end of the fuel cycle concept for spent High-Temperature Reactor fuel elements in Germany is based upon intermediate storage in shielded casks in a surface facility followed by direct disposal in a deep repository. Two storage facilities are in operation, whereas disposal in a salt dome repository is being designed. R&D results obtained so far support the chosen concept and underline the special safety features of the fuel elements, i.e. the coated particle fuel stabilised in a graphite matrix, which is extremely resistant against all conceivable attacks during storage and disposal.

1 Introduction

Based upon a former development at the Forschungszentrum Jülich (FZ Jülich, Research Centre Jülich), two high-temperature gas cooled and graphite moderated reactors (HTR) had been operated in Germany: a) the 15 MWe AVR reactor from 1967 until 1988 in Jülich, and b) the 300 MWe Thorium High Temperature Reactor (THTR 300) from 1985 until 1988 in Hamm-Uentrop. The status of their decommissioning has been reported in Session I of this conference.

Both reactors have in total produced about 1 Million of spent fuel elements during their operating time. The typical fuel element is a tennis-ball sized sphere from graphite, containing up to twenty thousand pinhead-sized fuel particles containing oxide or carbide fuel each. The particles are surrounded by a high-porosity buffer layer to limit the internal pressure from swelling and gas production, and coated with a high-density pyrocarbon layer (BISO) or with a combination of two pyrocarbon layers with a silicon carbide layer in between (TRISO) to retain radionuclides (see FIG. 1).

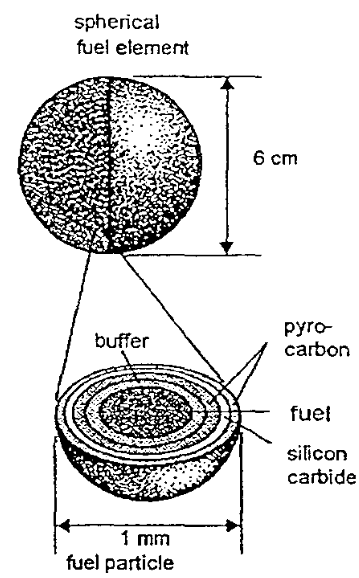

FIG. 1: Sectional view of a HTR fuel element

The spent fuel management concept for HTR in Germany is based upon intermediate storage followed by disposal in a deep rock salt repository without reprocessing.

Techniques for the intermediate dry storage in CASTOR-type transport/storage casks are available and practised with AVR fuel in Jülich as well as THTR fuel in Ahaus. Experiences are reported in Session II of this conference.

For disposal, emplacement in horizontal drifts using shielded casks, or in deep vertical boreholes using simple packaging were chosen to be the most promising concepts /1/.

This paper summarises the results obtained so far, as well as R&D still do be done to support intermediate storage and final disposal of spent HTR fuel. It supplements previous reports, e.g. /2/ through /13/.

2 R&D work on intermediate storage

Principal goals of the activities on interim storage at FZ Jülich are to demonstrate the safety of dry storage and to provide data for the licensing of corresponding commercial storage facilities. Complementing the storage of AVR fuel, a research program was initiated to measure the release of gaseous radioactivity under realistic storage conditions. FIG. 2 shows the equipment in the hot cell of

The ^{3}H inventory of a fuel element is mainly generated in the graphite matrix by ^{3}He and ^{6}Li impurities. The release is controlled by absorption and desorption processes at the grain boundaries of the graphite and diffusion in the grains. The complete ^{3}H inventory of a dry storage can amounts to about $2*10^{12}$ Bq.

The ^{14}C inventory is mainly generated by (n-p)-processes of ^{14}N impurities of the graphite. The release of ^{14}C during storage is initiated by corrosion processes of the carbon by contact with air and gamma radiation. CO_2 will be generated. The inventory of 950 fuel elements amounts up to $7*10^{9}$ Bq. Only 1% can be released, until the oxygen content of a storage can is consumed.

The ^{85}Kr inventory is generated by fission and amounts to $1*10^{13}$ Bq in a storage can. The release mainly depends on the number of defect particles. The specification of the fuel elements permits a defect rate of $3*10^{-4}$, that means a possible release of $3*10^{9}$ Bq. The extrapolation of the measurements shows a maximum amount of $1*10^{7}$Bq.

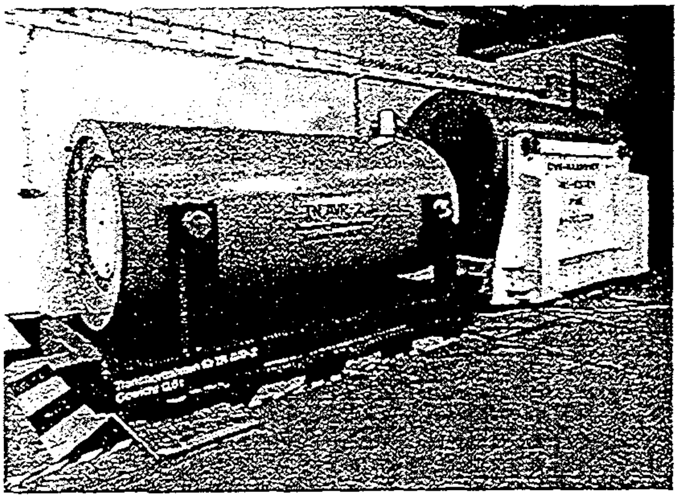

FIG. 7: 2 prototype storage casks, each loaded with 1900 fuel elements (= 2 AVR-cans)
burn-up: 12 - 16% fima

Further measurements were performed at the two prototype transport and storage casks (FIG. 7). FIG. 8 and FIG. 9 shows the distribution of gamma- and neutron dose rate at the surface. Each of the casks was loaded with two dry storage cans, filled with 950 fuel elements of the type GO and GK[b] with high enriched fuel $(U,Th)O_2$ each. The burn-up of the fuel amounted to 13 - 16 % FIMA. The measurements were done after different decay times. One year after discharge (1990) the main part of the dose rate at the surface of the 30 cm thick steel wall of the cask was generated by ^{144}Ce-^{144}Pr with the high gamma energy of 1,5 MeV. Fuel elements with lower burn-up (12%FIMA in the outer cans) produced due to the higher content of ^{235}U more short-lived fission products like ^{144}Ce. After several years of decay time, ^{137}Cs is the dominating dose rate source caused by the higher burn-up (16%

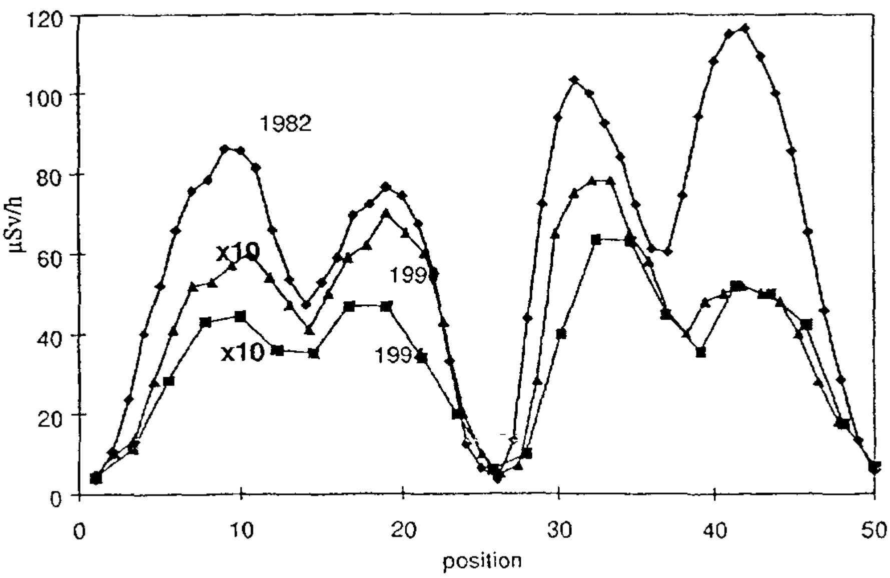

FIG. 8: Gamma dose rate of AVR storage casks

[b] compressed graphite matrix, carbide fuel

Safety Aspects of HTR Technology

Visit of the NRC – Delegation to Germany

23 to 27 July 2001

THTR operation experience, test programs, overview, highlights, lessons to be learned

Guenther Dietrich
Ivar Kalinowski

Juelich, 25 July 2001

HOCHTEMPERATUR-KERNKRAFTWERK
GESELLSCHAFT MIT BESCHRÄNKTER HAFTUNG
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THTR operation experience, test programs, overview, highlights, lessons to be learned

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Resume

- THTR 300 a prototype reactor
- THTR 300 the first step to commercialization of HTR in Germany
- Availability and outage times typical for prototype
- no problems with core
- design data reached and verified
- unavailability mainly given by refueling system
- low coolant gas radioactivity
- negative temperature coefficient verified
- airborne and waterborne release of radioactivity well below licensed limits
- low individual dose rates in operation and outages

The Situation of the THTR in October 1989

By R. Bäumer

Separate print from the English issue of

VGB Kraftwerkstechnik

Mitteilungen der VGB Technischen Vereinigung der Großkraftwerksbetreiber e.V.

Volume 70 ● No. 1 ● January 1990 ● Pages 7 to 13

The individual dose between 1987 and 1988 is represented in Figure 1. During normal operation the exposures remain below 10 mSv; for inspection and overhaul outages the values rise to 40 mSv. In this context, it must be taken into account that in the past all overhaul outages included repairs or investigations performed at the open primary circuit. At the end of 1987, flow cross-section were cut at the fuel sphere discharge system; at the end of 1988, investigations were performed at all six hot gas channels. The extremely low total individual dose absorbed during these investigations demonstrates the high retention capacity of the fuel elements and the excellent shielding effect of the prestressed concrete.

A position should also be given here on the critical remarks expressed in the media in recent months with regard to the technical condition of the plant. The operational reliability of the THTR was called into question, in other words the availability was described as being too low.

Figure 2 shows the mean annual load factor in the first years of operation for various reactor types. These figures have been taken from the nuclear engineering journal "Atomwirtschaft/Atomtechnik" and represent an average for all reactors world-wide. The availability of the gas-cooled reactors is lower than that of the water reactors. This is certainly because gas-cooled reactors have not yet seen their breakthrough on world-wide markets either. However, the availability increases as the operation of the individual plants continues. This also goes to show that all plants required further development work after commissioning.

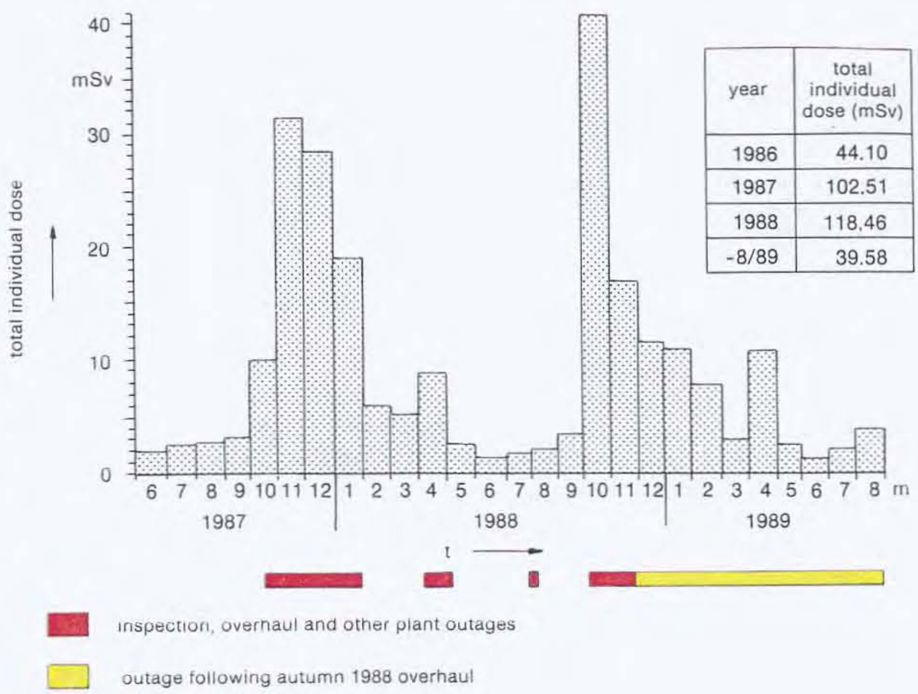


Fig. 1. Monthly accumulated total individual dose.

In this overall picture, the THTR occupies a respectable position indeed and could certainly have attained availabilities of more than 70 or 80 % if the operation had continued, especially because the problems of the fuel circulating system restricting the availability during the first years of operation could meanwhile be eliminated. The repair of the fuel discharge system should be mentioned in this context. Furthermore, the unscheduled shutdowns could be substantially reduced through optimization of limit adjustments and an improved control response.

It was furthermore maintained that the operation of the plant was substantially handicapped by damaged fuel spheres. During the past operating period, HKG could establish the connection between rigorous shutdowns with a compacted pebble bed during the commissioning and the associated fuel sphere damage rate. This fuel sphere damage rate descreased towards the end of the power operation to 0.6 % of the discharged spheres. A further reduction of the damage rate could have been expected if the operation had continued and the shutdown regime had been changed accordingly. A connection between coolant gas activity and sphere rupture could not be found. The coolant gas activity is in the expected range. The damage rate is not safety-relevant. In terms of operation, HKG has learned how to handle damaged fuel spheres. This was not expected to result in any further unscheduled outages.

There was also the persistent assumption that the hot gas channel damage identified during the scheduled outage at the end of 1988 ruled out the continued operation of the reactor. Figure 3 provides a look into one out of altogether six hot gas channels.

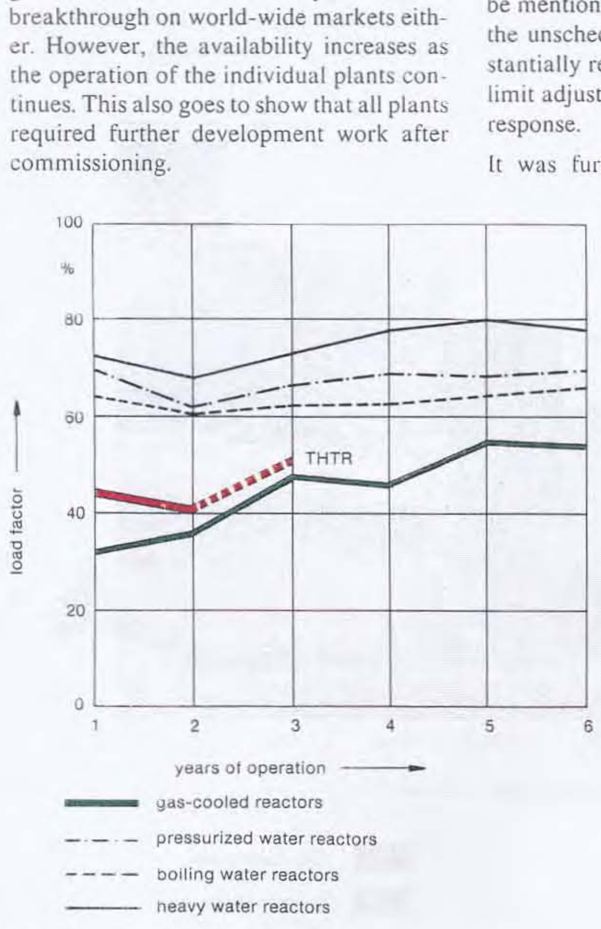


Fig. 2. Mean annual load factor in the first years of operation.

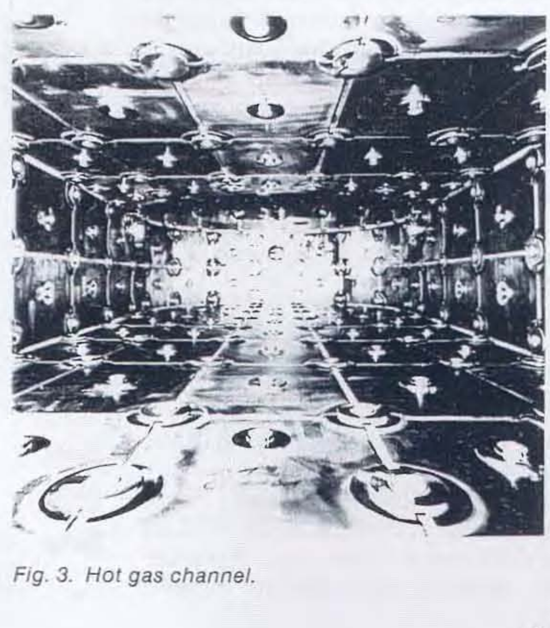


Fig. 3. Hot gas channel.

operating result of the plant would remain negative until the year 2000. This is mainly attributable to the high debt servicing charges in the initial years. After the year 2000, a positive operating result can be expected. Another important conclusion to be drawn from this diagram is that on the assumption of routine operation without any licensing and political risks the uncalled liability of DM 450 million provided for in the RSA would have been adequate.

The blue graph represents the respective financial reserve in the RSA fund. The main burden of the RSA occurs in the first three years of operation when high losses were incurred because the energy availability fell short of the 70 % target.

And this already highlights the problem. The profit and loss account is largely determined by the energy availability and hence the generated kilowatt hours. High losses are the immediate result at least in the first years even if the target of 70 % is only slightly missed. This predicament is exacerbated when analysing the external risks HKG would have been confronted with up until 1992.

The financial risks for the short-term period look completely different from the idealized expectations of a long-term operation over a period of twenty years. For the period up to 1992, additional risks would have been in store for the THTR that could have resulted in extended plant outages and hence loss of revenue. These risks have been summarized in Table 2. They are the following:

Table 2. Risks for continued operation external influence.

- fuel element supply
- transport pick-up hall
- Ahaus intermediate storage facility
- further operating licence

1. Fuel element supply

At the end of 1988, NUKEM stopped the production of fuel elements. Although the fuel element production was intended to be resumed by the Siemens/ABB industrial group, the smooth transition could not be guaranteed for the operation of the THTR. For this reason, HKG had to take into account an outage of the plant for about one year, especially because also the financing of a new production facility could not be settled.

2. The partial operating licence limited to 1100 full-load days and the waste management precautions

The valid operating licence is limited to 1100 full-load days and would have expired in 1992. This valid operating licence specifies that at the time of 600 full-load days HKG has to furnish proof of an available operating licence for the transport pick-up hall for the storage of low-level waste according to section 3 of the Radiation Protection Ordinance. Furthermore, the operator had to demonstrate that the external interim storage of spent fuel elements was secured. As far as the granting of a continuing permanent operating licence was concerned, HKG foresaw again licensing as well as political problems which could also have resulted in a plant outage of one year.

Up to this day, the required proof of waste management precautions cannot be regarded as furnished. HKG considered this to be another outage risk for the plant of about six months.

3. Precautions for decommissioning

For premature decommissioning, the Risk Sharing Agreement provides for funds which have to be reserved and should, together with the allocated decommissioning provisions, be adequate any time to safely and permanently enclose or dismantle the THTR plant. The amount of funds needed for decommissioning was determined by expert opinions prepared by engineering companies experienced in this field. It increased from DM 180 million in 1984 to DM 415 million in 1988.

All these risks have been evaluated and would have resulted in the described cumulative financial burden for the RSA (Figure 5) in the event of continued operation.

- Represented in red are the expected losses arising from normal operation with 70 % availability.
- Yellow is the financing risk through an outage of about six months to furnish proof of waste management capacities.
- Green stands for the outage risk resulting from delayed follow-up manufacture of fuel elements.
- Marked in blue is the loss of HKG incurred between April 1989 and October 1989 as a result of the financing negotiations when the plant had to be kept ready for start-up all the time.

Of central and decisive importance, however, are the precautions which have to be financed for the future decommissioning of the power station. This expenditure could not be covered with the existing funds of the RSA. Since the provisions are allocated continuously over the expected operating period of twenty years, the uncovered amount is of course very high during the first years. Together with the increased losses during the initial phase of operation the risk sum of the RSA therefore turned out to be far too low. Figure 5 reveals that it was exceeded in 1988 already.

These problems which HKG recognized back in April 1987 and indicated to the government authorities could not be resolved among the contracting parties of the RSA. Against this backdrop, the management felt compelled at the end of 1988 to lodge as a precaution an application for decommissioning in accordance with sec-

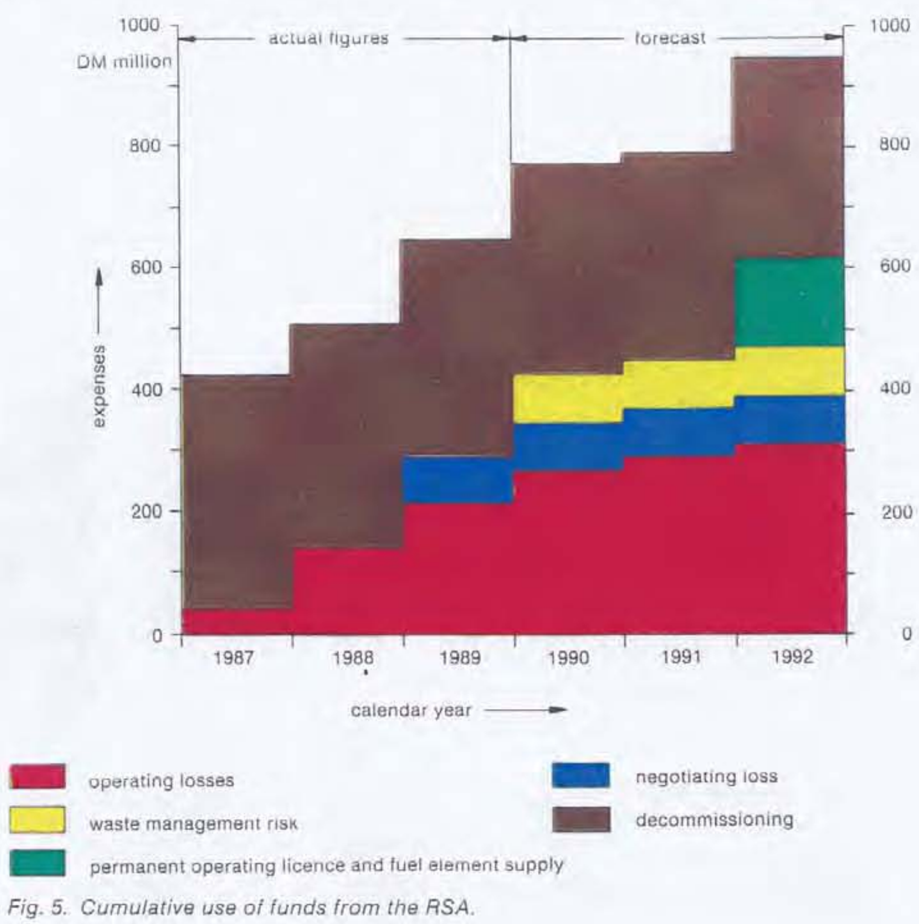


Fig. 5. Cumulative use of funds from the RSA.

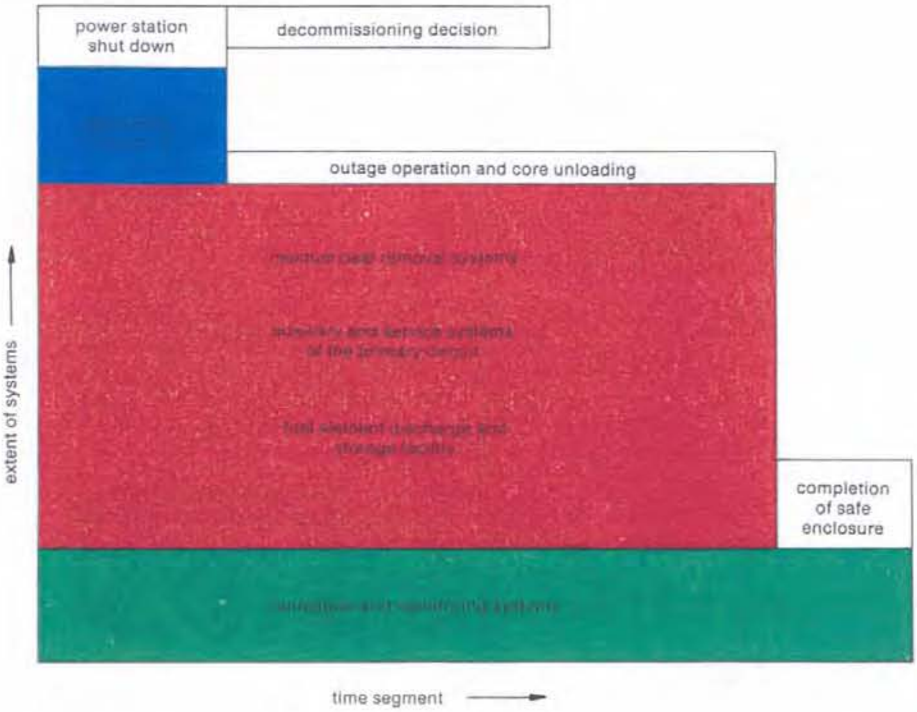


Fig. 8. Decommissioning of systems.

heat removal (liner cooling), gas purification system, fuel element discharge facility and ventilation systems. Upon completion of the safe enclosure, only the ventilation and monitoring systems will remain in operation.

On closer inspection, the core unloading turns out to be a very complex operation. The pebble-bed core consists of fresh and spent fuel elements arranged in layers (Figure 9). In order to maintain a balanced power density distribution, the fresh fuel elements (red) are added at the outside and the spent spheres (yellow) are inside. Elements of average burn-up are blue. When the reactor core is being unloaded via the fuel sphere discharge system, a discharge funnel will gradually form in the pebble bed (steps 1 to 4 in Figure 9) and the fresh fuel elements at the outside will roll to the inside thus enriching the centre of the reactor core with fissile material (red/yellow). This phase of enriching the core centre with fissile material has to be mathematically examined very carefully with accompanying sphere flow experiments in order to prevent local criticality. In this case it is planned to add absorber spheres in the centre of the pebble bed. The actual discharge operation is a time-consuming process and will take about 18 months. The foregoing makes it clear that especially the core unloading is a prototype task too.

After core unloading the safe enclosure can be carried out.

Figures 10 and 11 show where the bounds of the safe enclosure could be placed according to the present planning. Since waste management facilities for highly active plant components are currently not available, the gas purification plant, the fuel circulating system and the prestressed concrete vessel are to be included in the safe enclosure. This results in a very compact arrangement of the safe enclosure which could perhaps be monitored with a separate small ventilation system (red-rimmed).

There is no reason to question the technical feasibility of the safe enclosure over a period of twenty to thirty years and of the subsequent dismantling of the plant.

During the work for the safe enclosure all data obtained from the THTR are to documented, and new findings are to be added by internal inspections, material investigations and the removal of components.

Conclusions

Between 1972 and 1986, the THTR went through a difficult and time-consuming construction phase. A prototype licensing procedure was conducted. Until today, the plant could always be adapted to the current state of the art; This reflects the in

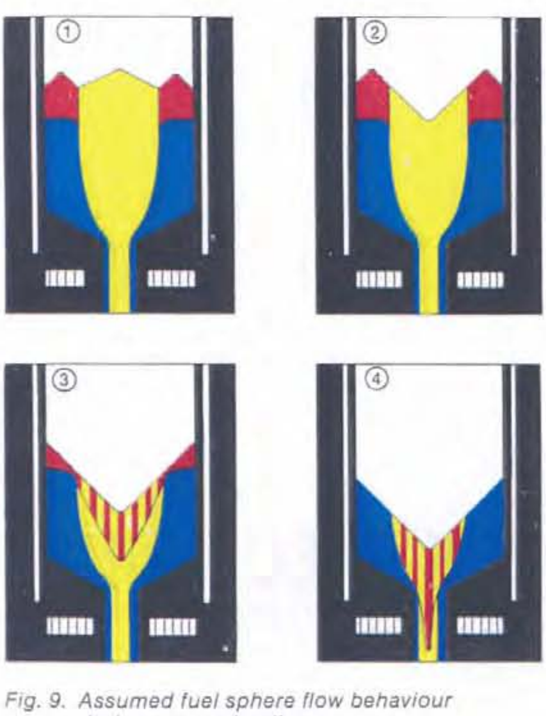


Fig. 9. Assumed fuel sphere flow behaviour during core unloading.

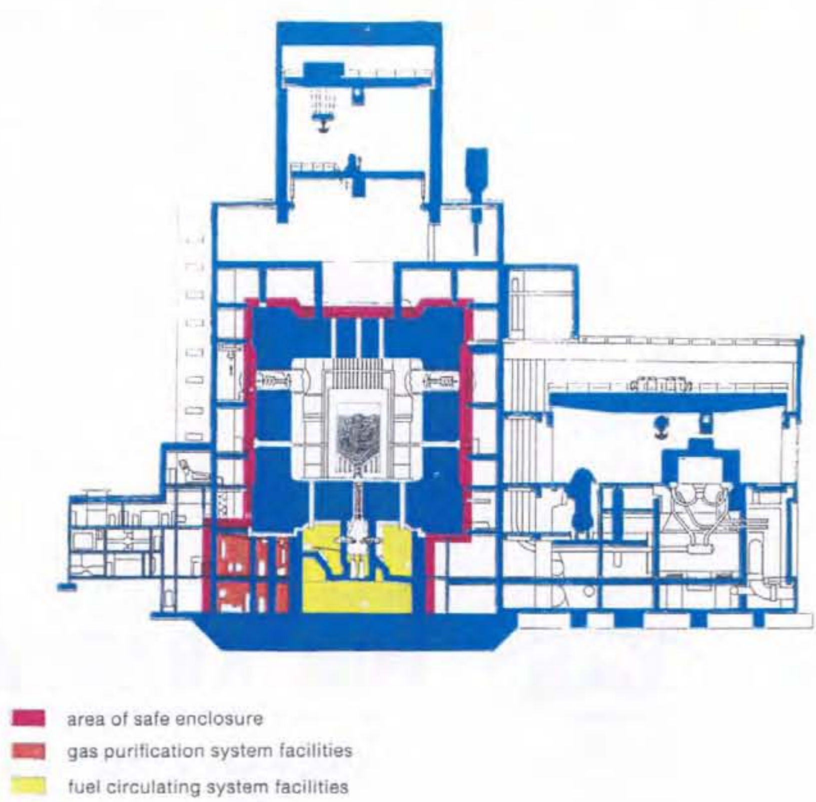


Fig. 10. Safe enclosure concept.

Ausgewählte Themen aus dem Betrieb des THTR 300

Von R. Bäumer

SONDERDRUCK AUS

VGB KRAFTWERKSTECHNIK

MITTEILUNGEN DER VGB TECHNISCHEN VEREINIGUNG DER GROSSKRAFTWERKSBETREIBER E.V.

69. Jahrgang ● Heft 2 ● Februar 1989 ● Seite 158 bis 164

Dieser uns bei Übernahme bekannte Mangel beim Kugelabzug wurde in der Revision 1 erfolgreich beseitigt. Diese große Aufgabe hat einen Zeitraum von fast vier Monaten eingenommen.

Weitere Arbeiten sind auf dem Bild ausgewiesen. So wurde eine Generatorrevision durchgeführt, wiederkehrende Prüfungen vorgenommen und die Verkleidung der Außenfassade instand gesetzt. Nach der Revision 1 folgte von Februar bis März 1988 eine Leistungsperiode, in der die starken Lastabsenkungen am Wochenende nicht mehr notwendig waren, aber sehr wohl aufgrund von Wärmeverbrauchsmessungen in verschiedenen Lastbereichen gefahren wurde. Es zeigt sich in dieser Zeit auch, daß aufgrund der Schädigungsrate der Kugeln vereinzelt Probleme bei der Handhabung der Kugeln in den Förderstrecken entstanden. So wurde in der Revision 2 im April 1988 eine Kugelverklemmung in der Beschickungsanlage beseitigt. Ebenfalls wurden die Behälter für die beschädigten Betriebselemente ausgetauscht, was nur bei abgeschalteter Anlage möglich ist.

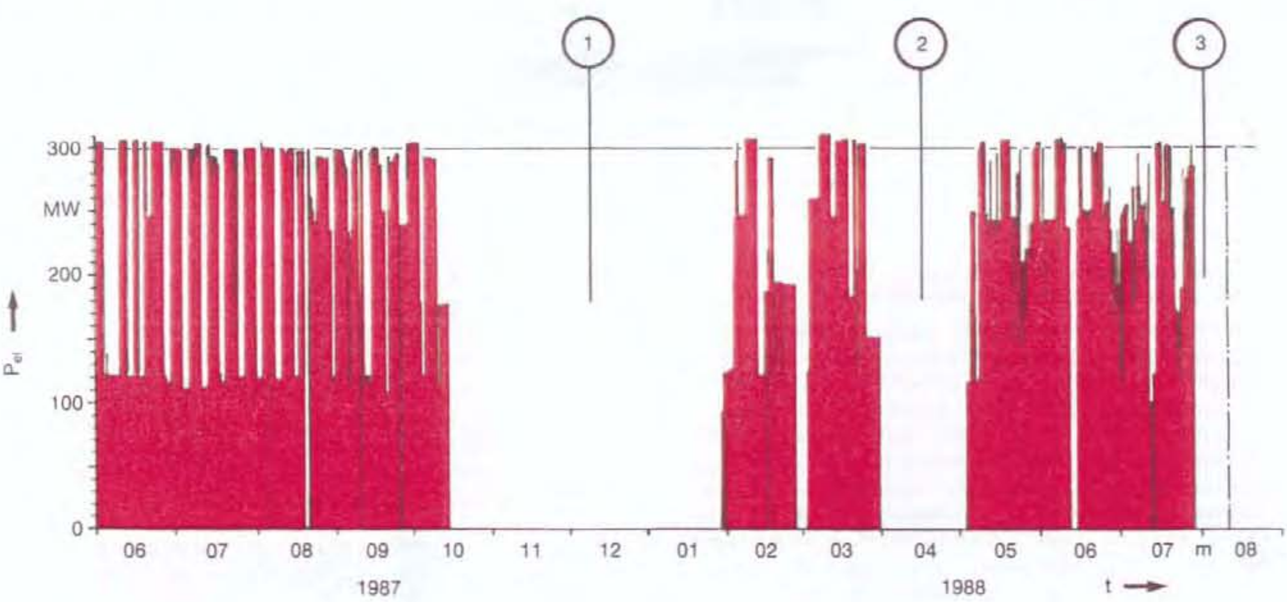
In den Monaten Mai bis Juni 1988 war die Anlage kontinuierlich am Netz mit Lasteinschränkungen unterschiedlicher Stärke. Diese Lasteinschränkungen hängen damit zusammen, daß in bestimmten Anlagenräumen bei hohen Außentemperaturen die genehmigten Grenztemperaturen nur eingehalten werden konnten, wenn die Last und damit die ZÜ-Temperaturen zurückgenommen wurden. Die Kühlanlagen für diese Räume sind nicht ausreichend dimensioniert und bedürfen der Nachrüstung. Auch dieser Mangel war uns bei Übernahme bekannt. Abhilfe soll in der nächsten Revision geschaffen werden, dann — wenn die Zusatzkühlanlage atomrechtlich genehmigt ist.

In der Revision 3, die auch noch im Juli 1988 durchgeführt wurde, erfolgte wiederum die Auswechslung eines Behälters für beschädigte Betriebselemente.

Aus diesen statistischen Daten läßt sich erkennen, daß die Anlage in dem betrachteten Zeitraum während 36,5 % der gesamten Zeit zur Leistungserzeugung nicht zur Verfügung gestanden hat. Während weiterer 15,3 % dieser Zeit war die Anlage zwar verfügbar, konnte aber nicht mit voller Leistung gefahren werden. Es soll nun analysiert werden, aufgrund welcher technischen Besonderheiten diese Nichtverfügbarkeitszahlen zustande gekommen sind. Hier gibt Bild 2 Auskunft.

In Bild 2 ist die Reaktorhalle mit dem großen Spannbetonbehälter dargestellt, ebenso der Kugelhaufen mit den 675 000 Kugeln, die über ein Kugelabzugsrohr abgezogen werden. Um den Spannbetonbehälter herum sind die Reaktorhallenbühnen angeordnet, und man erkennt weiterhin die im Spannbeton eingehängten Dampferzeuger mit den nach oben verlängerten Dampferzeugerringräumen. Links und rechts des Bildes sind die Ursachen für die Verfügbarkeitseinbußen ausgewiesen. Die Auswertung dieser Darstellung führt zu drei wesentlichen Schlußfolgerungen:

— Seit Übernahme der Anlage wurde die Nichtverfügbarkeit der Anlage maßgeblich bestimmt durch die Reparatur am Kugelabzug, Anteil 26,5 %. Schwierigkeiten bei der Hand-



1

— Wiederkehrende Prüfungen

— Generator-Hauptrevision und Kurzrevision an den Hilfsturbosätzen

— Verbesserungen im Bereich des Kugelabzuges

— Verbesserung der Außenfassade der Reaktorhalle

— Verbesserungs- und Instandhaltungsmaßnahmen

2

— Wechsel der beiden Bruchkannen in der Beschickungsanlage

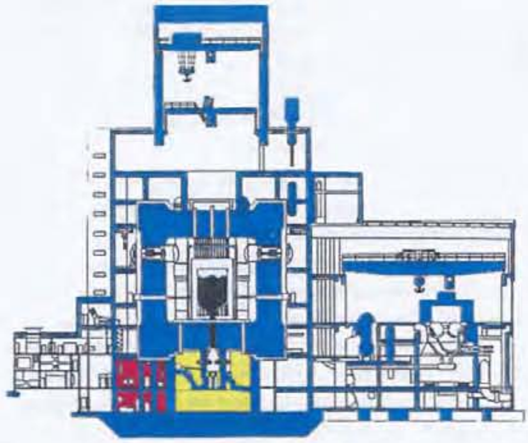
— Beseitigen einer Kugelverklemmung in Vereinzelnerscheibe/Wendelschrottabscheider der Beschickungsanlage

— Wiederkehrende Prüfungen

3

— Wechsel einer Bruchkanne in der Beschickungsanlage und Durchführung von Instandhaltungsmaßnahmen

Bild 1. Elektrische Leistung (1. Juni 1987 bis 31. Juli 1988).



Räume der Gasreinigungsanlage mit einer max. Ortsdosisleistung von 0,3 mSv/h im Arbeitsbereich

Räume der Beschickungsanlage mit einer max. Ortsdosisleistung von <0,3 mSv/h im Arbeitsbereich

Bild 4. THTR-Anlagenräume mit dem wesentlichen Beitrag zur Personendosisbelastung.

In Bild 4 ist dargestellt, in welchen Räumen die wesentlichen Beiträge dieser Personendosisbelastung zustande gekommen sind.

Der THTR ist im Schnittbild dargestellt mit der Reaktorhalle und dem Maschinenhaus. Es sind nur zwei Raumbereiche, die einen Beitrag zur Personendosis liefern, und zwar die Räume der Beschickungsanlage unterhalb des Spannbetonbehälters, in denen die abgebrannten Brennelemente ausgeschleust werden, und die Räume der Gasreinigungsanlage, in denen das Primärgas Helium gereinigt wird. In den Gasreinigungsanlagen beträgt die maximale Ortsdosisleistung 0,3 mSv/h, in den Räumen der Beschickungsanlage ist sie kleiner als 0,3 mSv/h. Alle anderen Räume der Anlage sind praktisch strahlungsfrei.

Nach dem ersten Betriebsjahr kann auch die Anzahl der wiederkehrenden Prüfungen genauer überblickt werden.

In Tafel 1 sind diese Zahlen denen eines Konvoireaktors gegenübergestellt. Im Anlagenzustand „beliebig" stehen 3500 wiederkehrende Prüfungen beim THTR; 1700 wiederkehrende Prüfungen beim Konvoireaktor gegenüber. Bei abgeschaltetem und unter Druck befindlichem Reaktor muß der THTR 430 Prüfungen machen und der Konvoireaktor 25. In der Gesamtheit muß der THTR 2,5mal soviel wiederkehrende Prüfungen machen wie der Konvoi-Druckwasserreaktor. Wenn auch die Aussagefähigkeit dieser Zahlen im einzelnen einer genaueren Bewertung unterzogen werden muß, ist hier doch klar zu erkennen, daß die vielen sicherheitstechnischen Aufrüstungen am THTR während der langen Bauzeit ihren Tribut gefordert haben. Aufgabe der nächsten Betriebsjahre wird es sein, die Vielzahl der wiederkehrenden Prüfungen zu reduzieren.

Tafel 1. Gesamtzahl der jährlich durchzuführenden wiederkehrenden Prüfungen.

Anlagenzustand	Anzahl der Prüfungen	
	THTR 300	Konvoi-DWR
beliebig	3590	1700
Reaktor abgeschaltet und unter Druck	430	25
Reaktor abgeschaltet und drucklos	50	160
Gesamt	4070	1885

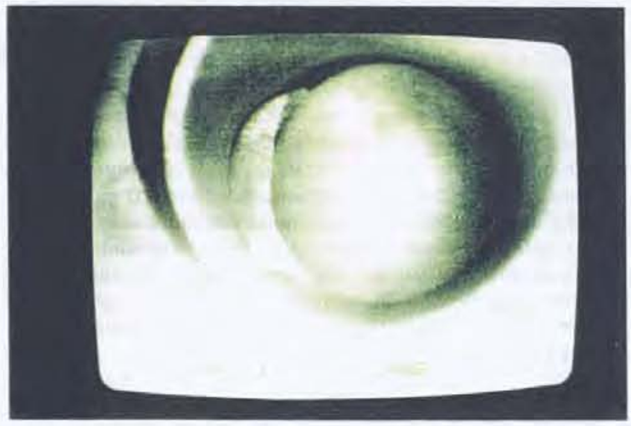


Bild 5. Verklemmte Kugel und Bruchstück beim Durchlauf durch die Vereinzelnerscheibe.

Erkenntnisse über das Kugelfließen im Reaktorkern

Der Kugelhaufenreaktor ist eine Anlage, die ihren Reaktivitätsbedarf durch ständige Zufuhr frischer Brennelemente und ständiges Umwälzen gebrauchter Brennelemente deckt. Der Vorteil dieser Beschickungsstrategie ist die geringe Überschußreaktivität des Reaktors, die sich bei den Sicherheitsanalysen sehr positiv auswirkt. Fällt die Beschickungsanlage allerdings für einen Zeitraum von 15 bis 20 Tagen aus, dann muß der Reaktor in der Leistung reduziert werden und in wenigen Tagen erlischt dann die Kettenreaktion ganz. Störungen in der Beschickungsanlage haben damit erheblichen Einfluß auf die Betriebsgestaltung des THTR. Sie sind in der Vergangenheit vielfach hervorgerufen worden durch beschädigte Kugelelemente. Die Schäden an den Kugeln sind in der Mehrzahl Oberflächenschäden, nur zum geringen Teil erscheinen direkte Durchbrüche. In Bild 5 ist eine abgeplatzte Kugelschale zu sehen, die zu einer Verklemmung geführt hat.

Der Verlauf der Kugel-Schädigungsrate ist in Bild 6 dargestellt. Hier ist die Schädigungsrate in Prozent der jeweils abgezogenen Kugelelemente aufgetragen. 10 % Schädigungsrate bedeuten, daß 10 von 100 abgezogenen Kugeln aufgrund von Schäden aussortiert worden sind. Von Januar bis Juli 1988 ist die Schädigungsrate sehr langsam, aber stetig zurückgegangen. Sie liegt derzeit bei ungefähr 0,6 %.

Der Verlauf der Kurve weist immer wieder Maxima auf. Diese Maxima entstehen immer dann, wenn die Beschickung für eine längere Zeit ausgesetzt wurde, zum Beispiel bei Abschaltungen oder nach Revisionen.

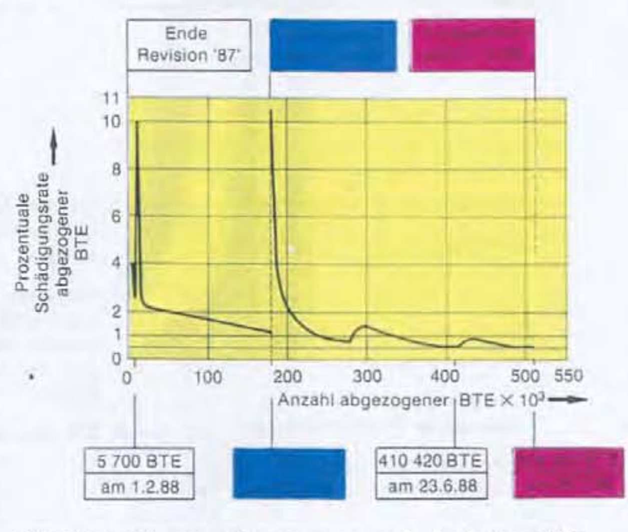


Bild 6. Entwicklung der Kugelbruchrate nach der Revision 1987.

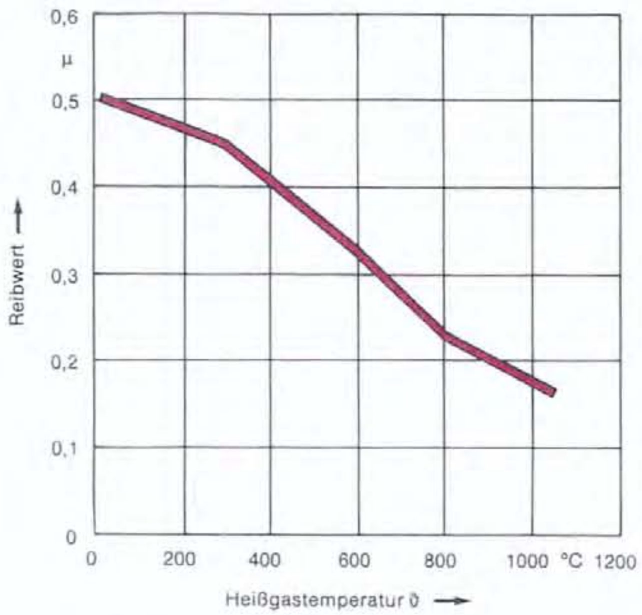


Bild 8. BTE-Reibwerte.

Durchschleusen frischer Brennelemente in der Kernmitte führt zu einem stark unausgeglichenen Temperaturprofil der Heißgastemperatur, mit hoher Innen- und niedriger Außentemperatur. Durch das Verlegen der frischen Brennelemente auf die Randzone des Cores muß für einen Ausgleich gesorgt werden (siehe Bild 9). Aufgetragen ist die Heißgastemperatur im Bodenreflektor über dem Coreradius. Durch Umstellung der Beschickung konnte die Heißgastemperaturspitze innerhalb von drei Wochen von 940 auf 910 °C gesenkt werden.

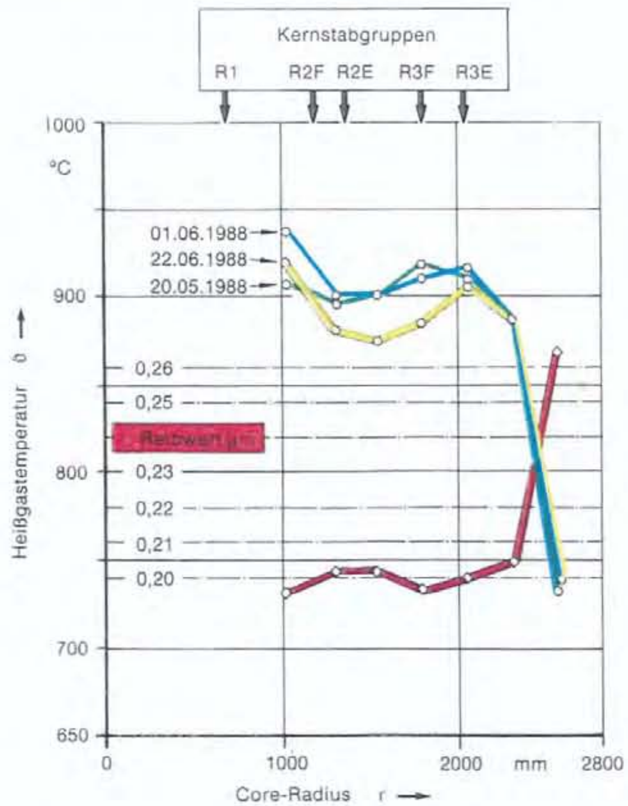


Bild 9. Heißgastemperaturverteilung im Bodenreflektor mit zugehörigem BTE-Reibwert.

Bild 10. Trockenkühlturm — Einzelansicht der Kühldeltas.

Bis zum Erreichen des Gleichgewichtscores in etwa ein bis zwei Jahren wird dieses geänderte Kugelfließen einer weiteren sorgfältigen Beobachtung bedürfen.

Reinigungsmöglichkeiten am Trockenkühlturm

Bild 10 zeigt links die Aluminiumverkleidung des Trockenkühlturms, die an einem Stahlnetzgitter befestigt ist. Rechts sind die im Inneren des Turmes aufgestellten Kühlelemente mit dem Kühlwasservor- und -rücklauf auf den fein gerippten Rohren für den Luftdurchtritt zu erkennen. Eine Reinigungsanlage für den Turm war im Liefervertrag von 1973 nicht enthalten. Während der Inbetriebnahmezeit wurde festgestellt, daß diese fein gerippten Rohre durch Pollenflug und Staubaufwirbelung sich sehr schnell zusetzten und die Kaltwassertemperatur des Hauptkühlwassers durch diesen Effekt um bis zu mehr als 3 °C anstieg. Durch das höhere Temperaturniveau des Hauptkühlwassers wurde eine Vakuumverschlechterung im Kondensator erzeugt, die eine elektrische Minderleistung von etwa 3 MW zur Folge hatte.

Es wurden verschiedene Reinigungsmethoden ausprobiert, die in Bild 11 dargestellt sind. Die erste Reinigung war eine Handreinigung mit Wasserdruckstrahlgeräten mit einem Spitzendruck von 80 bar. Die vier hintereinanderliegenden Rippenrohre wurden zweimal von oben gereinigt und einmal von unten. Mit dieser Grundreinigung, die sieben Wochen dauerte, wurde für die Abnahmemessung ein nahezu sauberer Turm erzielt. Die im unteren Teil des Bildes dargestellten Temperaturabweichungen sind Abweichungen der Kaltwassertemperatur von der Auslegungstemperatur von 26,5 bei 12 °C Lufttemperatur. Die vorher vorhandene Erhöhung der Kaltwassertemperatur um 3 °C ging durch die Grundreinigung auf die Auslegungstemperatur zurück, was in Bild 11 mit dem Wert Null dargestellt ist. In den nächsten drei Wochen wurde nicht gereinigt und der Turm verschmutzte in dieser Zeit sehr schnell, so daß wieder eine Kaltwassertemperaturabweichung von 3 °C zum Auslegungswert vorhanden war. Wir entschlossen uns daher, eine automatische Reinigungsanlage einzusetzen, die von Kühlelement zu Kühlelement umgesetzt werden kann und mit mehreren Düsen das Element besprüht. Gegenüber der Handreinigung wird mit höherem Wasserdurchsatz gearbeitet. Ein dreiwöchiger Reinigungszyklus für jedes Element läßt die mittlere Temperaturerhöhung der Kaltwassertemperatur des Hauptkühlwassers auf 1 °C gegenüber dem gänzlich sauberen Turm absinken. Andere Reinigungsverfahren wurden untersucht, aber verworfen.

Hochtemperaturreaktorlinie

THTR-300 – Erfahrungen mit einer fortschrittlichen Technologie

Von R. Bäumer, Hamm

Der seit dem 16. 11. 85 am Netz befindliche Thorium-Hochtemperaturreaktor THTR-300 hat als Prototyp die in ihn gestellten Erwartungen erfüllt und bereits heute wertvolle Erkenntnisse für die Weiterentwicklung dieser fortschrittlichen Reaktorlinie erbracht. Seine Aufgaben waren im wesentlichen, neuartige Kraftwerkskomponenten im großtechnischen Einsatz zu erproben, Erkenntnisse zu gewinnen für den Bau rein kommerzieller Anlagen sowie den Nachweis zu liefern, daß die technologische Linie des Hochtemperaturreaktors sich im täglichen Kraftwerksbetrieb bewährt. Ein Weiterbetrieb des THTR-300 kann über die jetzigen Betriebserfahrungen hinaus noch wesentliche Erkenntnisse liefern. Zu den weiteren Aufgaben zählen die Erprobung des Langzeitverhaltens der prototypischen Komponenten, die Erweiterung der Kenntnisse zum Kugelfließen innerhalb des Kugelbettes und zur Schädigungsrate der Kugeln sowie die Entwicklung von Ausbau- und Reparaturgeräten für die innerhalb des Spannbetons liegenden Bauteile. Zur Erfüllung dieser Aufgaben und zur Abdekkung des finanziellen Risikos ist die finanzielle Basis für das Projekt THTR-300 neu zu definieren.

1. Partner und Ziele des THTR-Projektes

Am 29. 10. 71 wurde der Liefervertrag für die Errichtung und Inbetriebnahme des Kernkraftwerkes THTR-300 unterzeichnet. Partner dieses Vertrages sind das *Bundesministerium für Forschung und Technologie (BMFT)*, das *Land Nordrhein-Westfalen*, die *Hochtemperatur-Kernkraftwerk GmbH (HKG)* als Bauherr und Betreiber sowie ein Errichterkonsortium, bestehend aus den Firmen *ASEA Brown, Boveri AG* (früher: Brown, Boveri & Cie AG), der *Hochtemperatur-Reaktorbau GmbH (HRB)*, Mannheim und der Firma *Nukem GmbH*, Hanau.

Anschrift des Verfassers:
Dr. R. Bäumer, Hochtemperatur-Kernkraftwerk GmbH (HKG), Siegenbeckstr. 10, 4700 Hamm. 1.

Bereits damals war klar, daß der Hochtemperaturreaktor eine bedeutende Innovation auf dem Energieversorgungssektor darstellt; daß er einen wichtigen Beitrag zur sicheren umweltschonenden und wirtschaftlichen Energieversorgung leisten kann. Ein besonderer Anreiz für die Entwicklung dieser Reaktorlinie war die Möglichkeit, mit der hohen Temperatur des Primärkreislaufes Wärme zur Nutzung für chemische Prozesse, u. a. zur Kohlevergasung oder zur Streckung von Kohlenwasserstoffen, auszukoppeln. Ebenso war ein wichtiger Anreiz die Chance, mit dieser Linie kleine Reaktoren in wirtschaftlichen Einheiten bauen zu können.

Großtechnische Technologieentwicklungen wie sie zur Einführung einer neuen Reaktorlinie notwendig sind, vollziehen sich nicht kurzfristig, sondern bedürfen einer stetigen Fortführung von Forschung und Entwicklung, die durch den Bau von Prototypanlagen begleitet wird. Auf diesem Wege von der Forschung zur kommerziellen Realisierung der Hochtemperaturtechnologie stellt der THTR-300 das Bindeglied zwischen dem 15 MWel-Versuchsreaktor AVR und einer späteren rein kommerziellen Anlage dar.

Projektziele sind daher die Bestätigung der technologischen Machbarkeit eines großen Hochtemperaturreaktors und auch das Gewinnen von Erfahrungen, die in den Bau einer späteren kommerziellen Anlage einfließen können. Für dieses Projekt haben von 1971 bis heute Teile einer ganzen Generation von Ingenieuren ihren Sachverstand eingesetzt. Nachdem der THTR-300 am 16. 11. 85 erstmals Strom in das öffentliche Verbundnetz lieferte und nachdem am 1. 6. 87 nach erfolgreicher Inbetriebnahme die Übergabe an den Betreiber erfolgte, kann aus heutiger Sicht die zielstrebige Arbeit der Ingenieure als ein großer Erfolg angesehen werden.

Bund und Land verfolgten die Absicht, die Linie des Hochtemperaturreaktors aus volkswirtschaftlichen Gesichtspunkten weiter zu entwickeln. Beide haben aufgrund dieser Zielsetzung bis heute die wesentlichen finanziellen Mittel zur Realisierung des Projektes beigetragen. In Abb. 1 sind die bis heute aufgebrachten Finanzmittel aufgeschlüsselt.

Das Lieferkonsortium wollte aus industrieller Sicht die technische Machbarkeit der Hochtemperaturtechnologie an einer Großanlage demonstrieren. Auch das *Konsortium THTR* stellte finanzielle Mittel zur Verfügung und führte mit dem Sachverstand seiner Ingenieure die technischen Detailplanungen und Untersuchungen durch.

Die *Hochtemperatur-Kernkraftwerk GmbH (HKG)* ist Bauherr und jetziger Betreiber wie atomrechtlicher Inhaber der Anlage THTR-300. Sie besteht aus den Gesellschaftern

Gemeinschaftskraftwerk Weser GmbH, Porta Westfalica-Veltheim	26%
Elektromark Kommunales Elektrizitätswerk Mark AG, Hagen	26%
Vereinigte Elektrizitätswerke Westfalen AG, Dortmund	31%
Gemeinschaftswerk Hattingen GmbH, Hattingen	12%
Stadtwerke Aachen AG, Aachen	5%

Die Prozente geben die Anteile am Stammkapital (90,0 Mio. DM) wieder.

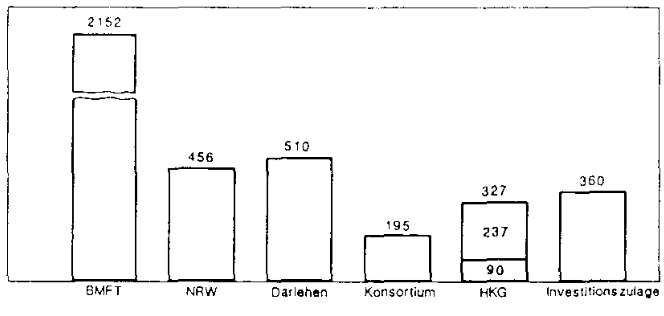


Abb. 1: Mittelherkunft für das Projekt THTR-300.

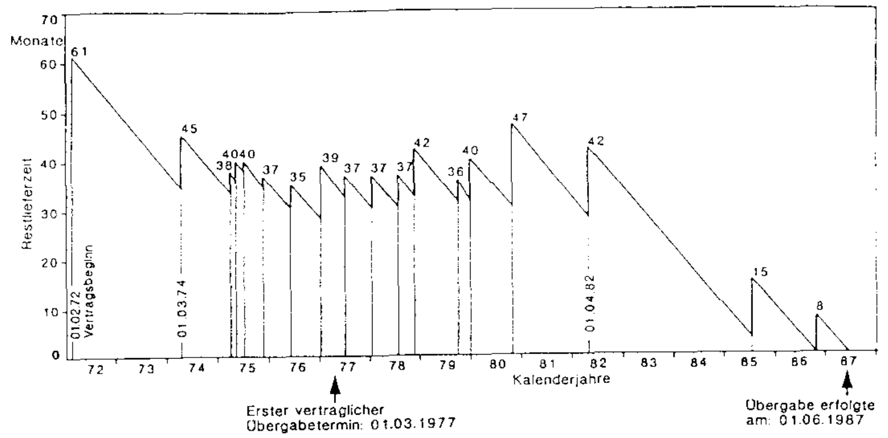


Abb. 2: Entwicklung der Lieferzeit THTR-300.

Die Betreibergesellschaft stellte den Standort für den THTR-300 zur Verfügung. Sie mußte auch die notwendige Leistungsreserve im Netz zur Verfügung haben, um den nicht immer kontinuierlichen Betrieb einer Prototypanlage im Netz auffangen zu können.

Des weiteren haben es die Gesellschafter übernommen, die elektrische Arbeit aus dem THTR-300 zu festgelegten Bedingungen zu übernehmen. Aus heutiger Sicht sind die THTR-300-Strompreise aufgrund der Kostensteigerungen während der Errichtung derart gestiegen, daß die HKG-Gesellschafter mit der Übernahme der Stromabnahmeverpflichtung einen weiteren erheblichen finanziellen Beitrag für die Existenz dieser Anlage leisten.

Die HKG hat sich immer als eine Gesellschaft verstanden, die die Erfahrungen und Erkenntnisse aus Planung und Errichtung des Forschungsprojektes THTR-300 allen Partnern aus Industrie und Elektrizitätswirtschaft auch auf internationaler Ebene freizügig übermitteln will. Sie hat sich frühzeitig um den Status eines *Gemeinsamen Europäischen Unternehmens* bemüht und hat während der Zeit der Errichtung dafür Sorge getragen, daß Lieferfirmen aus dem gesamten europäischen Raum an dem Bau dieses Projektes beteiligt wurden. So wurde z. B. die Innenisolierung des Spannbetonbehälters aus Großbritannien geliefert; die Panzerrohre für die Durchführungen im Spannbetonbehälter aus Italien; Frankreich war am Dampferzeuger beteiligt, ebenso Firmen aus der Schweiz.

Eine weitere Aufgabe, die von der HKG ganz wesentlich mitgefördert wurde, ist die Schaffung einer Infrastruktur für die Fortführung der Hochtemperaturreaktorlinie. Unter Infrastruktur ist hier der Aufbau einer Brennelementversorgung mit den Fertigungsanlagen für kugelförmige Brennelemente wie auch auf die Schaffung von Zwischenlager- und Endlagermöglichkeiten für abgebrannte Brennelemente zu verstehen. Hierfür hat sich die HKG in der Vergangenheit finanziell wie auch öffentlich immer wieder engagiert.

Heute nun ist das Projekt THTR-300 in finanzielle Schwierigkeiten geraten, die noch nicht gelöst sind. Auf jeden Fall ist es Zeit, einmal Bilanz zu ziehen.

2. Erfahrungen

2.1. Erfahrungen aus der Errichtung der Anlage

Zu Beginn der Errichtungsphase des THTR-300 gab es weder für den Hochtemperaturreaktor spezifische technische Regeln, noch ein Genehmigungsverfahren, das für die THTR-spezifischen Komponenten durchgeführt worden wäre. Im Zuge der THTR-300-Abwicklung bestand also die Aufgabe darin, die Anforderungen aus dem und an das Genehmigungsverfahren neu zu formulieren. Auch hier wird der Forschungs- und Entwicklungscharakter des THTR-300 deutlich.

Eine Analyse des Projektablaufes dokumentiert, daß nach zügigem Baubeginn bis 1974 dann von 1974 bis 1982 die Fertigstellung der Anlage stagnierte, obwohl mit vollem Personaleinsatz auf der Baustelle gearbeitet wurde. Dieser Effekt ist darauf zurückzuführen, daß im Jahre 1974 auf dem Gebiet der Kernenergie die Entwicklung des Technischen Regelwerkes stark voranschritt. So wurden die BMI-Kriterien festgelegt, wie auch die RSK-Leitlinien, alle aber bezogen auf den Druckwasserreaktor. Es wurden Anforderungen gestellt, die Anlagen auszulegen gegen

- Einwirkungen von Außen: Flugzeugabsturz, Druckwelle, Erdbeben, Objektschutz;
- Einwirkungen von Innen: Beherrschung spontaner Rohrbrüche und Behälterversagen und
- es wurden neue Vorschriften erlassen, wie z. B. Strahlenschutzverordnungen u. .a. spezielle Regelwerke.

Nachdem im Jahre 1974 der gesamte Baukörper, also der Rohbau der THTR-300-Anlage bereits fertiggestellt war, mußten planerische Änderungen am Gesamtkonzept erfolgen, um diese neuen Regelwerke in eine vorhandene Baustruktur HTR-gerecht zu integrieren. Dabei wurde mit fortschreitender Abwicklung die ursprüngliche Planung teilweise verlassen, und der Schwerpunkt der Arbeiten rückte immer mehr in den Bereich „Forschung und Entwicklung". Dies wird besonders deutlich, wenn man sich die Terminabwicklung in Abb. 2 vor Augen führt. Am 1. 3. 74 hatte das Projekt aufgrund einer ersten Terminanpassung eine Restlaufzeit von 45 Monaten auszuweisen. Durch eine Vielzahl von Terminverlängerungen wurde zwischen 1974 und 1982 trotz vollem Einsatz in Planung und Errichtung durch immer neu auftretende Anforderungen die Restlaufzeit immer noch im Bereich von 40 Monaten ausgewiesen. Der vertragliche Übergabetermin verschob sich vom 1. 3. 77 auf den 1. 6. 87.

Die Arbeiten schritten erst dann zügig voran, nachdem Planungsrichtlinien für den THTR-300 erarbeitet wurden, die die Umsetzung der BMI-Kriterien für Druckwasserreaktoren für den Hochtemperaturreaktor vollzogen. Hierdurch war eine formale Voraussetzung geschaffen, HTR-gerechte Kriterien im Genehmigungsverfahren anzuwenden.

Die Anpassung des Projektes THTR-300 an den Stand von Wissenschaft und Technik und die Anpassung an das im Wandel befindliche Genehmigungsverfahren führten zu einer erheblichen Projektkostensteigerung für die Anlage (Abb. 3). Der Zeitraum bis 1979 war durch das Anpassen des Projektes an

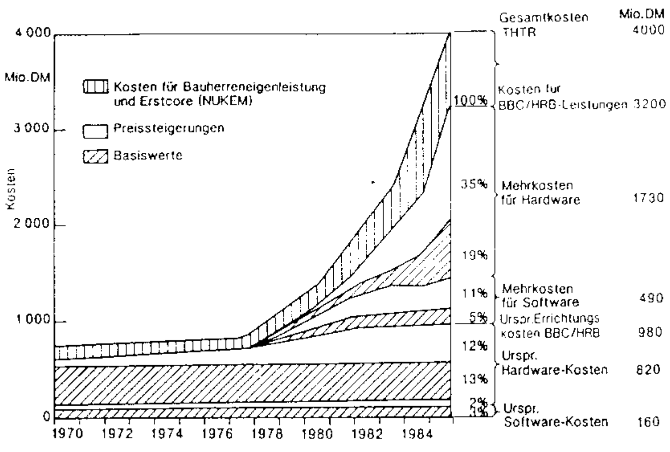


Abb. 3: Projektkostenentwicklung des THTR-300.

Tabelle 1: Hersteller der Prototyp-Komponenten

Brennelemente	Nukem, Hanau
Absorberstäbe	Brown, Boveri & Cie. AG
Gebläse	Brown, Boveri & Cie. AG
Liner	Steinmüller, Gummersbach
Spannbetonbehälter	Krupp Universalbau, Essen
Beschickungsanlage	Leybold-Heraeus, Hanau
Gasreinigungsanlage	Linde, München
Keramische Einbauten	Sigri, Augsburg
Trockenkühlturm	GEA/Balcke-Dürr, Ratingen
Rohrleitungen des Wasserdampfkreislaufes	Mannesmann, Düsseldorf
Dampferzeuger	Sulzer/EVT, Winterthur/Stuttgart
Bauteil	Heitkamp/Hochtief, Herne/Essen

die ständig erhöhten sicherheitstechnischen Anforderungen sowie durch die aus der Anpassung resultierende stop-and-go-Abwicklung mit einem relativ flachen Verlauf der Kostenkurve bestimmt. Der überproportionale Kostenanstieg ab 1978 ist im we lichen auf den erhöhten Personaleinsatz aufgrund des gestiegenen Bearbeitungs- und Nachweisumfanges, den Beginn der Montageaktivitäten sowie auf das Bestreben zurückzuführen, die entstandenen Terminverzögerungen aufzuholen. Es muß jedoch beachtet werden, daß die erhöhten Projektkosten, bezogen auf den Lieferumfang des Herstellers, ihre wesentliche Ursache in der Preisgleitung (60%) und den überproportional gestiegenen Ingenieurleistungen hatte.

Der eigentliche prototypische Teil der Anlage THTR-300 ist der Spannbetonbehälter mit allen seinen Einbauten. Für die Errichtung dieses Bauteils haben erhebliche Teile der deutschen Industrie Entwicklungsarbeit geleistet. Hier konnten auch die Erfahrungen des Versuchsreaktors AVR nicht voll zum Tragen kommen, da eine andersartige Technik angewandt wird; so z.B. für die Abschaltung des Reaktors mit in den Kugelhaufen einfahrenden Absorberstäben, so z.B. für die Einschließung des Primärkreis-Heliums mit einem Spannbetonbehälter, wie auch für die Strömungsführung des Heißgases innerhalb des Spannbetonbehälters selbst. In Tab. 1 sind einige Hersteller von Prototypkomponenten aufgezählt, die beispielhaft für die große Innovativkraft der deutschen Industrie sein mögen. Nach der Fertigstellung der Anlage THTR-300 kann daher guten Gewissens gesagt werden, daß die deutsche Industrie Erfahrung in der Herstellung von Prototypkomponenten für einen Hochtemperaturreaktor dergestalt gewonnen hat, daß sie jederzeit in der Lage wäre, für eine Folgeanlage das nötige Know-how zu liefern.

Aus den Erfahrungen bei Errichtung des THTR-300 können folgende Feststellungen getroffen werden:

1. Die wirtschaftliche Errichtung eines Folge-HTR ist nur bei kalkulierbarer Bauzeit möglich.
2. Eine wesentliche Voraussetzung hierfür sind weitgehend abgeschlossene Genehmigungsverfahren bei Baubeginn. Diese Erkenntnisse wurden bei der Planung des HTR-500 berücksichtigt.

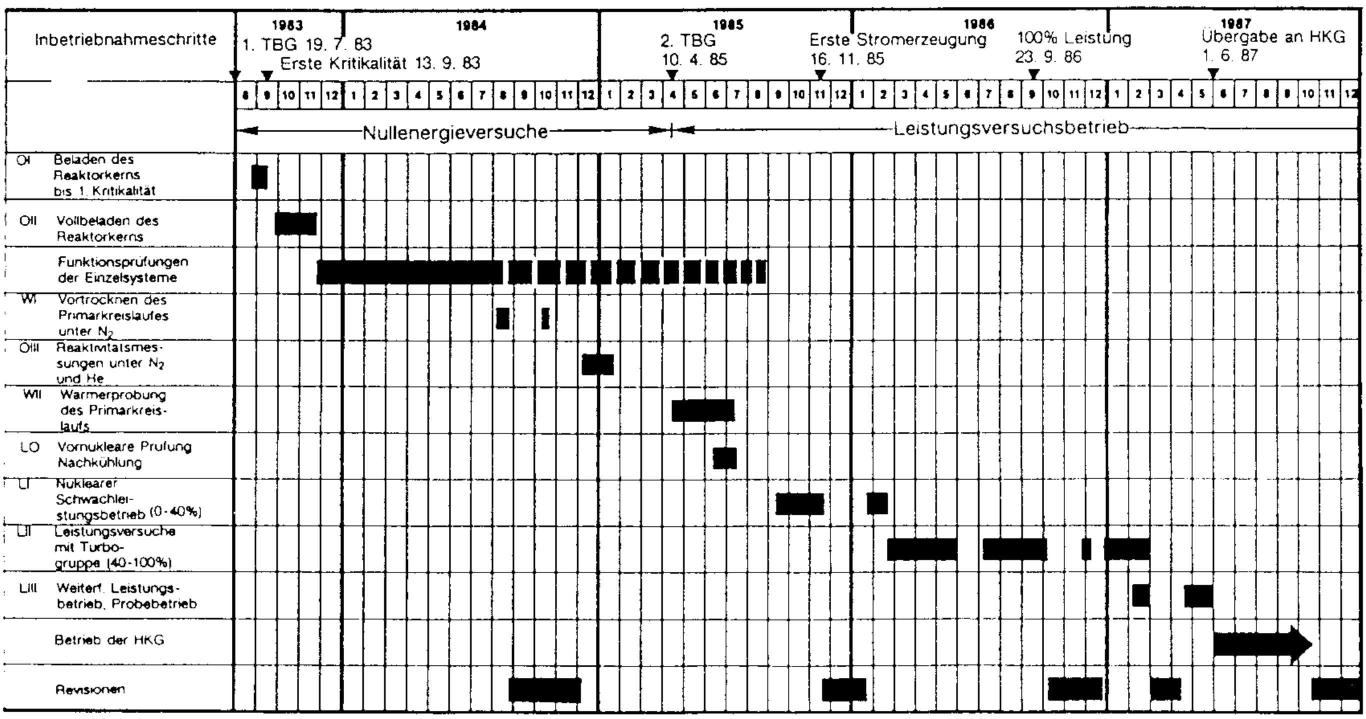


Abb. 4: Abwicklung der Inbetriebnahmeschritte.

2.2. Erfahrungen aus Inbetriebnahme und Betrieb

Die Inbetriebnahme des THTR-300 wurde in zwei atomrechtliche Betriebsgenehmigungen aufgeteilt:

Die erste Teilbetriebsgenehmigung für die Beladung mit Betriebselementen, die Versuche mit sehr niedriger Leistung (Null-Energie-Versuche) und die Vortrocknung des Primärkreislaufes unter Stickstoff wurde am 19.7.83 erteilt:

Die zweite Teilbetriebsgenehmigung für den Leistungsversuchsbetrieb lag am 9.4.85 vor; sie gilt bis zum Erreichen von 1100 Reaktorvollasttagen.

Die Strategie der THTR-Inbetriebnahme war dadurch gekennzeichnet, daß zum frühestmöglichen Zeitpunkt parallel zu den Restmontagen mit der Kernbeladung und den nuklearen Nullenergieversuchen begonnen wurde. Daraus ergab sich die Möglichkeit, die Kernauslegung so früh wie möglich zu verifizieren, die Funktion der Reaktorkomponenten des Primärkreislaufes rechtzeitig zu testen und eventuelle Korrekturen termingerecht zu veranlassen.

Die nukleare Inbetriebnahme wurde in neun Inbetriebnahmeschritte unterteilt (Abb. 4). Die äußerst sorgfältig und dem Charakter des Prototyps und der Forschungsanlage angemessene Abwicklung der Inbetriebnahme läßt sich auf sechs Eckdaten zurückführen. Diese sind:

Beginn	30.8.83
Erste Kritikalität	13.9.83
Erstmals Nukleare Leistung	6.9.85
Erste Stromlieferung	16.11.85
Erstmals 100%-Leistung	23.9.86
Übernahme durch HKG	1.6.87

Die bis heute gewonnenen wesentlichen Betriebsergebnisse nach Bereitstellung von 2891000 MWh elektrischer Arbeit und 16410 h Reaktorbetrieb lassen sich wie folgt zusammenfassen:

Die Zeitausnutzung der Anlage betrug in 1987 61%,
in 1988 52%.

Als eindeutig positive Ergebnisse lassen sich nennen:
- die Auslegungsleistung von 100% wurde auf Anhieb erreicht.
- die thermodynamischen Primärkreislaufdaten haben sich bestätigt.
- die Personendosisbelastung ist äußerst gering.
- Ausbauarbeiten am Primärkreis sind trotz schwieriger Zugangsverhältnisse durchführbar.

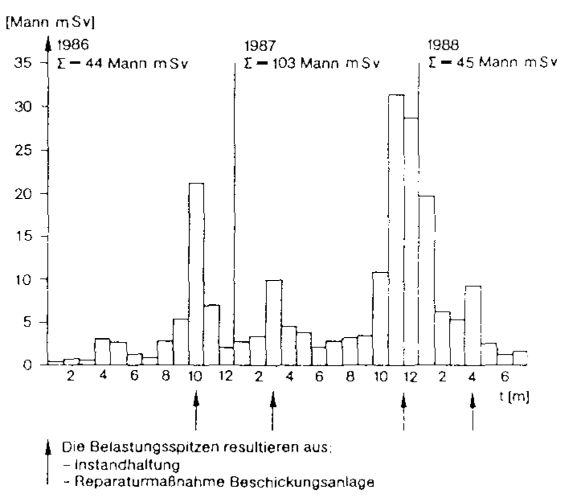


Abb. 5: Monatlich akkumulierte Gesamtpersonendosis.

Alle diese Ergebnisse bestärken uns in der Aussage, daß mit den nunmehr vorliegenden Erfahrungen aus dem THTR-300 ein technischer Anforderungskatalog erstellt werden kann, mit dem ein Folgereaktor in die entscheidende Planungsphase gehen könnte.

Als Beleg für die obengenannten positiven Betriebsergebnisse dient Abb. 5 mit der Darstellung der monatlich akkumulierten Gesamtpersonendosis.

Die Übereinstimmung der technischen thermodynamischen Datenvorhersage und die Meßergebnisse aus der Inbetriebnahme zeigt Abb. 6 am Beispiel der Temperaturverläufe für das Schnellabfahren aus 60% Leistung.

Neben diesen positiven Betriebserfahrungen wurden in der bisherigen Betriebszeit des THTR-300 wertvolle Erkenntnisse gewonnen. Abb. 7 zeigt das Diagramm der elektrischen Leistung vom 1.1.88 bis 29.9.88, aus dem sich auch die obengenannte Zeitverfügbarkeit für 1988 errechnet. Eine Analyse der Nichtverfügbarkeitszeiträume zeigt eindeutig, daß die Ursachen überwiegend im Bereich der Beschickungsanlage und im Bereich der Handhabung der Kugelelemente zu suchen sind. Diese Handhabung wird zusätzlich erschwert durch das Vorhandensein von beschädigten Kugelelementen, die während der Zeit der Inbetriebnahme durch die umfangreichen Fahrversuche der Kernstäbe verursacht wurden. Da die Kugelschädigungsrate rückläufig ist – sie liegt derzeit bei ca. 0,6% der abgezogenen Kugeln –, ist auch langfristig mit einem Rückgang der Schwierigkeiten bei der Kugelhandhabung zu rechnen.

Zur Bewältigung der Schwierigkeiten bei der Handhabung der Kugelelemente wurden bis heute verschiedene Maßnahmen ergriffen:
- Es wurde eine Reparatur am Kugelabzug in unmittelbarer Nähe des Primärkreises durchgeführt, die ein verbessertes Ausschleusen der Kugelelemente bei 100% Reaktorleistung und vollem Gasdurchsatz ermöglicht.
- Um die Kugelbeschädigungen niedrig zu halten, wurden die Kugelbelastungen dadurch reduziert, daß die Einfahrtiefe und die Zahl der Kernstäbe bei längeren Abschaltungen in Abhängigkeit der erforderlichen Abschaltreaktivität festgelegt wird. Ebenso wurde das Verfahren der NH_3-Einspeisung, das eine Schmierwirkung bei der Abschaltung erzeugt, optimiert.
- Die beschädigten Kugeln werden langsam aus dem Reaktorkern ausgeschleust.

Die Kugelbeschädigungen und die Probleme bei der Handhabung haben keine sicherheitstechnische Bedeutung; die Kühlgasaktivität liegt unverändert im Erwartungsbereich.

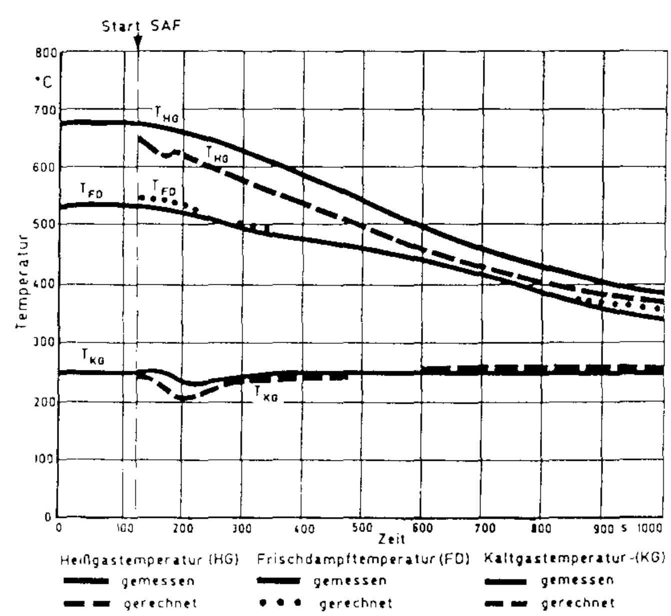


Abb. 6: Schnellabfahren aus 60%-Leistung (Temperaturen).

Damit kann auch ein signifikanter Bruch beschichteter Teilchen ausgeschlossen werden.

Die Zeitnichtverfügbarkeit gegen Ende 1988 wurde bestimmt durch den Schaden am Heißgaskanal. In einer Routinerevision besichtigte die HKG vorsorglich einen Heißgaskanal, der die Strömungsführung des heißen Heliums vom Reaktorkern zu den Dampferzeugern übernimmt. Abb. 8 zeigt einen Blick in einen Heißgaskanal mit dem rechteckigen Durchtritt durch den graphitischen Seitenreflektor. Die unteren Kanalblöcke aus Graphit sind jeweils mit einem Graphitdübel auf den Kohlesteinblöcken fixiert. In der Außenwand des Seitenreflektors sind diese Dübel in Durchgangsbohrungen der Blöcke angeordnet. Weiter ist in Abb. 8 der Anfang des metallischen Teils des Heißgaskanals zu sehen mit der Innenisolierung, bestehend aus metallischen Folienpaketen, 30 cm × 30 cm Abdeckfläche, die mit jeweils 4 Eckbolzen und 1 Zentralbolzen fixiert werden. Nachdem bei der ersten Kanalbesichtigung Schädigungen an den Befestigungsbolzen (Zentralbolzen) festgestellt wurden, entschloß sich die HKG, alle sechs Kanäle zu besichtigen und stellte dabei fest, daß von ca. 2600 Bolzen insgesamt 35 Bolzenköpfe abgesprengt waren. Es wurde außerdem festgestellt, daß die Graphitdübel zur Fixierung der unteren Außenkanalblöcke ausgehoben waren.

In der Folgezeit bis heute wurde dieser festgestellte Schaden gfältig untersucht und als Schadensursache liegt folgendes Ergebnis vor:

Die Bolzen haben versagt aufgrund einer Abnahme der Werkstoffduktilität infolge der Bestrahlung mit thermischen Neutronen und im Temperaturbereich größer 500 °C. Desweiteren ergaben sich Spannungskonzentrationen im Kopfbereich der Bolzen durch unterschiedliche Wärmeausdehnungsquofitienten der Materialien der 18lagigen Metallfolienisolierung und der Konstruktion der Haltebolzen.

Die HKG kommt nach sorgfältiger Prüfung gemeinsam mit dem Anlagenlieferer zu der Ansicht, daß ein Weiterbetrieb des

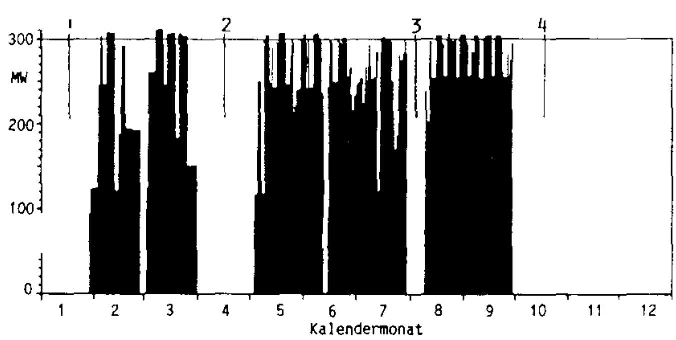


Abb. 7: Diagramm der elektrischen Leistung 1. 1. 88 bis 29. 9. 88.

1
- Verbesserungen im Bereich des Kugelabzuges
- Verbesserung der Außenfassade der Reaktorhalle
- Generator-Hauptrevision und Kurzrevision an den Hilfsturbosätzen
- Wiederkehrende Prüfungen und Instandhaltungsmaßnahmen

2
- Wechsel der beiden Kannen für beschädigte Betriebselemente
- Beseitigen von Kugelverklemmungen in Vereinzelnerscheibe und Wendelabscheider
- Wiederkehrende Prüfungen

3
- Wechsel einer Kanne für beschädigte Betriebselemente
- Instandhaltungsmaßnahmen

4
- Wechsel einer Kanne für beschädigte Betriebselemente
- Inspektion des Wendelabscheiders YE04 D010 und der Heißgaskanäle
- Wiederkehrende Prüfungen
- Instandhaltungs- und Änderungsarbeiten

Abb. 8: Blick in einen Heißgaskanal, im Vordergrund Steine des Graphitaufbaues.

THTR-300 mit den vorhandenen Schäden vertretbar ist. Dadurch, daß sich die Schäden im wesentlichen auf die Zentralbolzen konzentrieren, wird die Innenisolierung im metallischen Teil des Heißgaskanals nach wie vor durch die Eckbolzen gehalten. Damit ist die Funktionsfähigkeit der Innenisolierung auch jetzt sicher gewährleistet. Sollte es dennoch zum Ablösen von Teilen der Isolierung kommen, so läßt sich dieses anhand der betrieblichen Überwachung der Prozeßparameter Massenstrom und Druckverlust über dem Heißgaskanal erkennen. Die HKG wird gleichwohl weiterhin durch kürzere Besichtigungsintervalle das Schadensbild beobachten.

Die aus dem bisherigen Betrieb gewonnenen Erkenntnisse haben schon wesentlich zum gesetzten Forschungsziel des THTR-300 beigetragen und wichtige Erkenntnisse für die Planung einer Folgeanlage liefern können.

2.3. Erwartete Erfahrungen bei einem Weiterbetrieb des THTR-300

Ein Weiterbetrieb des THTR-300 kann über die jetzigen Betriebserfahrungen hinaus noch wesentliche Erkenntnisse liefern und damit den Forschungsauftrag sinnvoll abrunden. Insbesondere erwartet die HKG, daß sich die Kenntnisse zum Kugelfließen innerhalb des Kugelbettes, die Erkenntnisse zur Schädigungsrate der Kugeln und die Erkenntnisse über Aktivitätsfreisetzung aus den Kugeln noch sehr stark verbessern und erhärten lassen.

Eine weitere Aufgabe wäre die Erprobung des Langzeitverhaltens der prototypischen Komponenten. Der Heißgaskanal ist hier nur ein im Augenblick signifikantes Beispiel, aber auch die anderen Prototypkomponenten, wie Abschaltstäbe, Spannbetonbehälter, Graphitaufbau sind hier langfristig von großem Interesse.

Eine weitere Aufgabe, die bei einem Weiterbetrieb angegangen werden kann, ist die Entwicklung von Ausbau- und Reparaturgeräten für die innerhalb des Spannbetons liegenden Bauteile. Der bisherige Betrieb hat gezeigt, daß die Frage der Zugänglichkeit für den Betreiber von herausragender Bedeutung ist und die Entwicklung von Ausbaugeräten dringend erforderlich ist. Der Nachweis der Reparaturfreundlichkeit von Hochtemperaturreaktoren ist nach Meinung der HKG somit auch Aufgabe eines Prototyps.

Damit diese Aufgaben jedoch angegangen werden können, ist die finanzielle Basis für das Projekt THTR-300 neu zu definieren.

3. Der Risikobeteiligungsvertrag und die Abdeckung finanzieller Risiken

Bereits im Jahre 1971 waren sich die Partner des Projektes THTR-300 darüber im klaren, daß ein von Anfang an kommerzieller Betrieb des THTR-300 aufgrund seines Prototypcharakters und der mit ihm verfolgten Forschungsziele nicht zu erreichen ist. Aus diesem Grunde wurde in vielen Verhandlungen bereits zu Beginn des Projektes ein Risikobeteiligungsvertrag (RBV) geschlossen, der zur Abdeckung der wirtschaftlichen Betriebsrisiken und zur Abdeckung der Stillegung der Anlage eine Haftsumme von zum gegenwärtigen Zeitpunkt 450 Mio. DM ausweist. Diese Haftsumme von 450 Mio. DM wurde zu ⅔ vom Bund und zu ⅓ vom Land Nordrhein-Westfalen aufgebracht. 270 Mio. DM sind für den Ausgleich von Betriebsverlusten reserviert; 180 Mio. DM derzeit für die Stillegung der Anlage.

Der Vertrag sieht weiter vor, daß während der ersten drei Jahre die HKG-Gesellschafter 10% der Betriebsverluste übernehmen und 90% aus der Haftsumme des RBV gedeckt wird. Nach drei Jahren erhöht sich die Verlustübernahme der HKG-Gesellschafter auf 30%. Seit der letzten Anpassung des RBV im Jahre 1983 haben sich die Kosten für die Stillegung (Beseitigung) der Anlage gegenüber den Ansätzen im RBV erhöht. Die Kosten für die Beseitigung der Anlage sind von 180 Mio. DM aus dem derzeitigen RBV auf Basis eines Gutachtens heute auf ca. 450 Mio. DM angewachsen.

Durch das Auftreten neuer von außen an das Projekt THTR-300 herangetragener Risiken, die zu Stillständen führen können, sind die Gesellschafter der HKG der Ansicht, daß die Haftsumme von 450 Mio. DM nicht ausreichend ist. Alle diese neuen Risiken haben sich gegen Ende des Jahres 1988 konkretisiert. Sie sollen im folgenden kurz angerissen werden:

– *Stillstandsrisiko Brennelementversorgung*

Die derzeit durch Nukem gefertigten Brennelemente für den THTR-300 reichen für eine Betriebszeit bis Ende 1991. Zum 31. 12. 88 hat die Nukem mit der Produktion von kugelförmigen Brennelementen aufgehört. Eine termingerechte Anschlußfertigung ist derzeit nicht sichergestellt.

– *Stillstandsrisiko Brennelemententsorgung*

In der Betriebsgenehmigung für den THTR-300 ist nach dem Erreichen von 600 Vollasttagen, das wäre ein Zeitpunkt Anfang 1990, nachzuweisen, daß Zwischenlagermöglichkeiten für Brennelemente extern gesichert sind und daß weiterhin die Transportbereitstellungshalle zur Einlagerung von schwachaktiv-Abfällen auf dem Gelände des THTR-Standortes genehmigt ist.

Beide Bedingungen sind derzeit noch in Arbeit und nicht abgeschlossen.

– *Stillstandsrisiko Dauerbetriebsgenehmigung*

Die derzeitige Betriebsgenehmigung für den THTR-300 erstreckt sich auf 1100 Vollasttage, d. h. sie läuft Mitte 1992 aus. Für die daran anschließende Dauerbetriebsgenehmigung ist ein Genehmigungsverfahren noch durchzuführen. Es ist derzeit nicht abzusehen, unter welchen Auflagen und Kriterien dieses Genehmigungsverfahren durchgeführt wird. Auf jeden Fall wird die zuständige atomrechtliche Genehmigungsbehörde vor Erteilung einer weiterführenden Betriebsgenehmigung eine detaillierte Sicherheitsprüfung wahrscheinlich erwägen.

Diese Umstände haben die Gesellschafter der HKG veranlaßt, an die Partner des Risikobeteiligungsvertrages heranzutreten und eine Aufstockung des Vertrages auf die Haftsumme von 1,1 Mrd. DM zu fordern. Bei der Beurteilung der Erhöhung der Haftsumme muß immer wieder deutlich gemacht werden, daß es sich hier um die Abdeckung eines finanziellen Risikos handelt, das nicht mit Sicherheit eintreten muß. Würde z. B. der THTR-300 im Rahmen einer Langfristplanung mit 70,4% Verfügbarkeit betrieben werden können, so würde die Haftsumme maximal mit 340 Mio. DM beansprucht werden.

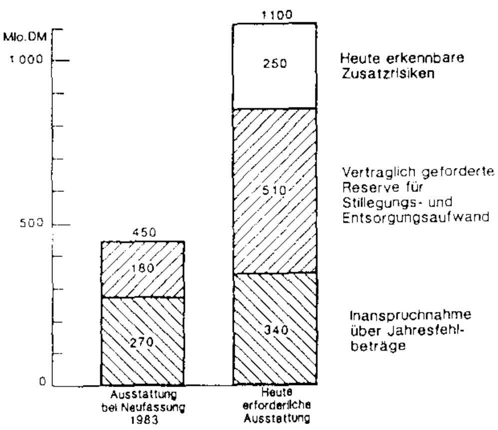


Abb. 9: Ausstattung des Risikobeteiligungsvertrages.

Es ist also eindeutig, daß nicht sicherheitstechnische Gründe diese Aufstockung der Risikobeteiligung notwendig machen, sondern daß es ausschließlich von außen herangetragene wirtschaftliche Faktoren sind, die die notwendige Erhöhung fordern.

In Abb. 9 sind die Bestandteile des Risikobeteiligungsvertrages und die von den HKG-Gesellschaftern für notwendig erachtete Erhöhung dargestellt.

4. Schlußwort

Die Hochtemperaturreaktorlinie genießt derzeit weltweit erhöhte Aufmerksamkeit und wissenschaftliches Interesse. Zusammenarbeitsverträge der deutschen Industrie mit der Sowjetunion und Verträge mit der Volksrepublik China sind hier nur ein Gradmesser für den Aufwind, der die Hochtemperaturreaktorlinie auf internationalem Gebiet trägt.

Die technologischen Möglichkeiten, die in der Hochtemperaturreaktorlinie stecken, sind nach wie vor technisch und wissenschaftlich unbestritten, sei es die Erzeugung von Wasserstoff oder die Bewältigung des CO_2-Problems. Die in Studien konzipierte Symbiose von nuklearer und fossiler Energie über den Hochtemperaturreaktor ist eine Option für eine umweltschonende Energieversorgung für die Zukunft. Bei dieser weitgespannten Zielsetzung ist zu fragen, ob eine neuartige Technologie Bestand haben kann, wenn der Betrieb der weltweit einzigen Referenzanlage aus finanziellen Erwägungen nicht fortgeführt werden kann. Der THTR-300 war als Bindeglied zwischen dem AVR und einer kommerziellen Hochtemperaturreaktoranlage gedacht. Wenn das Bindeglied fehlt, ist ein Bruch in der technologischen Weiterentwicklung vorprogrammiert.

In der heutigen Diskussion über eine langfristig ausreichende und umweltschonende Energieversorgung werden häufig neue Technologien unter den Begriffen Solarwasserstoff, Windenergie, Biomasse usw. aufgeführt.

Im Vergleich zum Entwicklungsstand dieser Energieträger ist mit dem THTR die großtechnische Reife der HTR-Linie erreicht. Er ist damit den Entwicklungen auf dem Sektor der sog. Alternativenergien um Jahrzehnte voraus.

Wir werden unter dem Gesichtspunkt zu entscheiden haben, ob es sinnvoll ist, diese Entwicklung zu unterbrechen oder abzuwarten, bis andere sie uns liefern können.

DK 621.039.524.2.034HTR

Eleventh International Conference on the HTGR

Dimitrovgrad, 19 - 20 June 1989

THTR Commissioning and Operating Experience

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1. Introduction

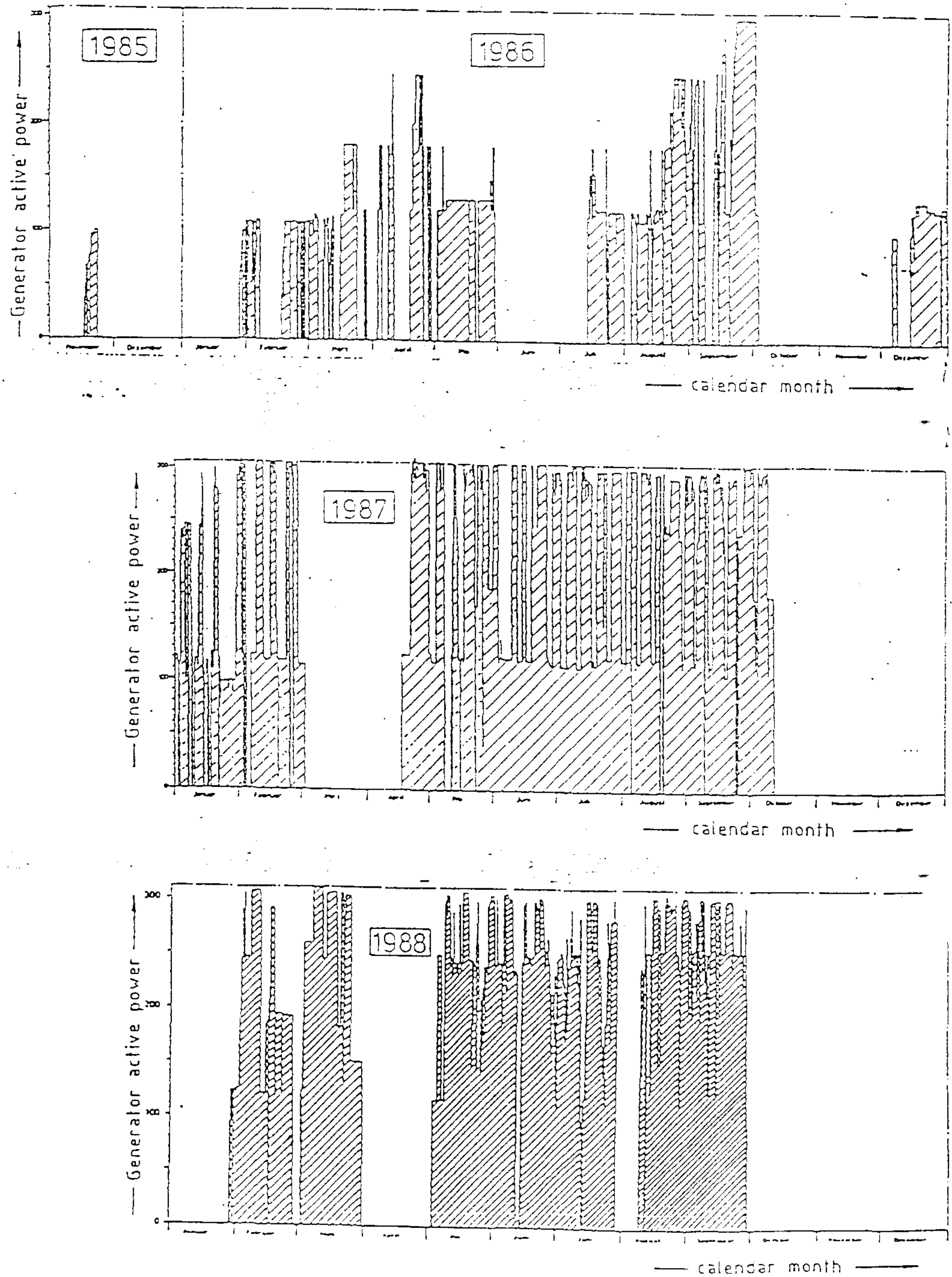
The Thorium-High-Temperature Reactor THTR 300 is the prototype power plant for a medium-sized pebble bed reactor. The commissioning period up to handover of the plant to the user was marked by the following milestones which characterize the extensive and time-consuming commissioning program:

Sept 13, 1983	first criticality
Nov 16, 1985	first synchronization to power grid
Sept 23, 1986	first 100 % power operation
Juni 1, 1987	completion of nuclear trial operation and handover of the plant to the user company HKG

Until today the plant was in operation 16 410 h and has generated 2 891 068 MWh. The time availibilty has been 61 % in 1987 and 52 % in 1988.

The diagram of the previous operating history is a spike curve which is characterized by frequent power changes and several prolonged plant downtimes.

HKG THTR 300	- 2 - Electric power output diagram during operating phase between Nov. 16, 1985 and Dec. 31, 1988	

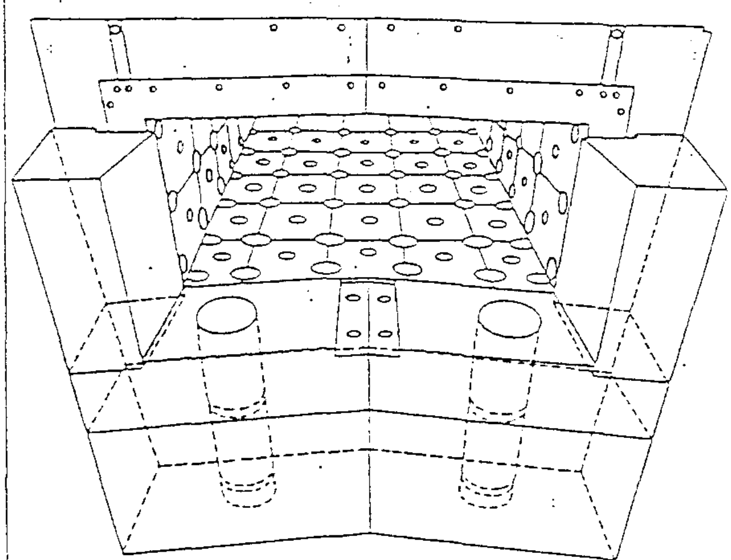


The power changes were initially caused by difficulties arising in the withdrawal of spherical elements from the reactor. In the beginning of the plant operation spheres could be withdrawn only at reduced plant power, since only with a reduced helium mass flow which is partly passed in countercurrent to the fuel element flow direction for cooling the fuel element discharge pipe, withdrawal of the spherical elements was possible. This defect was eliminated during the 1987/88 plant inspection. Further downtimes resulted from jamming of spheres in the singulizer disk of a helical damaged-spheres separator in the refuelling system and from the necessity to exchange the casks which collect the damaged spherical elements. Finally power reduction was repeatedly required in summer 1988 to keep the exhaust air temperature in those parts of the reactor hall within the permissible limits, which accommodate the components of the steam/feedwater circuit, e.g. the steam generator ring rooms. On September 29, 1988 the power plant was shut down for the scheduled 1988 inspection.

On the occasion of a routine inspection, we inspected - as a precautionary measure - a hot gas duct, the duct through which the hot helium passes from the reactor core to the steam generator. The figure shows an internal view of a hot gas duct with its rectangular passage through the graphite side reflector. The lower graphite blocks of the hot gas duct are each fixed to the respective carbon block by a graphite dowel. In the outer wall of the side reflector these dowels are positioned in bore holes penetrating the blocks. The figure shows the front part of the metallic section of the hot gas duct showing the inner insulation which consists of metal foil blankets, covered by 30 cm x 30 cm cover plates which are each held down and fixed by 4 corner bolts and 1 central bolt. After the inspection of the first duct had revealed damage on some attachment fixtures (central bolts), we decided to inspect all the 6 ducts, and it was detected that out of the approx. 2600 bolts 35 bolts heads had come off. In addition it was detected that several graphite dowels installed for holding in position the lower outer blocks of the hot gas duct had been displaced.

The damage has been thorougly analysed and the following causes have been determined: The bolt heads failed due to stresses which had concentrated in the range of the bolt head as a result of differential thermal expansion of the materials of the metal foil insulation consisting of 18 layers and the structure of the attachment fixture bolts. In addition a reduction in the material ductility as a result of thermal neutron irradiation in the temperature range above 500 °C was observed.

HKG THTR 300	Front part of the hot gas duct	



After thorough analyses we and the plant supplier have jointly come to the result that further operation of the THTR 300 is justified in spite of the existing damage.

Since the damage is essentially concentrated on the central bolts, the thermal insulation in the metal part of the hot gas duct is held down by the corner bolts as before. Thus the functional capability of the thermal insulation is safely ensured also in the present situation. In case that parts of the insulation were detached afterall, this would be detected by the operational monitoring of the process parameters mass flow and pressure loss. We have, however, the intention to observe the situation in future by inspecting the hot gas ducts in shorter intervals.

During the overall operation until shutdown of the power plant on September 29, 1988 for the 1988 inspection the plant has generated 2 891 068 MWh. For generating this electrical gross output the plant had to be operated for 423 full power days including the commissioning period.

In the following the main results of the plant operation are presented.

HKG THTR 300 | THTR - Operating experience

Safety-relevant conclusions

Operation

Normal operation	Shutdowns Plant outages	Inspections
Design	Shutdowns / Decay heat removal	Radiological protection data
Core dynamics	Shutdown rods	Graphite dust
Temperature distribution	Penetration isolation valves	Activity
Refueling / spheres damage	Emergency power supply	Inspection manual
Coolant gas activity		
Non-active impurities in the coolant gas		
Thermodynamics		
Measuring methods		

The evaluation of the operating data can be subdivided into three sections:

- power operation,
- plant downtimes including shutdown procedures, and
- inspections.

From all three sections important information has been obtained which will be discussed in the paragraphs below.

2. Evaluation of Operating Experience

2.1 Design Data and Power Operation

HKG THTR 300 — Comparison of measured and calculated operating data at 100 % power output on February 9, 1988

	Unit	Measured valve	Calculated valve
Thermal power of core and internals	MW	756	755
Circulator speed	min^{-1}	5407	5380
Helium flow rate	kg/s	48,26	49,12
Feedwater flow rate	t/h	151,6	151,7
Mass flow through reheater	t/h	144,7	144,1
Hot gas temperature at SG inlet	°C	750,3	750
Cold gas temperature at SG outlet	°C	246,8	246,3
Main steam temperature	°C	534,2	535
Main steam pressure	bar (abs)	186,9	186,7
Reheat steam temperature	°C	530,6	527
Reheat steam pressure	bar (abs)	48,5	48,4
Generator active power	MW	304,3	304,3
Cooling water temperature	°C	26,7	26,5

The design data which had been specified for the THTR power operation have been confirmed by measurements during operation. This fact is not evident for a prototype plant. It shows that the theoretical bases for the design of hightemperature reactors are available. From the point of view of safety engineering the following aspects are interesting in this context:

2.1.1 Core Dynamics, Control Behaviour, Power Distribution

The core power output can be controlled at all power levels and under all core conditions without any problems. Power changes are possible in the range between 40 % and 100 % power output in any steps desired. Power changes are performed by ramps of 8 % per minute within the main operation range.

The previous operation has, on the one hand, confirmed the design values for the core and the operational and safety procedures and, on the other hand, it has verified the functional capability of the control equipment and the components of the primary and secondary system.

During power changes the electrical unit output, the main steam pressure, the main steam temperature and the cold gas temperature are controlled. The control variables for this purpose are the helium mass flow, the position of the reflector rods and the feed water quantity.

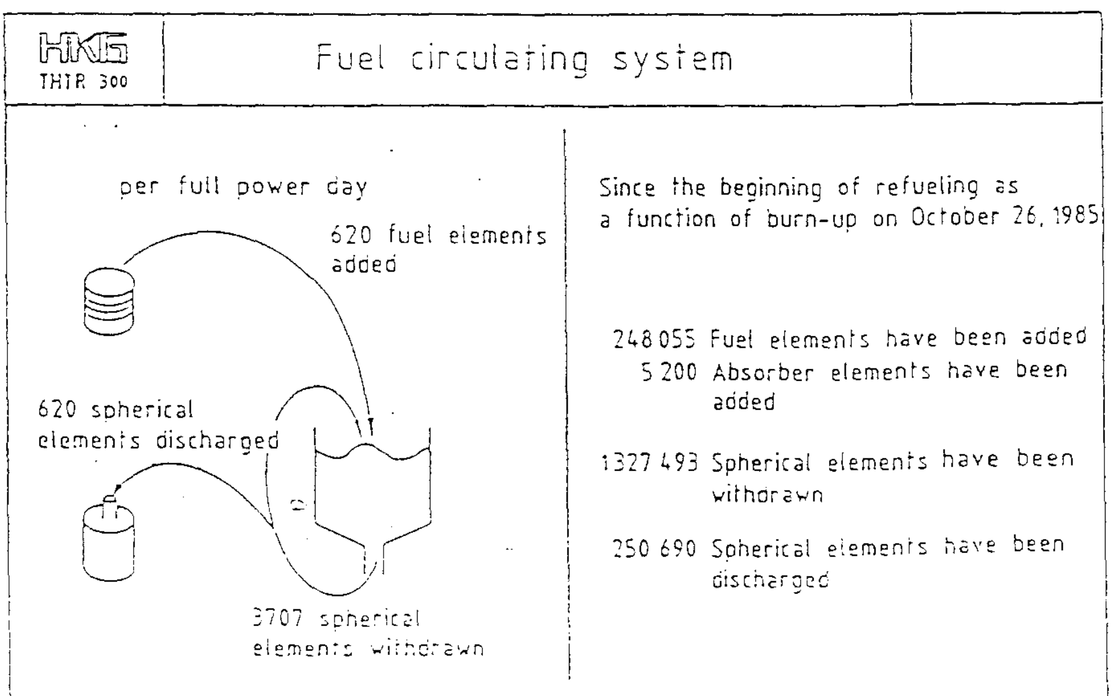
The control concept especially controls also upset operating conditions, such as the automatic power reduction to about 70 % in the event of failure of one circulator turboset, load rejection to plant auxiliary power, or turbine scram. Instabilities of the core behaviour never occur during such control porcedures, nor fluctuations of the power distributions (e.g. xenon fluctuations). The temperature coefficient of the THTR is negative in all power ranges. It is between Γ = - 12 mN/K and - 4 mN/K. For demonstrating the negative feed-back, the power and temperature curves were recorded at a thermal power of several per cent in the course of a controlled intentional "return to criticality" of the reactor. The curves showed the expected slow changes of power and temperature thus confirming the design calculations. The inherent safety of the THTR and its "good-natured" control behaviour has thus been verified experimentally.

2.1.2 Temperature Distribution in the Core

The requirements for the temperature distribution in the core result from the maximum permissible temperature of the fuel elements as well as from the maximum permissible insertion depth of the incore rods, which, in turn, results from the rod temperature which must not exceed the specified design values.

The permissible fuel element temperatures can be observed without any difficulties by manoevering the incore rods and the reflector rods so as to prevent power concentration in the lower core region. Another possibility of indirect control of the permissible temperatures is obtained by monitoring the hot gas temperature in the bottom reflector. Observance of the maximum incore rod tip temperatures is more difficult. For this purpose it is necessary to perform design calculations on the temperature and power distribution in the core in parallel with the operation. During the running-in phase these distributions continously change. Due to potential uncertainties in the calculated maximum rod tip temperatures in practice conservative safety margins for the permissible insertion depth are required. This sets a limit to the possibility of using the incore rods. Due to high excess reactivities, which occur for example after prolonged plant downtimes, relatively deep insertion of the incore rods is required also during power operation. This may result in power restrictions for a limited period (approx. 2 weeks) to ensure that the maximum incore rod tip temperatures are not exceeded.

2.1.3 Refueling and Damage of Spherical Elements

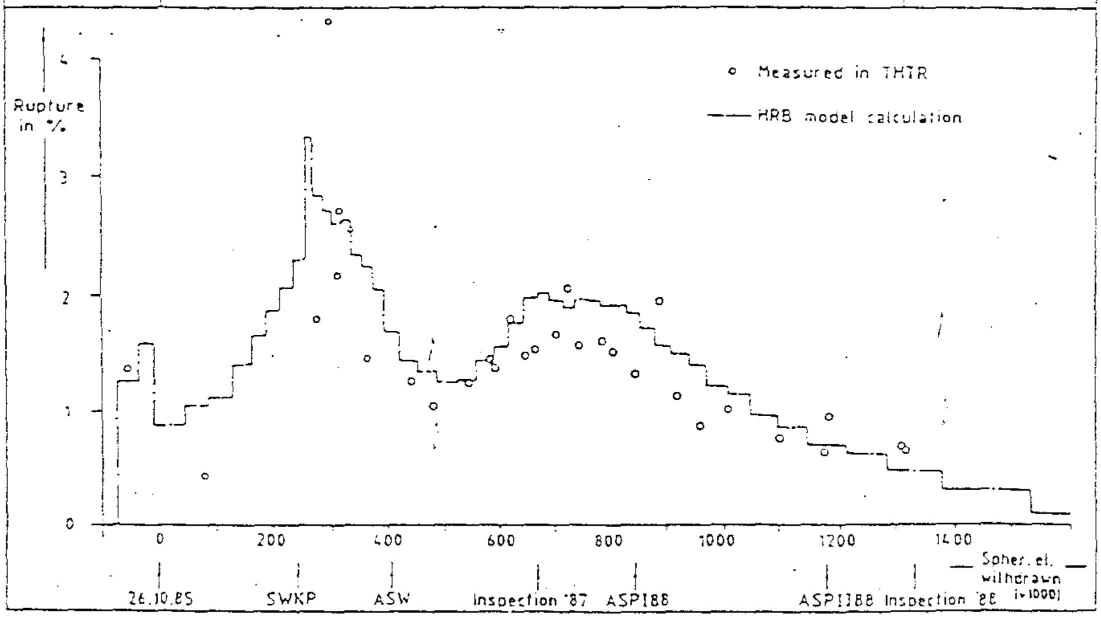


A special characteristic of the THTR is continuous refueling. 3707 spherical elements are withdrawn from the reactor core per full power day. 620 spherical elements are discharged from the circuit, the rest is returned into the reactor core. The 620 spherical elements withdrawn are replaced by 620 fresh fuel elements.

Up to 29.09.1988 a total of 1,3 million spherical elements from the core have been drawn off, from this figure 235 000 spherical elements taken away and replaced by a correspondant number of fresh spherical elements. Essential for the safety of reactor operation is the correct, i.e. refueling of the reactor core according to design. The spherical elements are added to the core according to a refueling strategy calculated in advance. This procedure has proven to be successful in previous refueling practice. The subsequent calculations will, however, require new reference data for calculations to actual measured values. In this aspect the calculation model can certainly be further improved, e.g. by using measured values on the flow behaviour of the spherical elements and the measured burn-up spectrum of the fuel elements discharged. The observance of the safety-relevant design data such as excess reactivity, power distribution and temperature distribution and, thus, the guaranty of the rod worths does not pose any problems. These data are continously verified experimentally and are thus ensured at any time independent of the calculations.

The practical performance of the refueling procedure met with some difficulties. They had no safety relevance and were eliminated as was described earlier. This applies as well to the unexpected high number of damaged sherical elements, which were sorted out by the helical scrap separator during withdrawal of the spherical elements from the reactor core. Up to the present time 10 casks have been filled with approx. 17.000 damaged spherical elements. The share of damaged spherical elements in the total amout of spherical elements withdrawn was about 1.5 % in the beginning of the refueling operation and is continously drecreasing. Recently the rate reached 0.6 %.

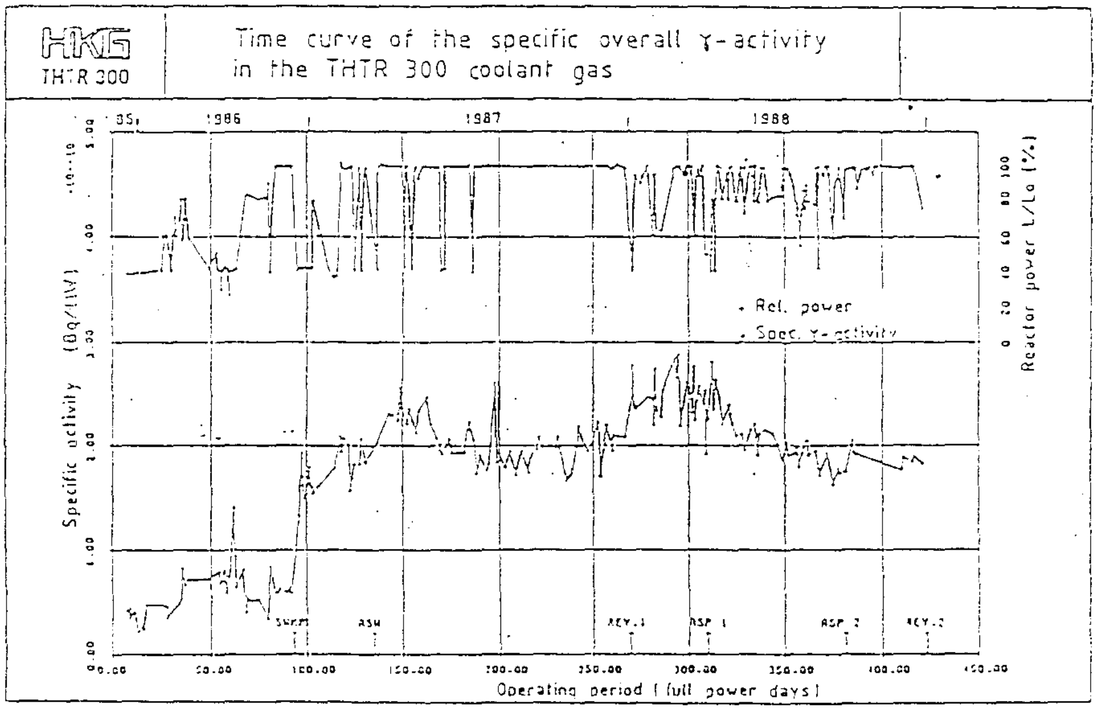
HKG THTR 300 | Development of spheres rupture during THTR operation



A model calculation was performed based on the assumption that the damage was mainly caused by frequent and deep insertion of the incore rods during the THTR commissioning phase. This assumption has been confirmed by the agreement with the experimental data. Since the damage in most cases only concerns the graphite shell in which the fuel is embedded, i.e. the coated fuel particles in the damaged fuel elements are intact in their greatest part, retention of the fission products is ensured as before. The flow behaviour of the spherical elements in the reactor core and the insertion of the incore rods is not impaired by the damaged spherical elements. Therefore the damage of spheres has no safety relevance.

Elimination of the disturbances of the process described above requires, however, a great effort, e.g. the exchange of casks for damaged spherical elements requires complete depressurization of the prestressed concrete reactor vessel. Therefore it is intended - in particular also for economic reasons - to change the mode of manoevering the incore rods so that damage of further spherical elements is reduced to a minimum, R + E work is carried out for an evaluation of the mode of spheres rupture and the mechanical behaviour of the pebble bed in order to obtain an exact analysis of all the effects occurring.

2.1.4 Coolant Gas Activity in the Primary Circuit



The coolant gas activity of the THTR does not exceed the expected values. The overall development of the coolant gas activity is shown in the figure. As had been expected, the coolant gas activity increased during the commissioning phase with increasing reactor power reaching almost constant values at continuous full power operation. It remains clearly and constantly below the design values. As for the AVR, the fission product retention capability of the fuel elements has thus been confirmed also for the THTR in power operation.

2.1.5 Non-Radioactive Impurities in the Coolant Gas

The impurities contained in the coolant gas, H_2O, CO_2, H_2 and in some rare cases also traces of O_2, which have an oxidizing effect on graphite, have removed 65 kg of carbon from the spherical elements and the graphite internals up to the present time.

This carbon quantity has to be considered in relation to the overall carbon inventory of the core which is 728 tons. The helium purification system of the THTR has been able to cope with all concentrations of impurities without any problems. The primary circuit with its auxiliary systems does not pose any problems with regard to chemical and radiochemical parameters.

HKG THTR 300 — Impurities in the THTR 300 Coolant Gas

		undisturbed	after injection of ammonia
H_2O	µbar/vpm	≤ 0,5/< 0,01	< 2/< 0,05
H_2	µbar/vpm	30 / 0,8	up to 4000/ up to 100
CH_4	µbar/vpm	4 / 0,1	up to 200/ up to 5
CO_2	µbar/vpm	8 / 0,2	
CO	µbar/vpm	16 / 0,4	
N_2	µbar/vpm	< 4 /< 0,1	up to 2000/ up to 50
O_2	µbar/vpm	n.n.	
Ar	µbar/vpm	n.n.	
Kr, Xe	Bq/m³ i.N.	3 × 10⁸	
³H	Bq/m³ i.N.	5 - 20 × 10⁵	
¹⁴C	Bq/m³ i.N.	100 - 300	
Aerosole	Bq/m³ i.N.	< 100	
NH_3	µbar/vpm	n.n.	up to 3800/ up to 100

In general the impurities in the coolant gas (H_2O, H_2, CH_4, CO_2, Co und N_2) are low. During steady-state operation they sum up to a maximum of 80 µbar (= 2vpm). Only during start-up the contents of hydrogen and nitrogen may rise by NH_3 decay up to the level of a few mbar. During shutdown of the reactor ammonia is fed into the core to reduce the friction factor of the incore rods in the pebble bed core.

2.1.6 Thermodynamic Parameters of the Primary System

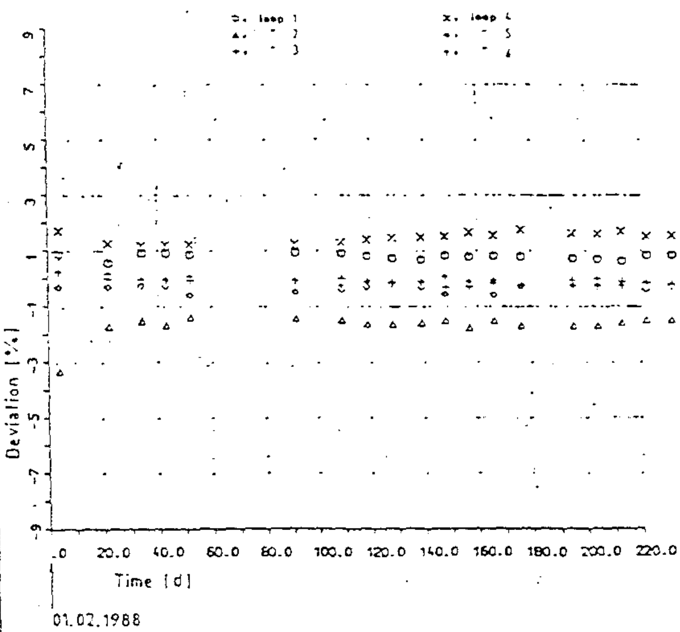
In addition to the operational data quoted at the beginning of this chapter, which are directly included into the power calculation, a number of additional data are measured to describe the primary system. This has shown that the bypass of the helium mass flow is higher than expected. It is defined to be being 18 % instead of 7 %, which had been expected. The core outlet temperature which has therefore to be higher by about ten per cent is below the design values for fuel elements and graphite internals even at full power operation, thus it does not pose any problems. In connection with the damage of the attachment fixtures of the hot gas duct insulation reference should briefly be made to another group of thermodynamic data of the primary system. Apart from the temperatures, these are the helium mass flows and the pressure losses of the 6 steam generator/circulator units. These data are continuously recorded and evaluated in the THTR. In addition, derived values such as e.g. the pressure loss coefficients are continuously determined.

These values are observed, on the one hand as mean values of all the six steam generator/circulator units for detecting uniform changes in all the 6 hot gas ducts and, on the other hand, they are evaluated as relative deviations from the mean value for determining irregular changes in individual hot gas ducts.

Evaluations performed during the latest year of operation have shown that changes of the above-mentioned data in the primary system are detectable with an accuracy of 1 %.

HKG THTR 300	Pressure losses in primary system	

Time-dependent deviations from mean pressure losses in the primary system (mean value of approx. 9 days)



These statements show that in addition to the measured values for the reactor core itself also the thermodynamic data of the primary system are stable and reproduceable. Therefore safety-relevant changes which may occur can be detected safely and early enough. Thus it is demonstrated that the design has been confirmed and that the components such as e.g. the helium circulator and the steam generator have proven their functional capability.

2.1.7 Measuring Methods

Another condition for safe plant operation is the correct acquisition and reliable processing of all the measured values required for plant safety and plant operation. The instrumentation concept of the THTR - including the elimination of incore instrumentation - and the practical application of the measuring facilities have proved to be efficient. This applies also to the special measuring facilities necessary for a prototype plant, such as neutron flux instrumentation, temperature measurements of the metal and ceramic internals, instrumentation for measurements in the helium circulators and steam generators, spheres counting equipment and burn-up measurement facility. The information on the plant required for safety reasons has been available at any time.

2.2 Conclusions from Shutdown Procedures and Plant Downtimes

2.2.1 Shutdown Procedures, Decay Heat Removal Systems

As shown earlier in the operational diagram, the THTR has been shut down relatively frequently during the commissioning phase and the power operation. Part of the shutdown procedures were scheduled and maintenance and repair measures, especially in-service inspections. In addition, especially during the trial operation, the excitation of the two automatic shutdown procedures was repeatedly triggered by the Plant Protection System: reactor scram (11 x, 4 of them as tests during the commissioning phase) or Decay Heat Removal 45 procedure (20 x). The causes were a too narrow adjustment of the limiting values, (this was eliminated during the comissioning phase), defective instruments, errors in detail planning of release logics and operator errors. The greatest part of the releases were not required for safety reasons. In all the shutdown procedures heat removal from the core and from the internals was effected according to the design principles. Minor irregularities in the procedures were never of safety relevance and were eliminated in the course of the commissioning phase. Experience has shown up to now that the decay heat removal systems which are partly identical with operational systems have a sufficient availability, an appropriate process design, and have proven their fuctional capability in practice. In the course of the overall operating period including the shutdown procedures several hundred measuring data are being recorded and evaluated in sections by the continuously operating long-term recording program of the process computer system. The "service life consumption" of the steam generators and the associated piping amounts only a few percent. Only some solid parts which could be exchanged, have reached a life time consumption of about 10 % up to now. Assuming a "normal" further power operation, there are no restrictions or safety-relevant problems to be expected from today's point of view for a further long-term operation.

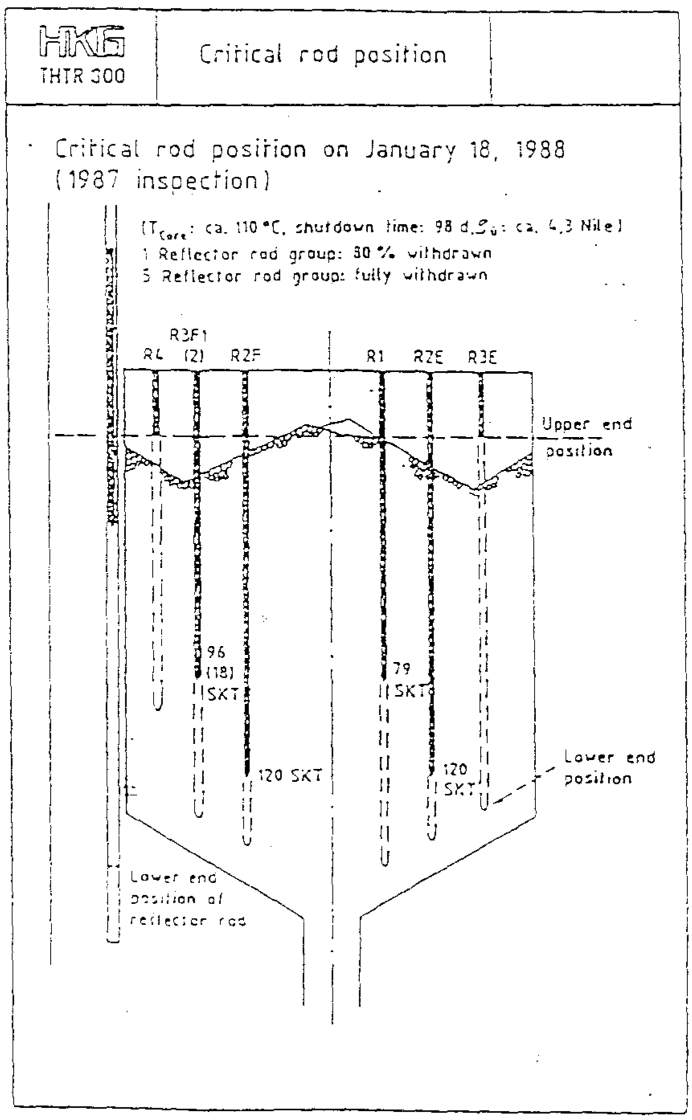
The cooldown procedure "Heat Removal 5" designed to come into action in the event of major disturbances, or the measures for resumption of heat removal after a prolonged interruption of decay heat removal (LUNWA) have not come into action up to now. Therefore it can be stated that the previous operating experience does not give rise to any new safety requirements with regard to detection of disturbances and release and sequence of cooldown procedures. It is currently being investigated, whether there is a possibility of simplifying the excitation logics of the Plant Protection System and improving the sequence of the cooldown procedures. The use of the absorber rods could be reduced, as will be demonstrated in the section below.

2.2.2 Shutdown Systems

The THTR ist equipped with two independent shutdown systems, the reflector rods (6 groups of 6 rods each) and the incore rods (7 groups of 6 rods each). Four reflector rod groups represent the shutdown system, the incore rods are inserted for long-term shutdown. In order to ensure sufficient subcriticality, it was claimed that during the running-in phase in the event of reactor scram in addition to the reflector rods a group of incore rods (group R3E) should be automatically inserted by the long-stroke pistion drive.

This claim has proven to be unnecessary at an early date, since it has been demonstrated during the commissioning phase on the occasion of scram tests from power operation that the reactor is still subcritical after 30 minutes by insertion of the reflector rod shutdown groups alone without additional insertion of the incore rod group and that the reactor remaine subcritical over the period of xenon build-up. This situation is maintained even under the most adverse conditions by definition (start-up after prolonged standstill, no xenon, low helium temperature). The claim for automatic insertion of an incore rod group in the event of reactor scram can therefore be eliminated.

For automatic long-term shutdown it was envisaged to insert all the 42 incore rods to their lower end position. Also for these conditions it has been repeatedly demonstrated that the measures for long-term shutdown of the reactor need not be applied to the extent originally envisaged. Even with the boundary conditions of maximum excess reactivity, low helium temperature, long-term subcriticality after prolonged operation, i.e. with full protactinium conversion, it is sufficient to insert 4 incore rod groups to a depth about 1 m above the lower end position. The figure below shows as an example the critical rod position after the 1987 Inspection.



The long-term shutdown system was designed too conservatively so that it is overdimensioned. For this reason the incore rods are only inserted in that number and depth which is required for safety reasons to ensure sufficient subcriticality during prolonged plant downtimes. Since the incore rods have to be considered to be the cause of the increased rate of damaged spherical elements, it is expected that this measure will result in a marked reduction of spheres damage.

From a process design aspect the incore rods and the reflector rods have proven to be efficient safety systems. The dropping times of the reflector rods corresponded to the design values, the insertion times and insertion depths of the incore rods when automatically inserted by the long-stroke pistons have ensured subcriticality of the reactor at any time.

2.2.3 Penetration Isolation System

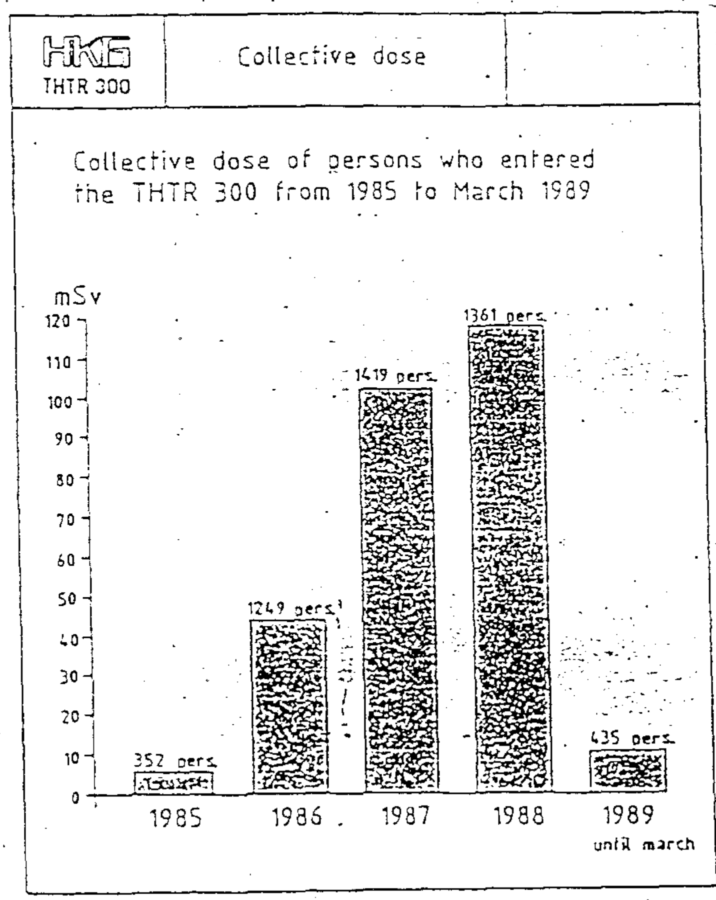
The penetration isolation system consists of shut-off valves equipped with diverse drive systems. Each pipe penetrating the PCRV and carrying primary gas is shut off by these valves to ensure activity confinement. Each line is equipped with two valves which close in case of demand upon excitation by the Plant Protection System. In the course of the THTR operation no disturbances have occurred up to now which would have required an activation of the penetration isolation system. Modifications or backfitting of these active engineered safety systems has not become necessary as a result of the previous operation.

2.2.4 Emergency Power Supply

The only case of emergency power supply occurred in the beginning of the commissioning phase. It was initiated by the attempt to switch over the electrical feed water pump form a supply line to a redundant line within approximately 1 second. This resulted in shutdown of the supplying transformer. This distrubance gave rise to several modifications of details, optimizations and definitions of process design. In principle, however, the concept for detection and activation of the emergency power supply system has been confirmed.

2.3 Inspections

2.3.1 Radiological Protection Data Referring to Plant Personnel



Radiation exposure of the THTR plan personnel is very low. The radiation exposure values for the previous operating period are indicated in the figure. The data demonstrate that the plant concept with a prestressen concrete reactor vessel has proven to be successful.

This fact applies also to the conditions prevailing in the event of maintenance or repair work on components of the primary. By using special disassembly facilities and tools and observance of the sequences of work planned in detail, these activities can be carried out with a low collective as well as single dose. When in April 1988 repair work on one of the helical damaged-spheres separators of the fuel circulating system had to be performed, the overall collective dose was 2.71 mSv and the maximum single doses were less than 0.2 mSv. We assume that such favorable values can be maintained also in future.

2.3.2 Graphite Dust

It has been detected on piping carrying primary gas and on components disassembled from the primary system that surfaces of components which are part of the helium circuits and the fuel circulation system are contaminated by radioactive graphite dust (mass deposition about 1 mg/cm²). The specific activity of the dust was determined to be 2×10^8 Bq/g at a maximum. It is mainly caused by the radionuclides Co-60, Nb/Zr-95, Hf-181 und Pa 233. The overall quantitiy of graphite dust detected corresponds to the expected weigth loss of the spherical elements during circulation by abrasion. Under the aspect of radiological protection it does not pose any problems for disassembly work. The only effects of the graphite dust on the operation of systems were noticed in the beginning of the commissioning phase, when individual moisture sensors in the moisture monitoring system of the steam generators failed. This source of failure was eliminated by installing simple dust filters upstream the sensors. It can thus be stated that the graphite dust does not pose any problems, neither with regard to operation nor to safety.

Measurements on piping carrying primary gas have shown that also in the event of a depressurization accident the graphite dust does not cause an increased release of activity.

2.3.3 Activity Release with Vent Air

HKG THTR 300 — Activity release

Activity release with exhaust air
1988

	Release in Bq	Licensed annual limit value in Bq	Release in % of annual limit value
Inert gases	2,504 E11	6,66 E14	6,037 %
Aerosols	8,968 E07	3,7 E08	24,2 %
Jodine	1,086 E07	3,7 E08	2,9 %
H3-Control area	3,471 E12	8,14 E12	42,8 %
C 14	2,682 E10	7,4 E12	0,36 %

The activity release with vent air measured in 1988 is presented in the figure. It was no problem during power operation to remain below the low limiting values specified in the THTR license, because at that time only minor repair work was performed on components of the helium circuit.

To reduce the release of radioactive aerosols to the environment, i.e. the release of activity carried by graphite dust, it has proved necessary in the course of the commissioning phase to provide all exhaust paths with filters. This has been done and has proved to be a successful solution.

Contrary to normal operating conditions, during inspections the PCRV is often depressurized and open to perform some work on integrated components. To maintain a specified flow direction, the PCRV is kept under a slightly negative pressure during the performance of the above-mentioned repair work. For this purpose a small partial quantity of the helium inventory is withdrawn from the PCRV and released to the atmosphere with the vent air. Since the graphite internals still contain tritium after depressurization, which in case of moisture enters the gas phase via exchange reactions, the gas mixture withdrawn form the PCRV has to be passed through catalysers and a molecular sieve before it is released to the atmosphere. By this measure it is ensured that even in the event of complete ventilation of the PCRV no safety problems will arise.

3. Experience Expected form Further Operation of the THTR-300

A further operation of the THTR-300 is expected to furnish essential know-how in addition to the present operating experience and would thus allow to come to a valuable completion of the research contract. It is especially expected by us that it will be possible to extend and confirm by experiments the know-how on core design, spheres damage rate, and the activity release from the spheres.

Another objective is the verification of the long-term performance of the prototype components. The hot gas duct is an example which is significant at the present moment, but also the long-term behaviour of other prototype components such as shutdown rods, PCRV and graphite internals is of great interest.

A further task which could be persued during a further operation of the THTR is the development of disassembly and repair equipment for the components installed within the PCRV. The previous operation has demonstrated that the problem of accessiblity is of utmost importance to the operating company and that the development of disassembly equipment is urgently required. In our opinion it is another task of a prototype to verify the easy repairability of High-Temperature Reactors.

For initiating these tasks it is, however, necessary to obtain a new definition of the financial basis for the THTR-300 project.

4. The Risk Participation Contract and Covering of Financial Risks

As early as in 1971 the partners cooperating in the THTR-300 project had realized that because of the prototype character of the THTR-300 and the research objectives pursued with this reactor it would not be possible to achieve a commercial operation of the plant from the very beginning. For this reason a risk participation contract was negotiated and concluded already at the beginnig of the project earmarking a liability sum of DM 450 million to cover the economic risks of the plant operation and the decommissioning risks of the plant operation and the decommissioning costs. Two thirds of this sum was furnished by the Federal Government and one third by the Federal State Government of North Rhine Westphalia. DM 270 million are reserved for compensating losses from plant operation, and DM 180 million are presently envisaged for decommissioning of the plant.

It is further stipulated in the contract that during the first 3 years 10 % of the operating deficit is covered by the HKG partners and 90 % is furnished form the sum guarranteed in the risk participation contract.

After three years the share assumend by the HKG partners increases to 30 %. Since the latest up-dating of the risk participation contract in 1983 the costs of decommissioning (dismantlement) of the plan have increased compared to the costs earmarked in the risk participation contract. Based on an expert opinion the costs of dismantlement of the plant, quoted at DM 180 million in the existing risk participation contract, have now increased to about DM 450 million.

As a result of new risks affecting the THTR-300 project from external sources, which might result in plant outages, the HKG partners are of the opinion that the guaranteed sum of DM 450 million is not sufficient. All these new risks came up in concrete form late in 1988.

In the following they will be briefly characterized:

Risk of Standstill due to Fuel Element Supply Problems

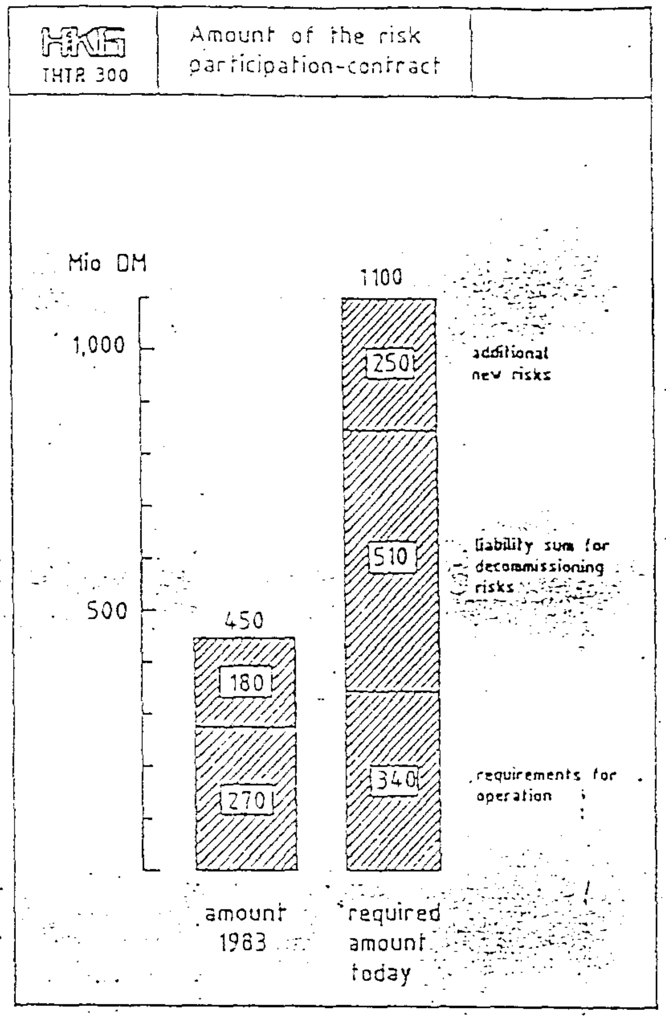
The fuel elements for the THTR-300 fabricated by NUKEM up to the present time are sufficient for an operating period until end 1991. On December 31, 1988 NUKEM terminated the fabrication of the spherical fuel elements. Continuation of the fuel element fabrication in due time is currently not ensured.

Risk of Standstill due to Fuel Element Disposal Problems

It is claimed in the operating license for the THTR-300 that it has to be given evidence at the end of 600 full power days, this would be some time early in 1990, that external intermediate storage facilities for the spent fuel elements are available and that the license has been obtained for the transport preparation hall for storing low-activity waste on the THTR plant site. Both conditions have not yet been met at the present time.

Risk of Standstill due to Problems Regarding the Permanent Operating License

The present operating license for the THTR-300 covers 1100 full power days, i.e. it will expire in mid 1992. The subsequent permanent operating license requires another licensing procedure. At the moment it cannot be predicted which will be the requirements and criteria of this licensing procedure. In any case there is a high probability that the competent nuclear licensing authority will perform a detailed safety investigation before granting a license for further plant operation. In view of this situation the HKG partners have asked the partners of the risk participation contract to increase the contractual amount guaranteed to DM 1.1 billion. In evaluating the increase of the sum guaranteed it has to be emphasized that it is intended to cover a financial risk which must not occur with certainty. If for example a further operation of the THTR-300 at an availability of 70.4 % was possible within a long-term program, the sum guaranteed would be claimed only to a maximum of DM 340 million.



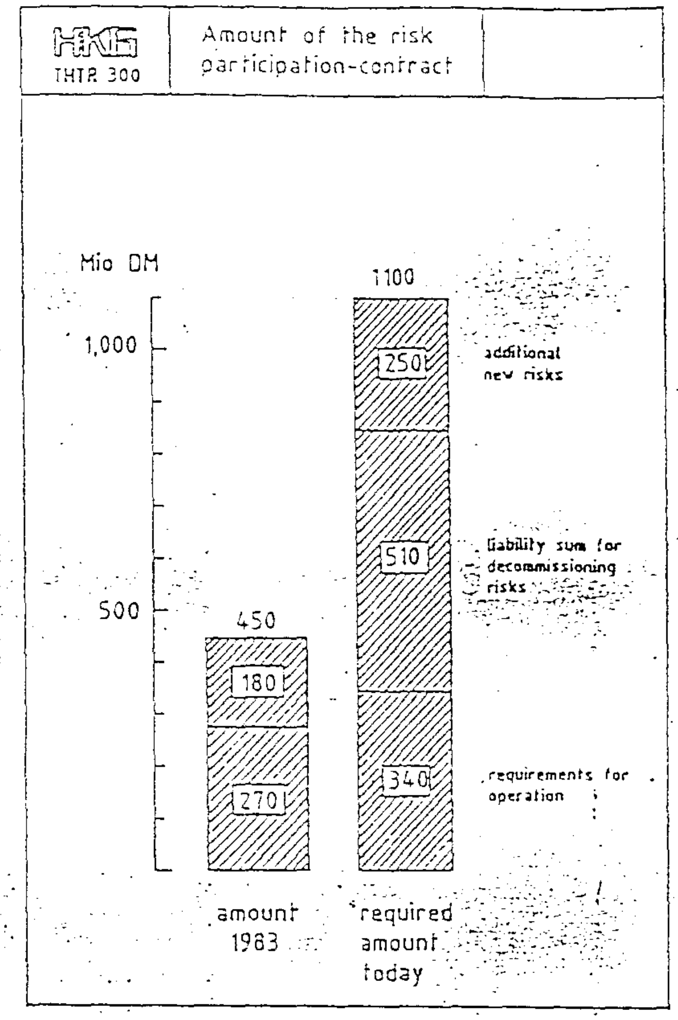
The figure shows the individual items of the risk participation contract and the increase considered necessary by the HKG partners.

5. Summary

The evaluation of the operating experience gained from the THTR up to now comes to an absolutely positive result. The principal design data have been confirmed.

The THTR-300 represents the successful connection link between the 15 MW_{el} AVR experimental reactor and a future commercial plant. On the basis of the present know-how obtained from the THTR operation another optimized high-temperature reactor can be designed and constructed thus representing a further step towards commercialization of advanced reactors.

It is evident that the necessity to increase the risk participation contract does not arise from safety considerations but exclusively from economic factors affecting the THTR from outside.



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Core Physics and Pebble Flow, Examples from THTR Operation

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Salzgitter

1 Introduction

To introduce myself: for 7 years I have been working at the THTR plant, starting with the commissioning till the end of operation. I left shortly after the decision to definitely shutdown the plant.

I have been involved in the evaluation of physical properties of the THTR-Core, the comparison of calculated design features and measured effects during the commissioning phase. And in particular I was involved in refuelling the core, which did take more attention than expected.

Since this is a long time ago, I do not have nice drawings, but more or less sketches with handwriting. I apologize for this, but I think the idea is the main point and I take efforts to present it clearly.

There has been no conclusive evaluation of the pebble flow within the THTR Core. There have been many experimental evidences on how and why the core moves, but an evaluation taking into account <u>all</u> the effects is missing. In fact the operation time of the THTR was too short to have definite evidence. We only had 423 full power days of load and were far away from the equilibrium core. This is really a disadvantage, for the pebble flow is the central question of having command over the dynamic core of moving and flowing contents.

So far for the introduction.

There are only 2 items I want to talk about, these are the pebble flow and how to keep it going from the physical point of view and its connection to the resulting gas temperatures. I leave out influence on absorber rod values, not touching bypass effects of core bottom cooling and the rest of core physics.

2 Gas temperatures

$\boxed{\text{Transparency 1}}$ Reactor vessel and core as model, $dm_{He}/dt \approx 300$ kg/sec

Transparency 2   Flow lines and core rods, to have an impression

R2 are control rods, exchanged form time to time, to avoid bending because of the pebble flow. Gas temperatures at the core outlet (mass flow downwards) are measured by NATE in bottom reflector at symmetric positions Transparency 3.

The aim for the outlet gas temperature is to have a flat profile. This is best for passing into the hot gas duct – with cold bypass from below and above than – where the gas goes through the mixing plate ahead of the steam generator.

It takes some time to really have a "stationary" state (because of Xe and Protactinium) with also similar rod positions in R2. But the conditions I used are rather stationary and there were repeated measurements.

A word how to characterize burn-up and core in general: We used AVLT/BVLT (abgebrannte Vollasttage or burned up full power days, beschickte Vollasttage or refuelled full power days). A full power day is a good measure for the burn-up of the core and usually there is a difference between the used power and the necessarily refilled pebbles. Which doesn't matter if the difference is not so large. It has influence on rod bending and in particular on the possibility of fast core rod insertion in emergency shut down. I leave that out.

Transparency 4   So you see in the upper sketch at 40 % of power that the temperature profile is bending with increasing burn-up; it gets colder at the side reflector and more hot at the inside.

Later as we reached 100 % power, the same feature (lower sketch) showed up.

For each power level the rod positions were similar (they depend on the power level and refuelling), so the change of the temperature profile is a real fact.

3 Refilling and fuel circulation

A short view on the total fuel circulation – in principle Transparency 5 and the loading pipes positions on top of the core Transparency 6.

The first core was staticly filled with only two mixtures, one in the inner core IC and one in the outer core AC. We used 1:6 refuelling, reactivity requirements asked for the daily fresh number $\boxed{\text{Transparency 7}}$.

To understand the moving core – bulk material in a silo – we used the channel flow model with rather straight vertical flow lines $\boxed{\text{Transparency 8 + 9}}$, the velocity characterized by the amount of horizontal steps defined by the fixed time interval. $\boxed{\text{Transparency 9}}$ is obviously faster in the outer region than $\boxed{\text{Transparency 8}}$.

One can define refuelling parameters α_i using 6 assumptions:

1 volume conservation, numbers of discharges pebbles have to be replaced with refuelling ratio β = pebbles IC/pebbles AC

2 fuel ratio IC/AC

3 absorber ratio IC/AC

4 splitting age T* within the refilled fuel distribution $\boxed{\text{Transparency 10}}$

5 more used fuel is only recycled to the IC

6 absorbers are recycled to the IC (temperature reasons)

The resulting α_i are rather lengthy expressions. The difficulty is, that there is not always a solution with these six assumptions fixed <u>and</u> one has also to distribute into the core what is coming out of the core, without interim storage.

Well, we managed that. It was difficult, but by far not the most difficult task.

As I mentioned, the first core was built with only two fuel mixtures. So, being through first time with the innermost channel should be seen in the discharged fuel elements and the remaining fuel content.

We had this zero-power reactor to measure the discharged pebbles, where we could distinguish by the pebble signal what it is and how much fuel remained $\boxed{\text{Transparency 11}}$. Being through the first time, one could recalculate the "measured" pebble flow which turned out to be much higher than designed:

refuelling ratio	β =	0.83	WB1	β_{design} = 0.56
		0.78	WB2	
		0.76	WB3	
		0.76	WB4	Transparency 8 + 9 again.

We than changed the flow model for pre-calculation of reactivity development from β = 0.56 to β = 0.78.

Later on we could verify – in looking at the discharged pebble distributions – that also WB 5 – 8 (2nd time arrival of the innermost channel) and WB 9 – 11 moved with essentially β = 0.78. In fact this meant that fuel in the outer core is staying there for a very long time getting rather burned up, which in turn shows up as low integral temperatures at the core bottom. This was evidently different from the design calculations with β = 0.56, and not the inserted core rods could be taken as explanation, because than they were only slightly inserted.

Unfortunately the refilling was made – due to decision of the manufacturer of the plant - with β = 0.70 and fuel ratio higher to IC than at the beginning. Due to calculations this was the direction to the equilibrium core. People were convinced, the actual core would aline with the equilibrium core calculations. But the core did not. At least not than.

In fact this worsened the situation with the bottom temperatures because of the following fact: the friction coefficient of graphite in He increases with decreasing temperature Transparency 12. So, the rather cold temperature at the side reflector hindered the pebble flow which in turn would increase β further and so on. A self-amplifying process.

The design-β rested on many experiments done in models with different scales, 1:6 and 1:2, as far as I remember. But all experiments were done in air and without any temperature or even temperature profiles. So, something was missed.

It took us as long as WB 16 (May '88) to convince the people to change the fuel ratio, thus loading more fuel elements to the AC. It had the effect of increasing the outer core temperatures. We finally succeeded to have the effect wanted.

But the decision for decommissioning the plant came soon afterwards (WB 23, AVLT = 423). And than this same very good idea turned out to be a bit of a problem for de-loading the core! If nothing is refilled, the core behaves as an egg-clock. The inner core flows out and the upper side of the outer core plunges into the inner channel and makes a rather reactive assembly there.

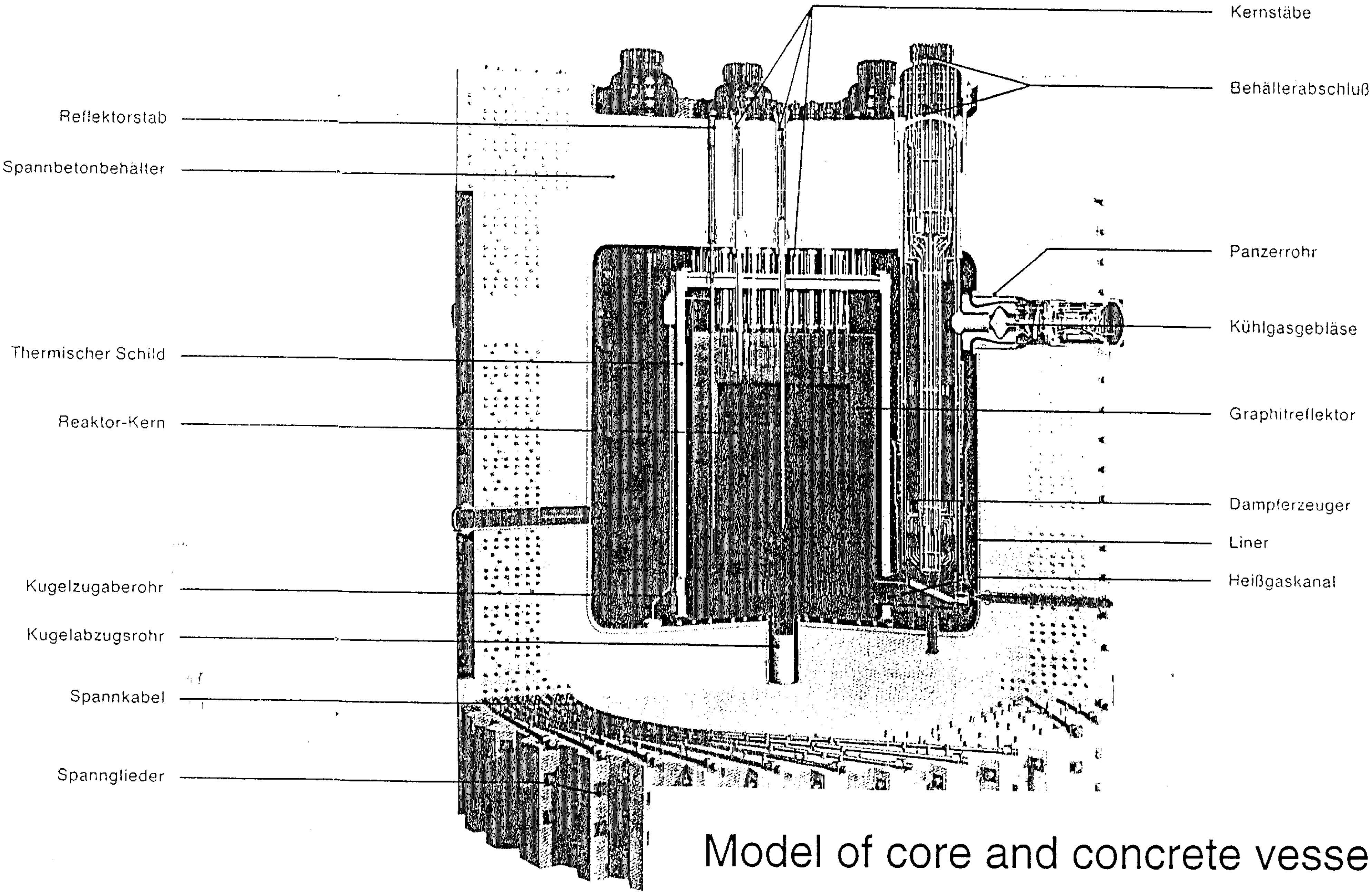
But I want to mention another experiment we made for the pebble flôw, we added a small amount of fresh absorber at a later time with only minor reactivity effects. With the zero power reaction we could definitely distinguish these fresh absorber being once through from others, which we mixed in the first core. The amount showing up at the bottom was a direct function of the integrated channel area on top of the core where we added the absorber. This was another clear prove for the flow channel model. Unfortunately, the test was not finished because of the decision for decommissioning.

4 Conclusion

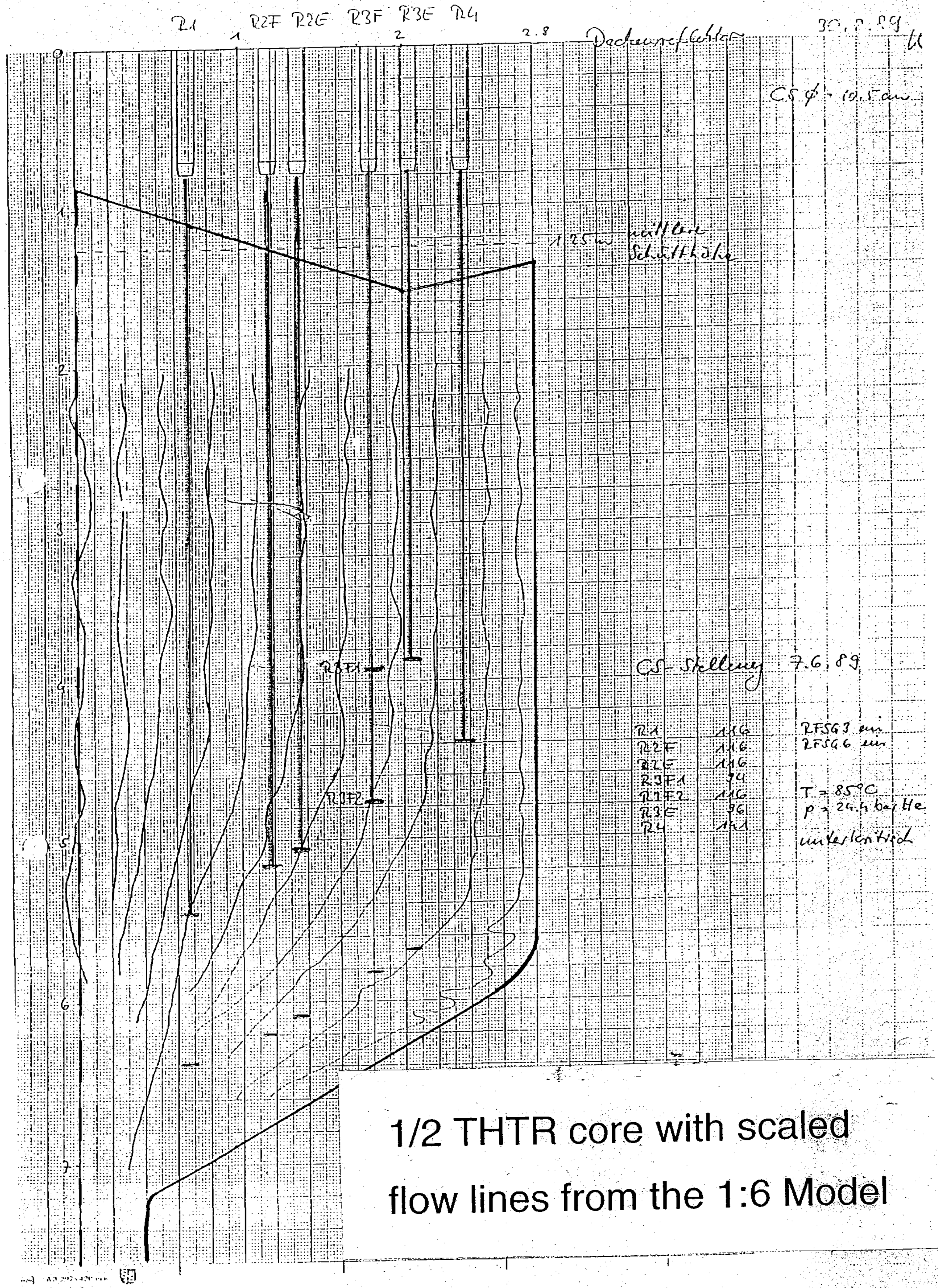
So, what is the point I want to make?

To have a moving assembly, like the core of a pebble bed reactor, one should have a thorough look at several physical properties from time to time, to confirm that you have the core you thought. It is not sufficient to believe in model calculations, even if they pretend to have included all the experiments that are available. There may be aspects missing, as we have seen. The verification of the core by means of measurements of the physical properties is absolute essential. It is more essential than in the common water reactor. (But these water reactors have many other difficulties, a gas cooled graphite reactor will never have).

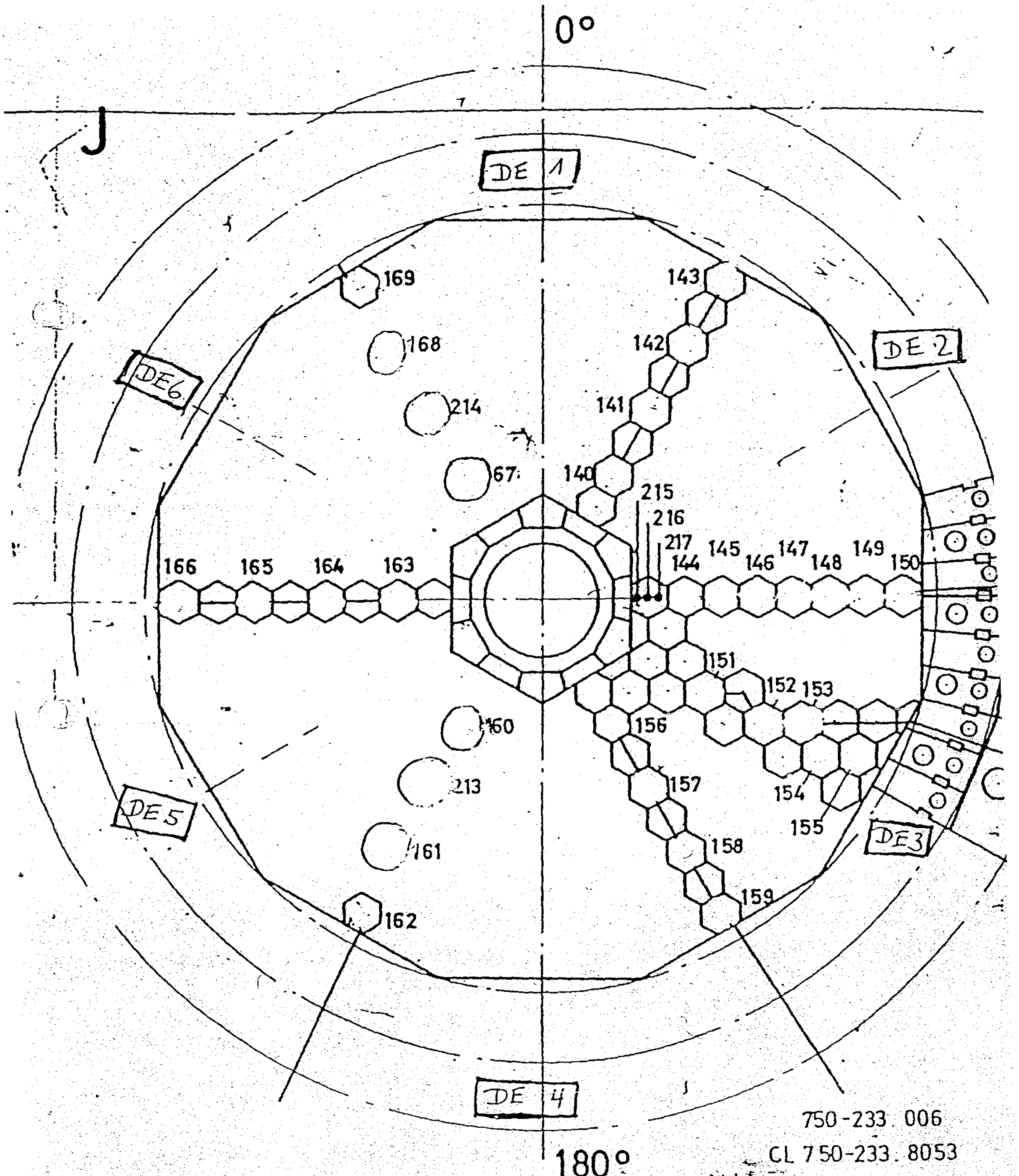
Physical properties to be looked at are, beside the temperatures in the core bottom, the rod values (total and differential), reactivity (in rod equivalents), rod insertion time in case of core rods and, most important, discharged pebbles distributions including test charges to measure the pebble flow.



Model of core and concrete vessel



1/2 THTR core with scaled flow lines from the 1:6 Model

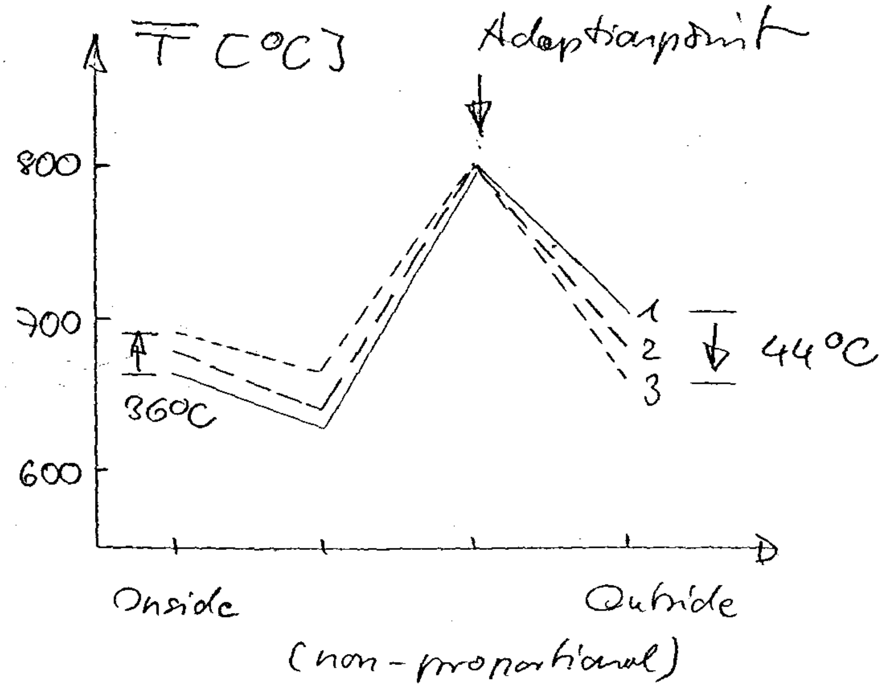


Position of NATE in corebottom reflector

P ~ 40 %

AULT/BULT

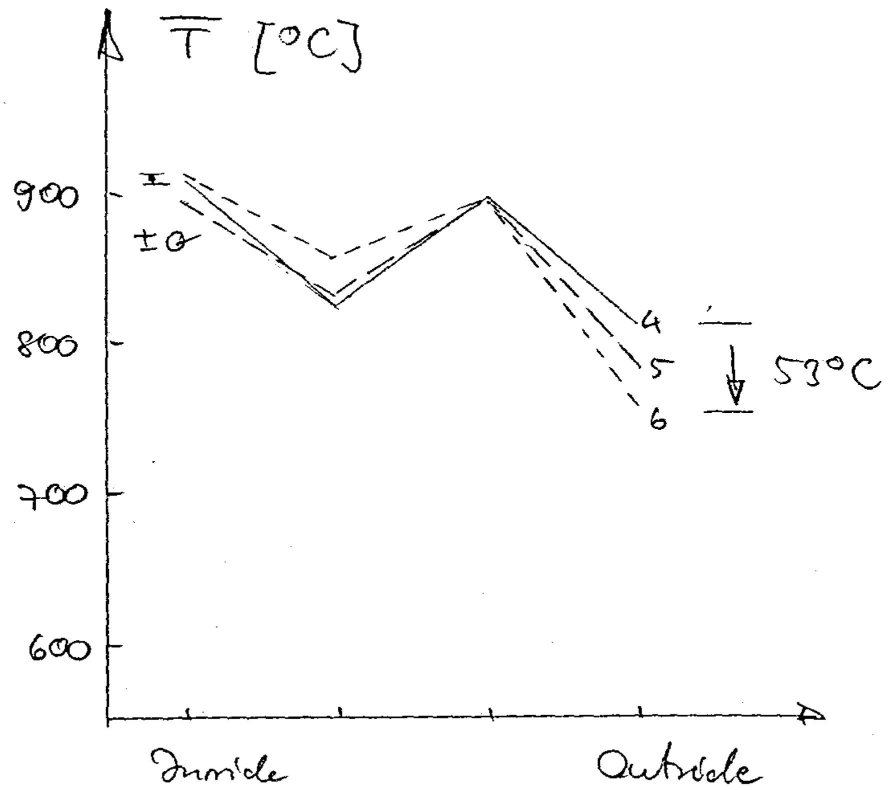
1	17/12 (~WB1)
2	24/25 (~WB2)
3	46/39



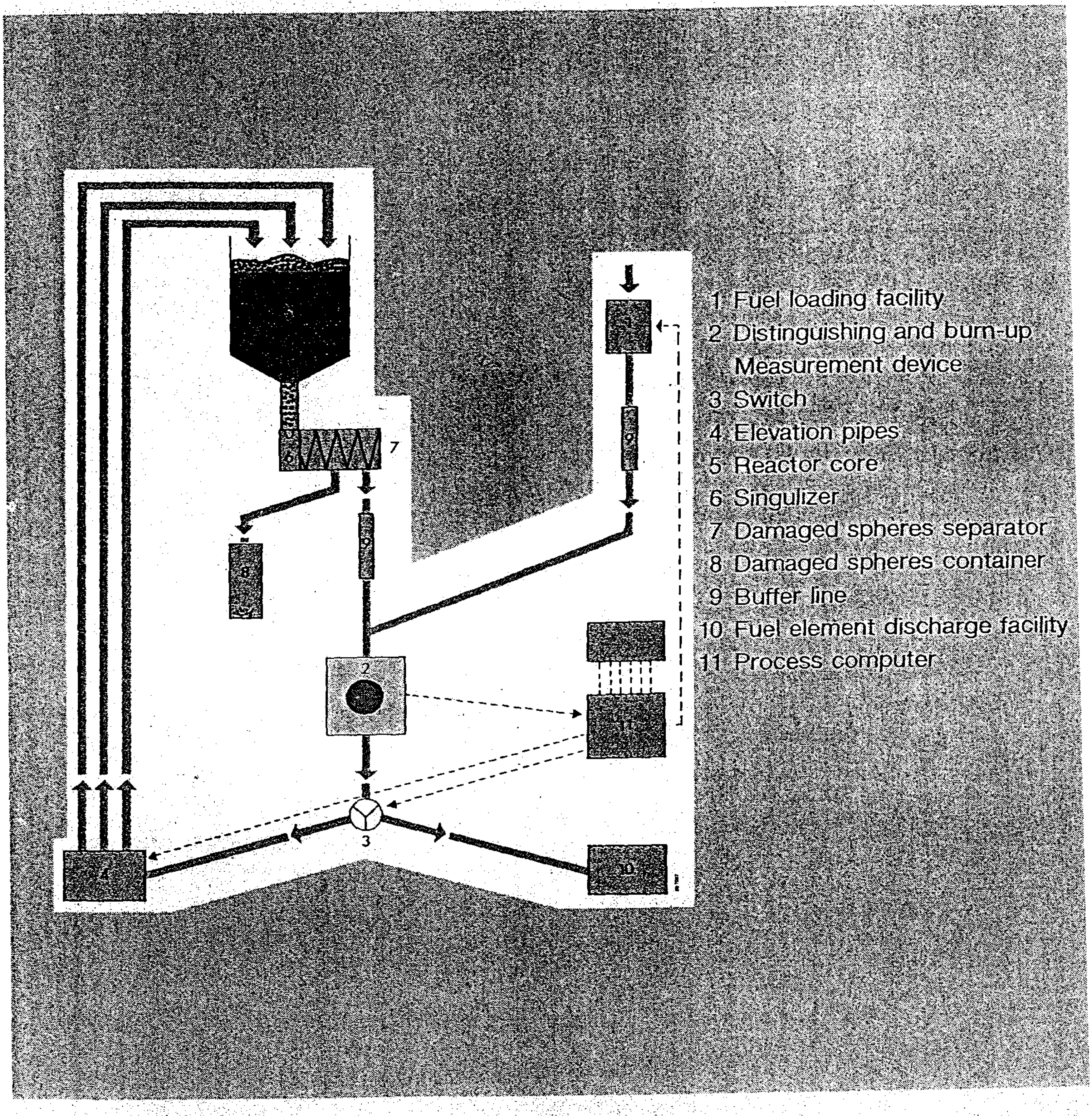
P ~ 100 %

AULT/BULT

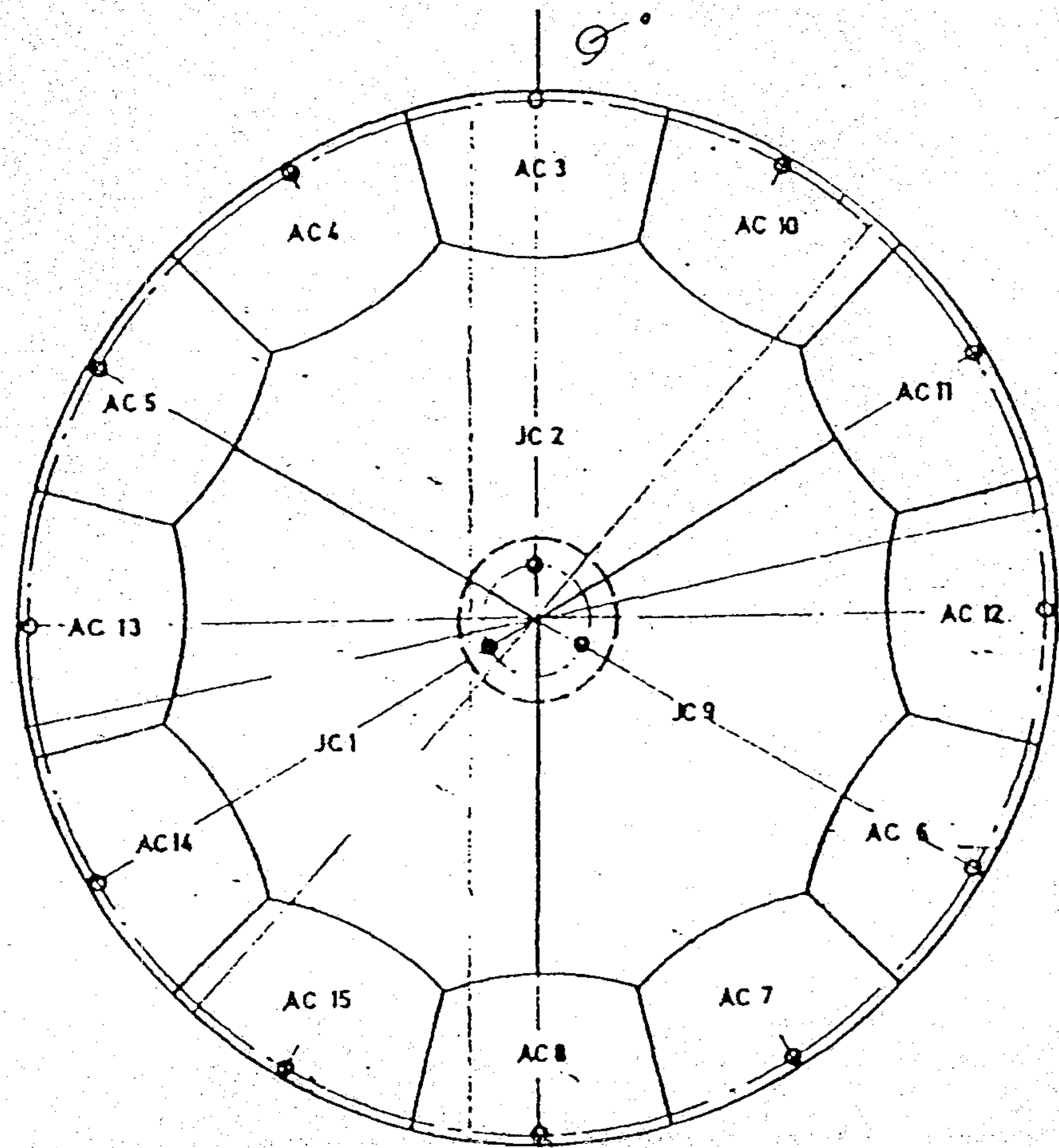
4	90/57
5	125/89 (WBC)
6	161/120 (WBF)



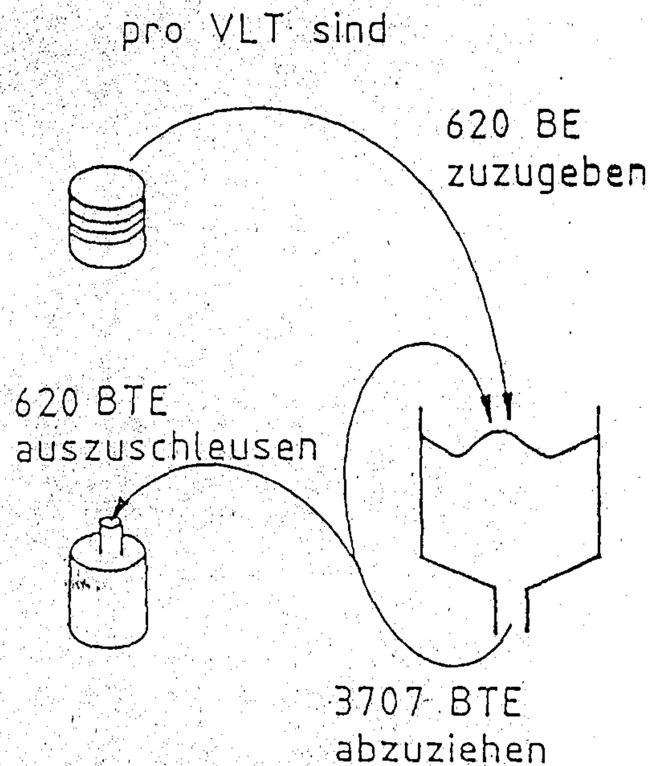
Temperature profile in bottom reflector
at 40 % and 100 % power and increasing burn up



Fuel circulation facilities (very simplified)



Position of fuel loading pipes on top of the THTR core



Seit Beginn abbrandabhängiger Beschickung am 26.10.85 wurden insgesamt bisher

zugegeben	248 055	Brennelemente
	5 200	Absorberelemente
abgezogen	1 327 493	Betriebselemente
ausgeschleust	250 690	Betriebselemente

Refuelling THTR

BF 3517
Blatt 24

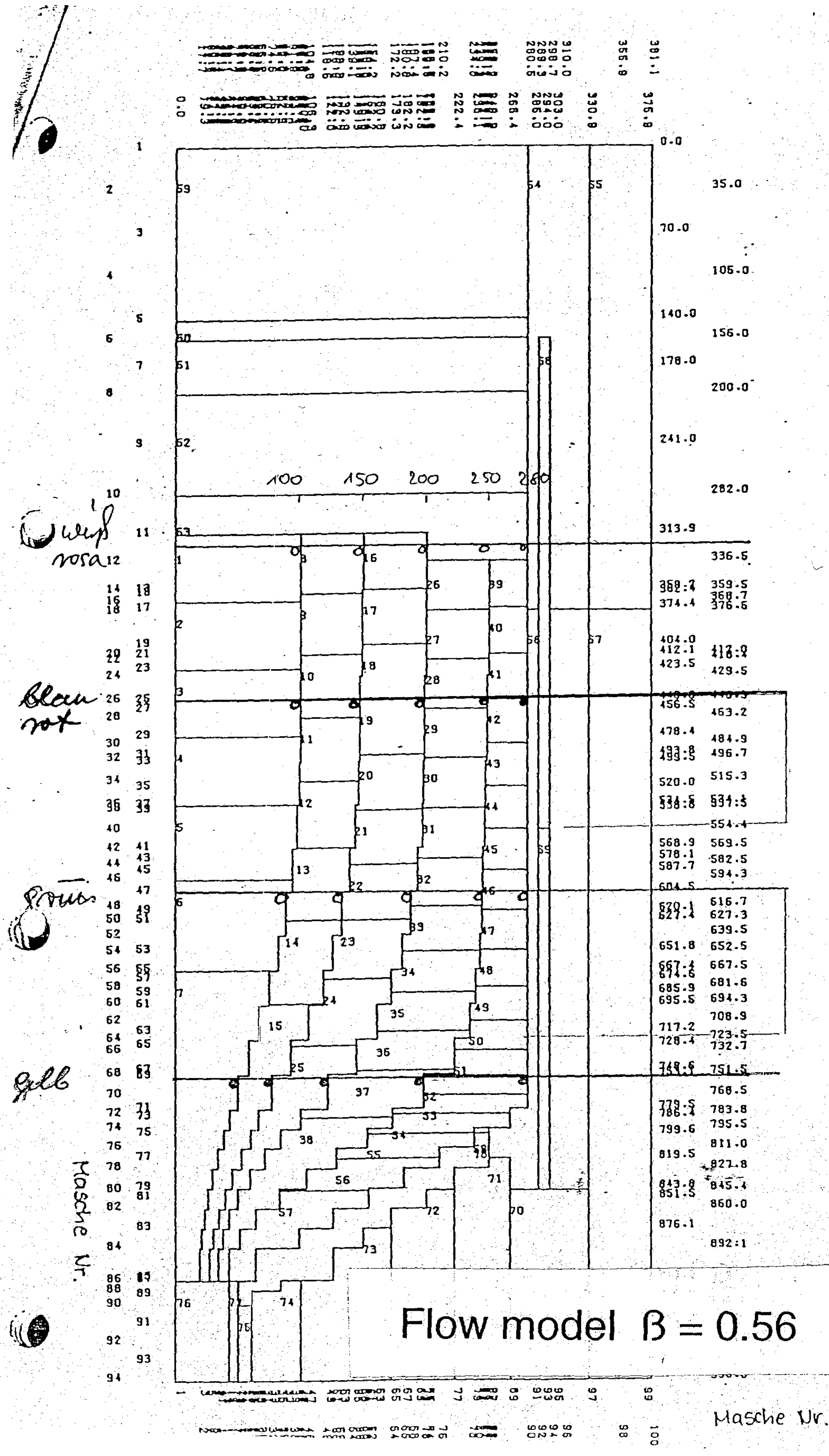


Abbildung 2 Abbrandzonen "Fließmodell β = 0.56" Schächten 7 8 10 13 19, 57

BF 3517
Blatt 23

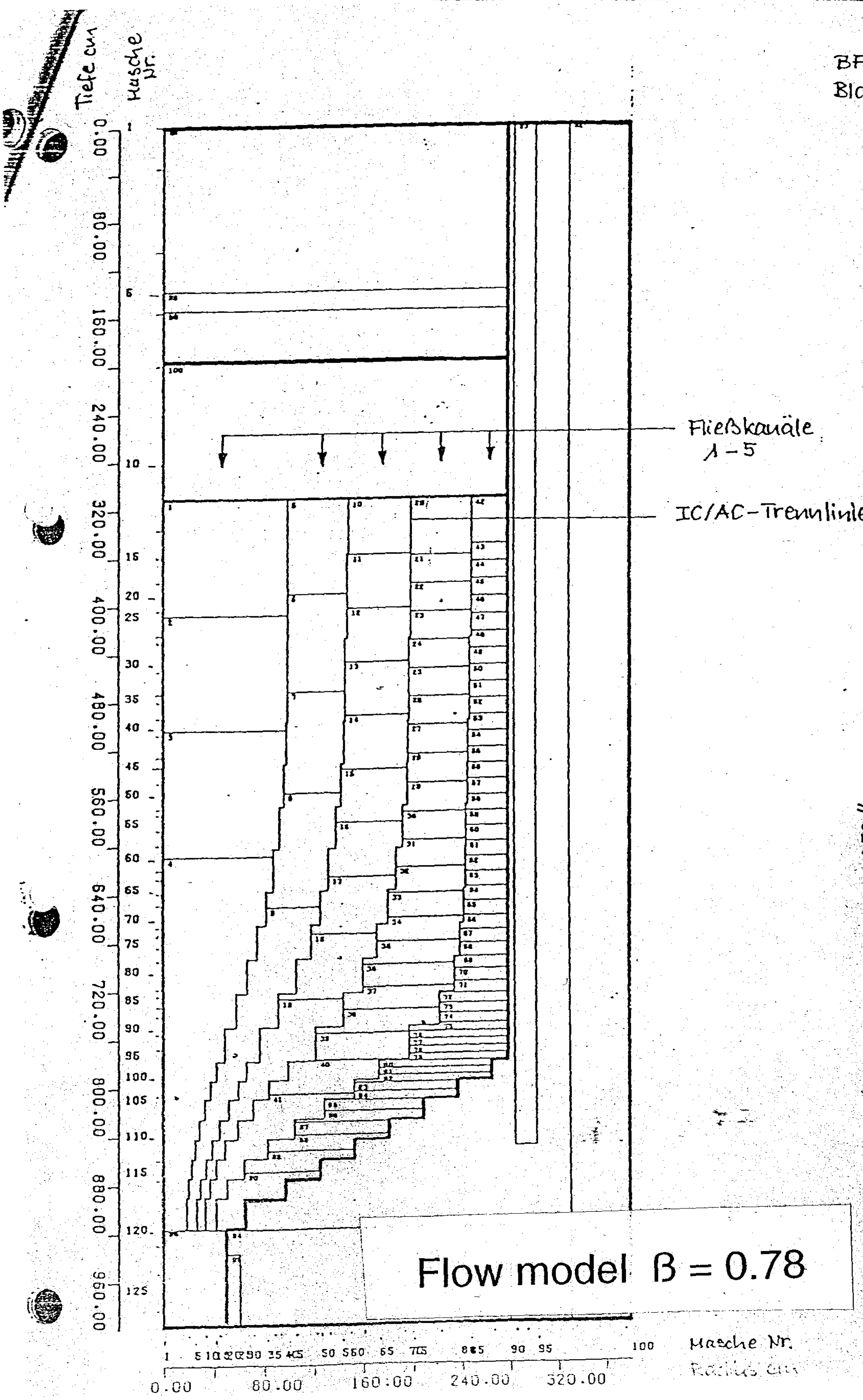
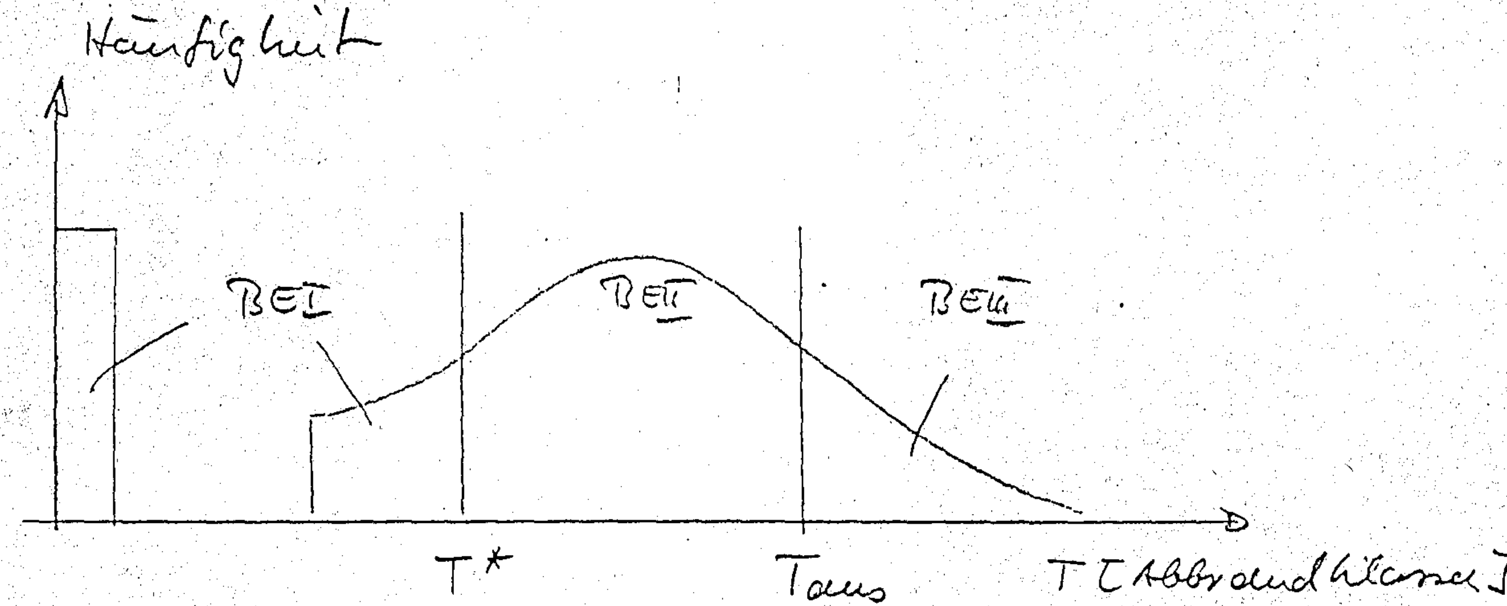


Abb. 1: Abbrandzonen "Fließmodell β = 0.78" (Benutzt in Standard-Nachrechnung THTR 03)

Mit fortschreitendem Abbrand ergibt sich die Notwendigkeit, zwischen frischeren und älteren BE zu unterscheiden, das Trennalter T^* wird eingeführt mit BEI jünger als T^* und BEII älter als T^*.

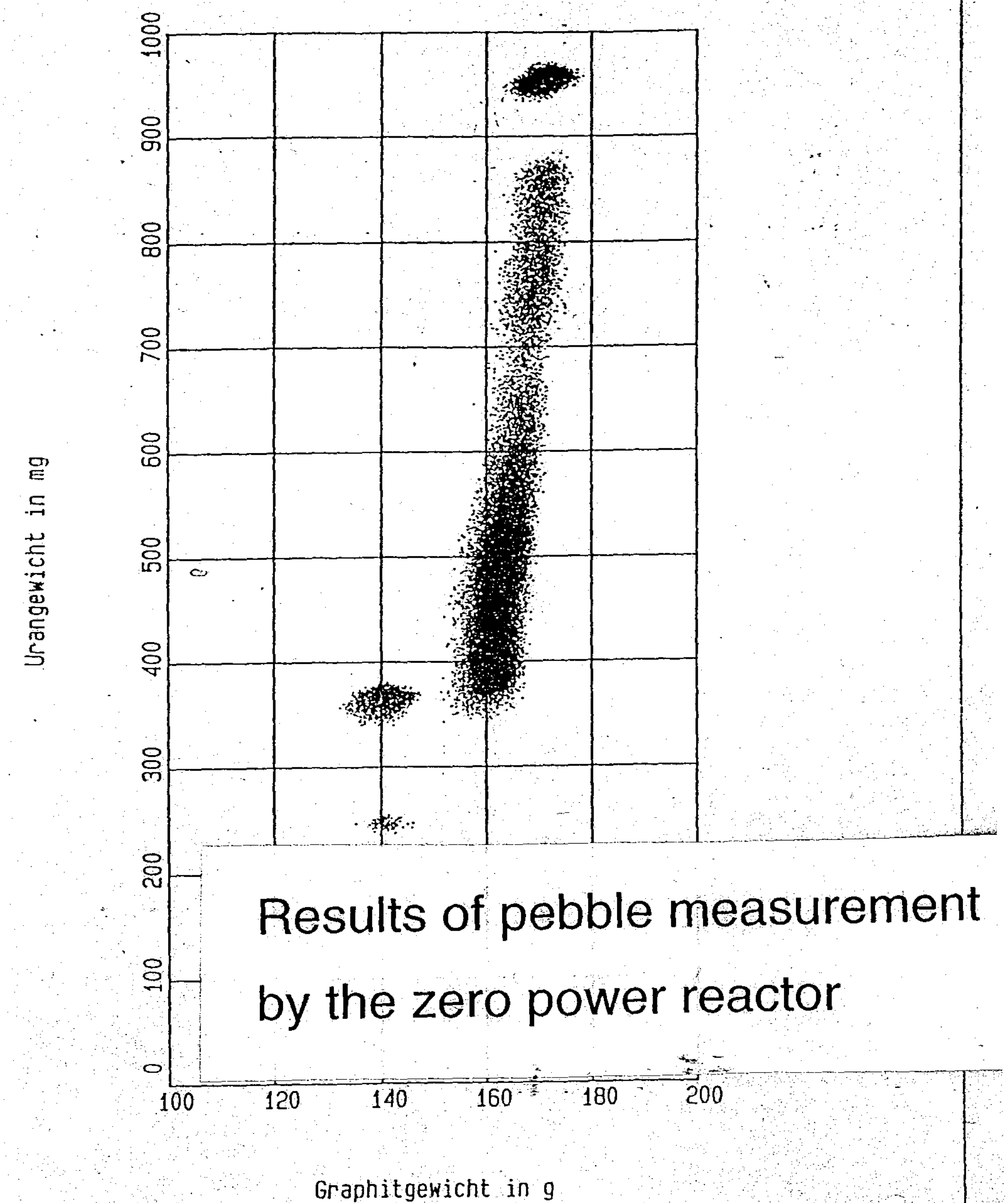


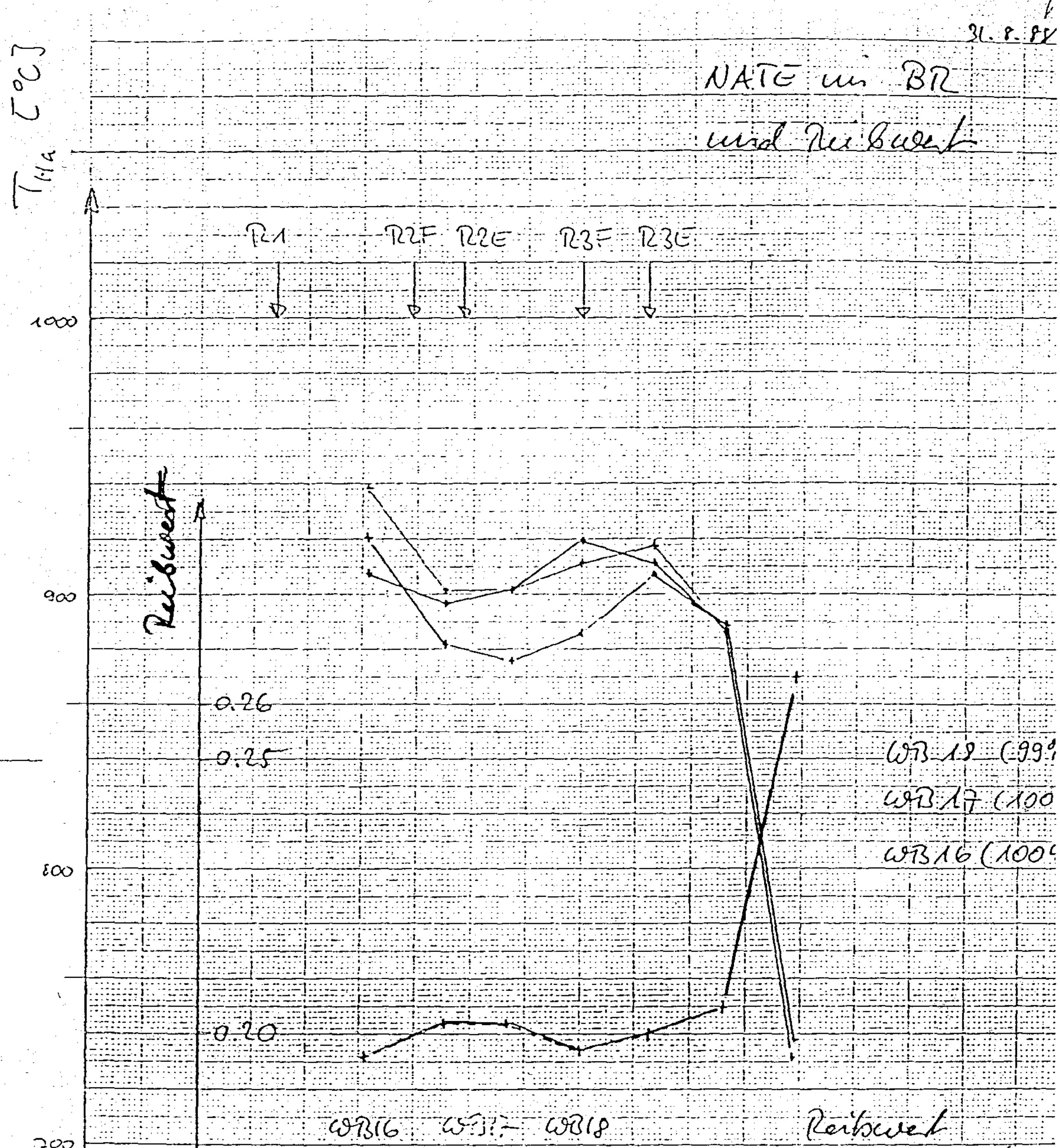
Age distribution for burned fuel

Uran-und Graphitgewichte der abgezogenen bzw. zugegebenen BE und GE

WOLKE 88/36 30.08.88 13.50 UHR NACH UB 240/1

ausgewertet wurde jedes 1te BTE; Anzahl BE : 31264, Anzahl GE : 2636





	WB16	WB17	WB18
R1	70	70.7	71.3
R2F	58.3	60.7	65.5
R2E	57.0	62.0	63.0
RESG	1633	1578	1594
	20.5	1.6	22.6
NATE	800	800	800

Reibwert
He ausgegast

T °C	Reibwert
300	0.45
600	0.33
800	0.23
1000	0.18

Temperature profile and friction coefficient

600

100 200 250 r [cm]

REVIEW OF SOME ASPECTS OF RADIOLOGICAL INTEREST DURING THE ESTABLISHMENT OF THE SAFE ENCLOSURE OF THE THTR 300 PLANT

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XA9848083

Abstract

One of the first activities with the establishment of the safe enclosure was the disassembly of the reactor of the burn-up measurement facility. This was a graphite-moderated, air-cooled reactor with strip-shaped fuel elements made of an aluminium uranium alloy. The reactor contained 3.9 kg of high-enriched uranium (93% U-235), the thermal power output was 500 W. Because of the highly cramped conditions, the acceptable dose level and the limited number of fuel stripes, the decommissioning was executed almost exclusively manually. To reduce the collective dose of the personnel, an extensive training with a 1:1 scale mock-up was carried out prior to decommissioning. The removed fuel elements were put into special baskets and were shipped to the interim storage facility BZA in two CASTOR THTR/AVR casks.

In order to clear place for the installation of components of the new ventilation system and other systems, the components for high-purity helium compression and storage had to be dismantled. More than 90% of the metal were unconditionally released as iron scrap.

Extensive measurements had to be carried out on the dismounting and inspection equipment which had been mostly already in use during the 3 year time of operation. As a result 3 Mg had to be stored in the remaining controlled area, app. 183 Mg were stored within the supervised area and app. 49 Mg were released as free of contamination.

Due to the high tritium inventory, two containers with barrels filled with waste could not be shipped to external storage sites and therefore had to be stored in the remaining controlled area within the envelope of the safe enclosure.

Another interesting aspect of the low contamination level of the THTR 300 plant was the release of buildings from the restrictions of the Atomic Energy Act and reduction of the controlled area to a supervised area. Based on statistical methods we were able to prove the low-level contamination status with an acceptable amount of measurements.

Finally a new system for monitoring of released radioactivity with the new exhaust air system was designed and built. Government authorities requested a system with advanced sensibility for low emissions of tritium and carbon-14. The design especially had to consider the highest mean time between failures and the lowest mean time to repair possible.

1. Introduction

This paper is to give a brief review of some aspects of radiological interest during the establishment of the safe enclosure of the THTR 300 plant operated by the Hochtemperatur Kernkraftwerk GmbH (HKG). During the establishment of the safe enclosure, the consortium KSE (Noell-KRC and STEAG Kernenergie) as a general contractor was also responsible for radiation protection organization and waste management and provided one of the radiological health and safety officers and all of the health physics personnel.

2. Dismantling of the burnup measuring reactor

Loading and unloading of the THTR reactor core was carried out while the burnup of the pebble-shaped fuel elements was monitored by means of a burnup measuring system. The main component of this system was the burnup measuring reactor (Solid Moderated Reactor) in which a reactivity effect is caused by operating elements as they pass through the reactor. Evaluation of this effect permits determination of the type of element and, in case of fuel elements, the burnup of the element.

Unloading of the THTR reactor core was completed by 28 October 1994 with the establishment of the state "reactor core free of nuclear fuel". By then, the task of the burnup measuring reactor was completed so that dismantling of the burnup measuring reactor could be initiated in order to remove the nuclear fuel contained therein.

2.1. Initial situation

The burnup measuring reactor was a graphite moderated thermal reactor with a rated output of 500 W, arranged in the reactor hall below the prestressed concrete reactor vessel. The reactor core (1.0 m · 1.2 m · 1.1 m) consisted of various graphite plates provided with grooves for accommodation of the 767 strip-shaped fuel elements. The fuel elements have a rectangular cross-section (15 mm · 1.1 mm) and a length of between 89 and 711 mm. They contain 93% enriched uranium in a U-Al alloy (20% uranium, 80% aluminum). Total uranium content of the core was 3.9 kg.

The core was enclosed by a graphite reflector consisting of plates similar to those of the core. Outside dimensions of the SMR were thus 1.8 m · 2.0 m · 2.0 m. The operating element guide tube, used to guide the operating elements rolling through the core by gravitation, passed through the center of the reactor core. The entire reactor composed of graphite plates was mounted on a steel slab anchored to the floor and was supported by a steel structure installed around the reactor. Reactor instrumentation, absorber rods and the neutron source were arranged in twelve vertical drill holes through the reactor core. Figure 1 gives a general idea of the burnup measuring reactor.

The initial radiological situation was determined by a burnup of approx. 3.1 MW·h/kg U after unloading of the THTR reactor core. Overall activity, originating mainly from the fission products, totaled $2 \cdot 10^{12}$ Bq/kg U. The measured dose rate in the room of installation of the SMR ranged from 500 to 800 µSv/h as regards gamma radiation and was below 1 µSv/h as regards neutron radiation. Values ranging from 1.5 to 20 Bq/cm² were determined for non-fixed contamination. When dismantling work was initiated, indoor air activity concentration was below 40 Bq/m³.

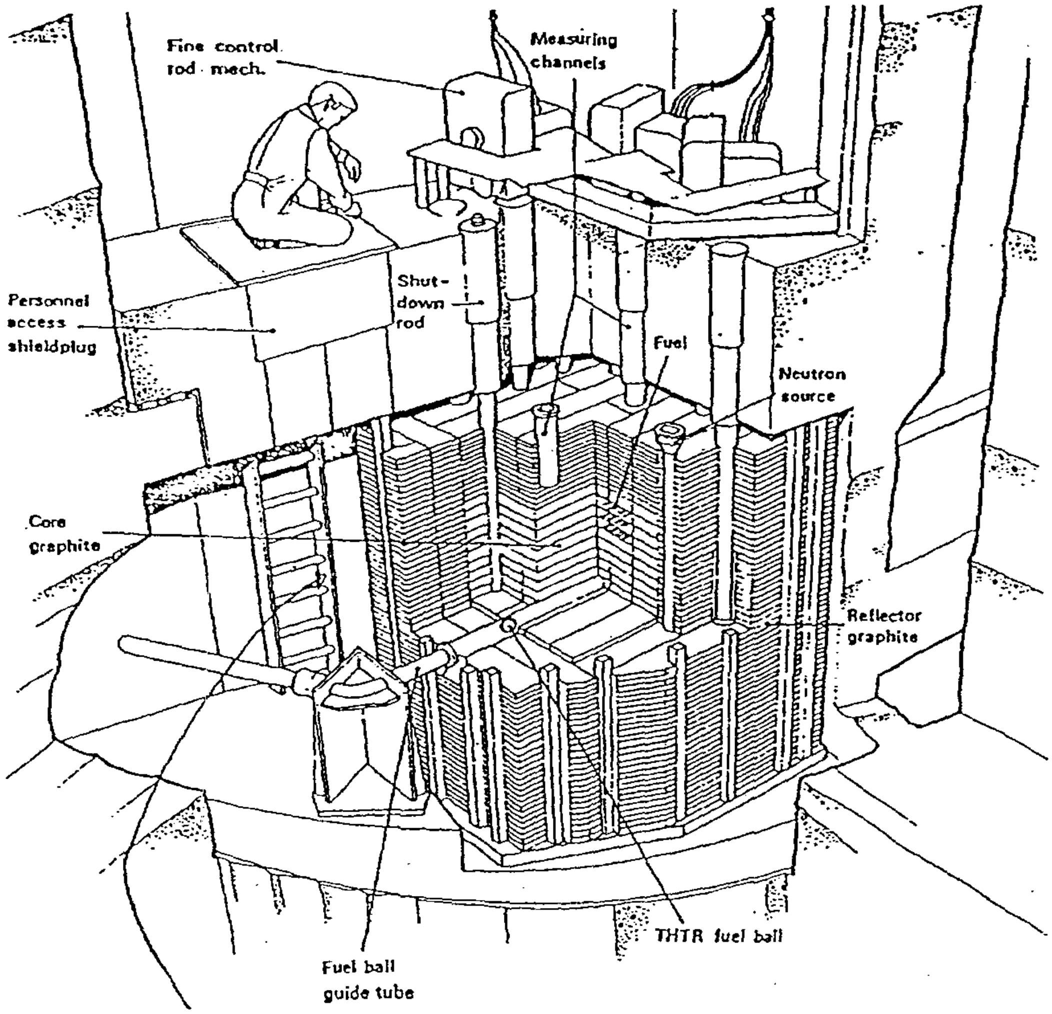

FIG. 1. Installation of the burnup measuring reactor

In terms of technology to be used for dismantling, the initial situation was characterized by extremely cramped spatial conditions in the SMR room of installation and difficult access to this room.

2.2. Preparatory activities

Under the given radiological and spatial conditions, a dismantling concept was chosen which was based on manual dismantling using suitable auxiliary equipment and installations. Preparatory activities included mainly the installation of a 1:1 mock-up, staff training and testing of the auxiliary installations.

The mock-up consists of an SMR room of installation and the access area arranged above. Height of the mock-up totals approx. 7 m.

The mock-up was used to test all equipment and installations designed specifically for dismantling of the reactor (particularly the mobile shield) both individually and in interaction and the devices were optimized.

Testing of the devices was followed by training of the dismantling personnel. Training concentrated on

- assembly of the auxiliary installations under the cramped conditions;
- video-monitored handling of the fuel elements and graphite plates by means of auxiliary equipment;
- handling of the fuel element shielding container (up to storage of the fuel elements in the THTR 300 fuel element storage facility);
- conduct in case of incidents and management of abnormal situations.

The work papers and step sequence plans used for testing and staff training were revised on the basis of the experience made with the mock-up and provided thus the basis for a comprehensive set of work papers and step sequence plans for dismantling of the SMR.

2.3. Dismantling of the burnup measuring reactor

The burnup measuring reactor had to be dismantled only to the extent necessary for removal of the nuclear fuel.

In addition to the equipment and installations that had already been installed at the mock-up during testing, the following preparatory activities had to be developed in the THTR 300 nuclear power plant:

- preparation of the transport path leading up to the SMR room of installation, including assembly of the transport means (inclined haulage and hoist);
- provision of a charging aid for the shielding device, reloading into transport and storage cask transport cages;
- installation of an auxiliary ventilation system for the SMR room of installation.

The activities in the THTR 300 plant - from preparation via removal of fuel elements, loading into shielding device and until transfer to the THTR fuel element storage facility - were carried out by a staff of approx. ten persons in one shift over 30 work days. The collective dose for the personnel was only approx. 10% of the maximum value of 200 mSv stated in the application that had been filed for the licensing procedure under nuclear law, and approx. 20% of the dose expected according to the initial step sequence plans.

The SMR fuel elements were loaded into two transport and storage casks CASTOR THTR/AVR. Shipping of the two CASTOR casks to the Ahaus fuel element interim storage facility on 10 March 1995 completed the activities for management of the SMR fuel elements.

3. Some main sources of waste from decommissioning

The wastes arising from the relevant work for the implementation of the safe enclosure and their destination - except waste containing nuclear fuel – are compiled in figure 2. The following are some of the main sources which are discussed in detail.

License	Year	Works	Shipped: Unrestr. Release [Mg]	Shipped: Radioact. waste [Mg]	Stored in THTR [Mg]
7/12a					
	1994				
		Defueling of the core	14 1)	16	56 2)
Am.No.1	1995	Final inspection Disassembly of the measurement reactor	16	5 3)	17
Am.No.2		Release of the steam-feedwater-circuit	88 4)	-	-
		Removal of mufflers	168	-	-
Am.No.3, 4		Sealing components, Helium compressor	62	1	60
	1996	Dismount. equipment	64	2	186
7/12b		New vent systems	88 5)	-	1
Am.No.1		Evaporator plant, Change status rooms	-	33 6)	-
		Reconnect vent syst.	21	3	10
		Remove vent stack	79	-	-
	1997				
		Total: 7)	730 8)	83	350

1) Pulverized resins from the condensate cleaning system
2) Graphite and absorber elements within the spent fuel element storage
3) Solid organic waste
4) Removed parts are included only.
5) Rubble
6) Evaporator concentrate and mud
7) Including works not listed above.
8) + 7,400 Mg structural steel and components
+ 44,400 Mg reinforced concrete

FIG. 2. Sources of waste during decommissioning

3.1. No-contamination-measurements for components of the secondary system

With a second amendment to the core unloading license (7/12a), no-contamination-measurements of components in the turbine hall and in the adjacent feedwater tank building and the disassembly of the steam-feedwater-cycle mufflers on the roof of the reactor hall were permitted. Only the waste water discharge station in the supervised area of the turbine hall continued to be subject to measurement after having achieved the state of safe enclosure.

The no-contamination-measurements of components of the steam-feedwater-circuit were performed at this early stage to enable reuse of these components at any other site.

Theoretical investigations based on measurements with pulverized resins from the condensate-cleaning system and certain experience from AVR led to the conclusion that the complete steam water cycle would stay clearly below the threshold value for unconditional release of iron scrap. The number of measurements to be made was comparatively small. Prior to granting of the second amendment to license 7/12a, HKG took 10 measurements at components that were easy to exchange (e.g. valves) and components easy to access (e.g. low pressure section of turbine above the condenser) to verify the theoretical model. After granting the second amendment an additional 20 measurements were carried out together with the experts. These gamma-spectrometric measurements were partly made in the laboratory and partly in situ. For the in-situ-measurements, the background level was subtracted and the calibration factor was calculated for the actual geometry. The detection threshold referring to Co 60 was 0.006 Bq/cm^2 (equivalent to 0.001 Bq/g at a minimum thickness of 8mm). The nuclear supervising authority approved the release of these components on October 20, 1995.

Six non-contamination-measurements of the mufflers were performed under the supervision of the Technical Inspection Service at two representative mufflers. Approval for disassembly of all 6 units and their unconditional release for scrapping was given on July 19, 1995. The total masses of iron scrap arising from these activities are shown in figure 2.

3.2. Disassembly of the high-purity helium compressors

The 4th amendment to license 7/12a was issued on October 27, 1995. It permitted the disassembly of components of the high-purity helium compression and storage system, aiming to clear space for the installation of components of the new ventilation and activity monitoring system.

The components to be removed were installed in the supervised area of the plant. Thus they actually had to be non-contaminated. It was known, however, that certain inner surfaces of pipes had been slightly contaminated due to back streaming gas during plant operation.

Contaminated and non-contaminated piping segments had to be determined. During these measurements, it was found that the two heavy four-stage helium compressors were slightly contaminated in their first stages. They were disassembled prior to being subjected to a thorough investigation. As far as necessary they were decontaminated.

Parts of the system for which no-contamination-measurements were easy, were brought to closed containers installed outside and stored there until approval by the authorities had been obtained. Parts for which the state of no-contamination was to difficult to prove, were packed into 200 l-barrels and stored in the supervised area for the time of safe enclosure.

The final no-contamination measurements for the helium compressors were made in March 1996. More than 90% of the material (approx. 62 Mg) was unconditionally released as iron scrap.

3.3. Measuring of the dismounting and inspection equipment

The THTR nuclear power plant was equipped with a partly shielded dismounting and inspection equipment. This equipment was used for work on the fuel circulating system, the helium purification system, the absorber rods and for the inner inspection of the prestressed concrete reactor vessel. In parts, this equipment had already been in use and was therefore contaminated.

Due to the fact that some of the equipment had already been disassembled, the number of contamination-measurements amounted to several hundred. As a result of the measurements, the single parts were classified into three groups: parts with a contamination higher than

5 Bq/cm^2 were stored within the remaining controlled area; parts with contamination between 5 and 0.5 Bq/cm^2 were stored in the supervised area of the remaining plant; and parts with a contamination level below 0.5 Bq/cm^2 were unconditionally released for scrapping.

As a result, 3 Mg had to be stored in the controlled area, approx. 183 Mg were stored within the supervised area, and approx. 49 Mg were released as free of contamination.

3.4. Waste from external conditioning

In September 1996, two containers with barrels filled with tritium-contaminated waste were stored within the cover of the safe enclosure. The 16 D350-barrels of high-grade steel were filled and sealed at the Karlsruhe research center. Due to the high tritium inventory of the barrels, storage at the final repository Morsleben is not possible today. The tritium-activity amounts to 2.9E+12 Bq per Barrel.

4. Downgrading from controlled area to supervised area

With the first amendment to license 7/12b (establishment of the safe enclosure) issued on July 15, 1996 HKG was allowed to change the status of rooms outside the cover of the safe enclosure from controlled to supervised area. Therefore it had to be proved that the surface contamination of the buildings and components did not exceed 5 Bq/cm^2 and that the dose rate did not exceed 7.5 µSv/h in this area. In addition, HKG demanded that in rooms which should be accessible without any restrictions to persons who are not occupationally exposed to radiation, the dose rate should not exceed 2 µSv/h.

For a total area of about 12000 m^2 (170 rooms) the fulfillment of the above conditions had to be proved. The proof was provided in two steps: through the analysis of the history of operation of the plant and through measuring at representative locations. For regions on the floor with high probability of contamination, the number of measurements was 1 measurement per 2 m^2; for regions with low probability of contamination and for walls up to a height of 2 m the number of measurements was 1 per 10 m^2. For components, the number of measurements was 1 per m^2. Dose rate measurements were done in every room. Spots with higher dose rates were either decontaminated or shielded. Components that continued to fail meeting the specifications of the supervised area even after decontamination were dismantled and stored in the remaining controlled area.

Due to the low-level contamination of the former controlled area the change to a supervised area was achieved with a relatively small amount of measurements. The total number of contamination measurements was 2316. Only 87 measurements showed values higher than 0.5 Bq/cm^2. All measurements were taken in the presence of members of the Technical Inspection Service.

5. Release of buildings from the scope of nuclear legislation (AtG)

The first amendment to license 7/12b allowed initiation of measurements for the release of buildings from the area of application of nuclear legislation. All buildings of the site outside of the safe enclosed plant were to be released from the restrictions of the Nuclear Energy Act, that is they were no longer subject to nuclear legislation.

From the analysis of the history of plant operation the buildings were divided into three classes:

- AE: slightly contaminated;
- BE: probably not contaminated;
- CE: clearly not contaminated.

All buildings outside the supervised area belonged to class CE. The turbine hall and the feedwater tank building belonged to class BE just as the health physics laboratory and some rooms of the access and safety building. Only the waste water disposal duct and parts of the waste water discharge station in the turbine hall were known to be contaminated and therefore allocated to class AE.

After thoroughly cleaning the waste water disposal duct of contaminated mud and dismantling of the contaminated parts of the waste water piping, it was possible to provide proof of no-contamination. For that proof, it had to be shown that the surface contamination was below the limits of the German radiation protection ordinance: 0.50 Bq/cm^2 for most of the beta/gamma-nuclides and 0.05 Bq/cm^2 for alpha-nuclides. From experience it was known that only Co 60, Cs 134, Cs 137 and Sr 90 (and in special cases H 3) were relevant in most cases.

Material samples to prove falling below the mass-specific clearence level of 0.1 Bq/g were taken from the waste water duct and from several sumps in the turbine hall and the feedwater tank building. In other cases, proof was provided by means of gamma-spectrometric in-situ measurements.

The number of contamination measurements was 1 per 25 m^2, however, at least 3 per room, in order to permit evaluation of representativeness. Locations with an increased probability of contamination were chosen as measuring points, such as floor drains, sumps and transport paths. At the outset, the number of measurements to be taken at certain components was increased to 1 measurement per 10 m^2.

All gamma-spectrometric measurements showed values of contamination by Cs 137 exceeding significantly the clearence level. By means of the relations of activities of Cs 137/Cs 134, it was possible, however, to prove that this contamination was due to the Chernobyl incident and not due to the THTR 300 operation.

A total of 729 contamination measurements had been carried out. About 30 samples were tested with the gamma-ray spectrometer and 10 in-situ measurements were taken with a portable gamma-ray spectrometer.

6. Summary

In general, exposure to radiation during all activities for the establishment of the safe enclosure was significantly lower than expected. Due to the low level of contamination in the controlled and supervised areas, it was possible to furnish the radiological proof required for downgrading of the controlled area to the supervised area and for release of buildings from the supervision under nuclear law by means of a relatively small number of measurements.

THE THTR-300 COOLANT GAS ACTIVITY, AN INDICATOR OF FUEL PERFORMANCE

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Abstract

During incommissioning and the following 423 full power days (fpd) of THTR operation, the activities of 9 noble fission gas nuclides in the primary coolant have been measured quasi continuously. The radioactive decay times of these nuclides cover a range of more than 3 orders of magnitude. The sources and the mechanisms of the fission gas release can be derived from the dependence of the steady-state release fractions (R/B-values) on the half life times t1/2 of the various nuclides. Thus, the assessment of the fuel performance under operational conditions is primarily based on the routine measurements of the coolant gas activity.

During the first 100 fpd of the THTR-operation the slope s of the curve, R/B-values of noble gas nuclides versus t 1/2 in double logarithmic scales, was measured to be s = o.3. This result is in accordance with the design model, where the dominant release source is given by the manufacture induced uranium contamination of the graphitic matrix material. The relatively low slope-value of o.3 is caused by the superposition of the release fractions from two material components with different diffusion constants.

Within the subsequent operation the coolant gas activity increased by a factor of about two. However, the maximum did not exceed 4 % of the design value of the THTR coolant gas activity. The slope s decreased to 0.2. This change can be explained by the contribution of fission product atoms directly recoiled from the surface of damaged fuel elements into the coolant. In fact, a small fraction of the fuel elements have been mechanically damaged by the frequent insertions of control rods into the pebble bed core under the specific conditions of the THTR incommissioning procedure.

From the measured coolant gas activity data a fraction of coated particle failures less than 8 10-5 due to external forces was evaluated. In-service failures of particles embedded in the matrix can be neglected. This evaluation demonstrates its capability to identify specific fuel failure modes during operation. In this way, the coolant gas activity can be used as an indicator of the fuel performance.

1. Introduction

The radiation protection scheme of any gas cooled reactor requires the regular and reliable measurement of the coolant gas activity during all operational periods. The noble gas activities, predominantly detected in the coolant, determine an important source term for the radiological assessment of the plant during normal and upset conditions. In the case of the THTR-300, the steady-state noble gas activities in the coolant together with the iodine concentrations also constitute the leading sources for the maximum radiological design accident, the total loss of coolant. For the safety analysis of this accident the iodine release from the fuel elements is evaluated by the known relation between iodine and noble gas release.

Apart from the above safety aspects, the measurement of the coolant gas activity provides also a means to asses the actual quality of the fuel elements in the core. In particular, incipient fuel failures can be identified, so that adequate countermeasures can be developed in due time. In this context it is noteworthy, that the commercial guarantee of the in-service quality of the THTR fuel elements (burnup guarantee), which was contracted between the fuel manufacturer and the utility, was based on a certain value of the coolant gas activity (well below the design limit). Thus, the coolant gas activity can be used as an indicator not only of the fuel performance but also of potential financial obligations.

At any rate, the reliable interpretation of the coolant gas activity in relation to the fuel performance necessitates a sound knowledge of the fuel element material properties and the fission gas release mechanisms. This will be given in the following sections in advance of the actual THTR measurements and their evaluation.

2. THTR Fuel Elements

The THTR fuel elements are designed for the high enriched thorium / uranium cycle. The inner fuel zone of the spherical fuel element contains about 38000 $(Th,U)O_2$ kernels coated with pyrocarbon layers (BISO particles). The outer pressure holding layer of the coated particle is derived from methane at deposition temperatures up to 2100°C (HTI-layer). This extreme temperature causes a relatively high uranium contamination in the outer HTI layer. During the final heat treatment of the integral fuel element at 1950°C part of this uranium migrates into the surrounding matrix causing a finely dispersed uranium contamination in the matrix material. This contamination is the prime source for the fission gas and iodine release from the fuel elements during normal and accident conditions of the THTR. The effect of potential manufacture induced coating defects is negligible compared with the contamination.

The frequency plot in Fig. 1 summarizes the quality control data for the uranium contamination of the fuel elements for the initial THTR core (identical product specification and manufacture for initial core and reload fuel elements). The mean value of this most important property amounts to $3.0 \cdot 10^{-4}$ with a standard deviation of $7 \cdot 10^{-5}$ for the single values (3 measurements per batch, maximum batch size : 2000 fuel elements).

The key design data of the THTR fuel elements are given in Tab. 1.

During operation a small fraction of the coated particles is expected to fail. Based on irradiation testing of THTR fuel elements, the in-service failure fraction at the end of the residence time in the core (fast neutron fluence: $4.5 \cdot 10^{21}$ cm^{-2}, E > 0.1 MeV, burnup: 11.5 % fima) is estimated to be $2 \cdot 10^{-3}$. These defectives contribute only < 10 % to the total fission gas release from the THTR core

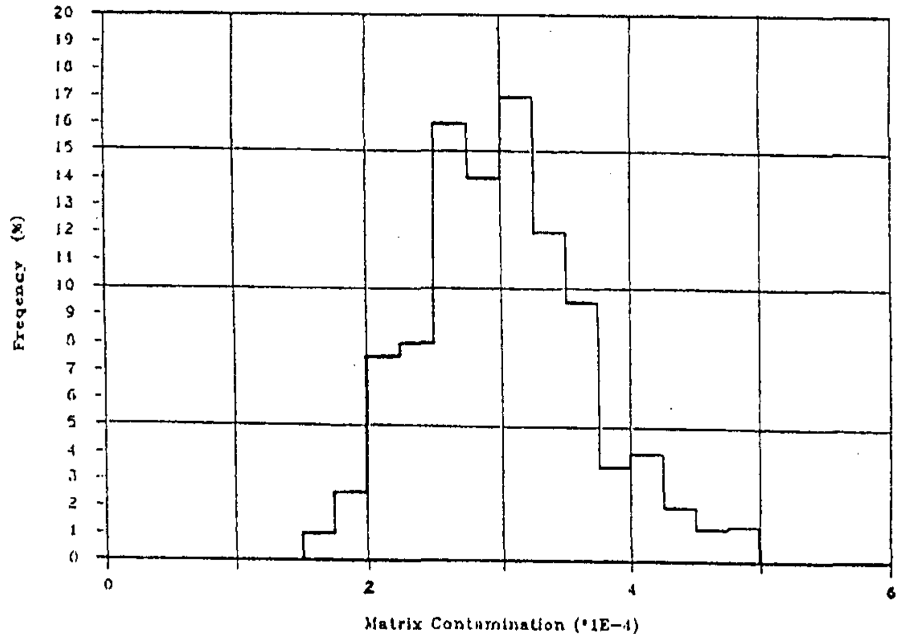

Fig. 1 Distribution of the uranium contamination data for the fuel elements of the THTR initial core (n = 770)

(exception: long-lived Kr 85). Nevertheless, the fission gas release mechanisms must be known equally well for both, the matrix contamination and the failed particles, if unexpected changes of the coolant gas activity are to be explained.

3. Fission Gas Release Models

3.1 Uranium Contamination of Matrix Material

The steady-state fission gas release from THTR fuel elements is calculated with a model which was derived from in-pile measured xenon and krypton release data of a series of irradiation experiments with THTR prototype fuel elements.

Table 1 Design Data of Spherical THTR Fuel Elements

Fuel Element		
Matrix Material		A3-3
Outer Diameter	mm	60
Diameter of Fuel Zone	mm	46
Thorium Loading	g/FE	10.2
Uranium Loading	g/FE	0.96
U-235 Enrichment	%	93
Free Uranium Fraction a)	-	$3.0 \cdot 10^{-4}$
Coated Fuel Particle		
Type		HTI - BISO
Fuel Composition		$(Th, U) O_2$
Kernel Density	g/cm^3	9.9
Dimensions		
Kernel Diameter	µm	400
Porous PyC	µm	80
Sealing PyC	µm	30
Outer PyC	µm	80
Uranium Contamination of Outer PyC	-	$1.0 \cdot 10^{-3}$

a) weight of uranium outside intact coated particles divided by total uranium loading of the fuel element

This model was successfully submitted to the THTR licencing procedure. Details of the model are given in /1/, so that only a qualitative survey is needed here.

The graphitic matrix material of the fuel element is treated as a two-component system. Component 1 may be attributed to the graphite grains of the raw material, and component 2 to the amorphous, non graphitized binder coke between the grains.

The primary fission products are distributed homogeneously in both components by direct recoil. The gas atoms diffuse from the recoil sites to the open porosity of the fuel element. This volume diffusion in both components is described with Booth's "equivalent sphere" model yielding equation (1) for the fractional release of a contamination-born fission gas nuclide from component m:

$$(1) \qquad R_m / B = 3 K_m F_m \cdot (\coth K_m^{-1} - K_m)$$

$$\text{where } K_m = \sqrt{D_m / (r_m^2 \lambda)}$$

- R_m - release rate from component m into the open porosity
- B - contamination induced birth rate in the fuel element
- F_m - fraction of recoil sites in component m
- D_m - diffusion constant of xenon or krypton in component m
- r_m - equivalent sphere radius of component m
- λ - decay constant of a given xenon or krypton isotope.

The two diffusion processes working in parallel in both components are followed by gas phase transport trough the open porosity of the fuel element into the coolant. This process is also described by equation (1) where F_m is given by the sum of the R_m/B-values of both matrix components and r_m equals the radius of the fuel element.

The application of this model with optimized material parameters to the xenon release data measured in THTR irradiation experiments (T = 700°C, p = 1 bar) is shown in Fig. 2. The superposition of the contributions from both components leads to a mean slope of the R/B versus λ-curve of -0.33 in double logarithmic scales.

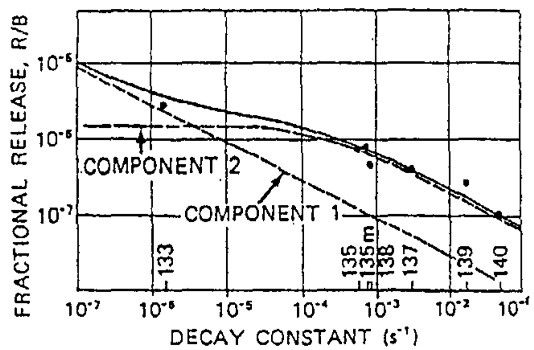

Fig. 2 Calculated and measured xenon release from THTR fuel elements in irradiation experiments at 700 °C, 1 bar (uranium contamination: $3.0 \cdot 10^{-4}$)

The model /1/ was improved by taking into account the fraction of direct recoil atoms which are stopped in the open porosity of the fuel element. This effects only short-lived isotopes at high pressures. Under THTR conditions (700°C, 38 bar) the release of Xe 137 e.g. is increased by about 20 %.

The flow pattern of this enlarged model for the release of contamination induced noble gases is shown in Fig. 3a.

3.2 Failed Fuel Particles

The gas release model for failed fuel particles embedded in the matrix was developed on the base of an irradiation experiment with known contents of artificial defect (Th, U)O_2-particles (laser drilled coatings). The dependence of the in-pile measured gas release data on the radioactive decay constant and on temperature (780°C - 1100°C) is described by a quasi three-component system consisting of

- kernel grains,
- "grains" of the porous buffer layer,
- gas filled, open pores in the kernel and in the porous buffer layer.

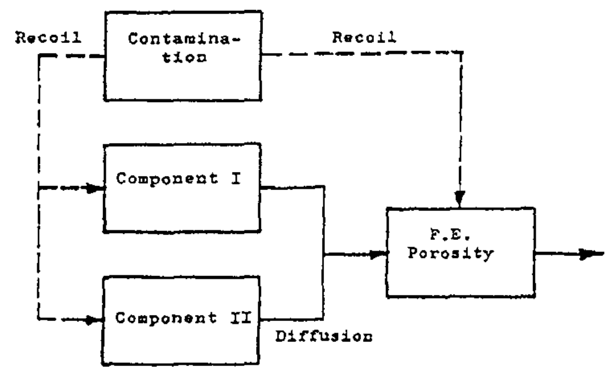

Fig. 3a Gas release from uranium contamination of matrix material

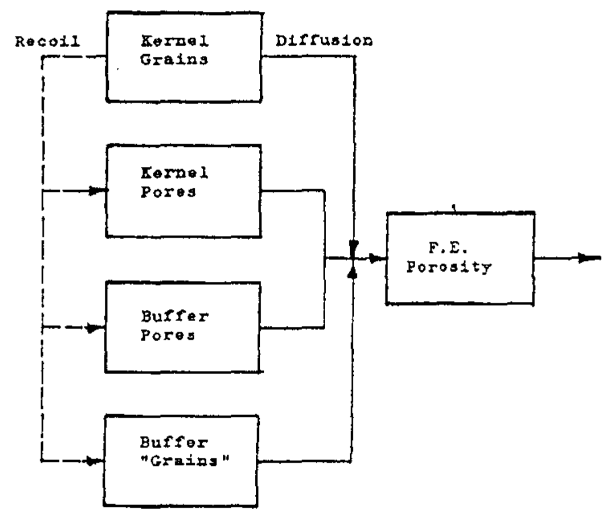

Fig. 3b Gas release from failed fuel particles

The birth rates of primary fission products in these components are calculated with the known relations for the recoil stopping ranges (Bragg-Kleemann Rule). It is assumed that the open porosity of the kernel and the buffer layer is filled with helium at system pressure. In addition to the direct recoil fraction from the kernel into the buffer layer (2.8 % for Xe, 3.7 % for Kr from 400 μm kernels) the manufacture induced uranium contamination of this layer (3 %) is taken into account.

The fractional noble gas releases from the two grain components are treated in the same way as for the contamination-born release using equation (1). The retention capabilities of the open porosity of the kernel and the porous layer as well as the eventual delay of the transport through the coating defects (e.g. hairline cracks) are neglected. The release from the failed particles is followed by the gas phase transport through the open porosity of the fuel elements.

The flow scheme of this release model is depicted in Fig. 3b. Fig. 4 shows the break-down of the release fractions from failed particles (400 μm (Th, U) O_2 kernels) calculated with the above model using optimized material parameters for the krypton isotopes at typical THTR conditions (700°C, 38 bar). At the relatively low temperature of 700°C the diffusional release from the kernel grains (activation energy of the diffusion constant: 310 kJ/mole) is negligible. The kernel release becomes significant only at temperatures above 1000°C.

In preparation of the following discussion of the THTR coolant gas activity, in Fig. 5 the mean slopes of the fractional release curves versus the radioactive half lives $t_{1/2}$ of the various nuclides (from Kr 89 with $t_{1/2}$ = 190s up to Xe 133 with $t_{1/2}$ = 5.3d) in double logarithmic scales are plotted against temperature. It is distinguished be-

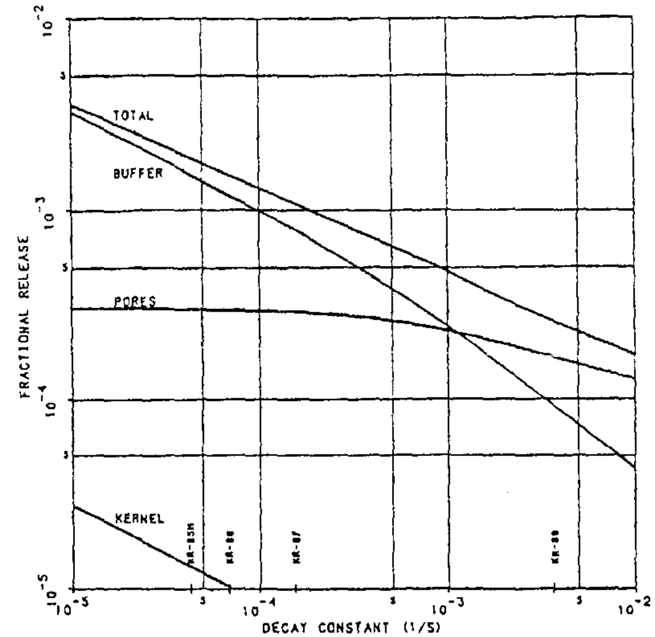

Fig. 4 Calculated krypton release from failed particles in THTR fuel elements at 700 °C, 38 bar

tween pressures at 1 bar (full lines) and 38 bar (broken lines). Both models lead to rather similiar mean log R/B-log $t_{1/2}$-slopes. Neither temperatures, nor pressures influence these slopes significantly.

4. THTR-300 Coolant Gas Activity

4.1 Measurements

The coolant gas activity of the THTR was continuously monitored by β-counting devices. This measurement enables immediate detection of relative changes, but at relatively high uncertainties of the absolute values. Therefore, the

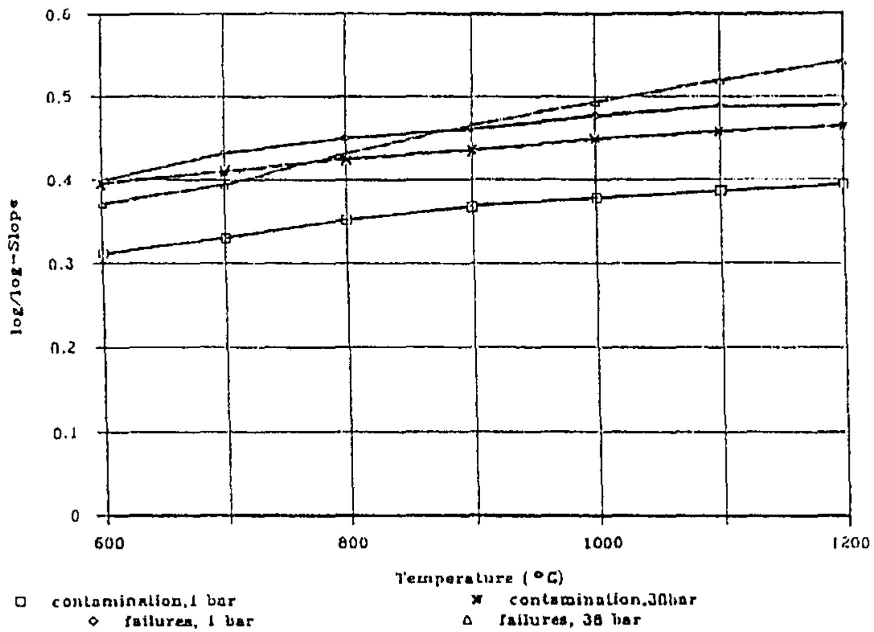

Fig. 5 Slope of log R/B versus log $t_{1/2}$ of the release models for contamination and failed particles (averaged for all nuclides between $t_{1/2}$ = 190 s and $t_{1/2}$ = 5.3 d)

detailed evaluation was entirely based on the γ-spectrometric analysis of isolated gas samples. At regular intervals the activities of 4 krypton and 5 xenon isotopes were measured at power levels from 10 % to 100 % of rated power.

The sum of these activities, covering about 90 % of the total coolant gas activity, is drawn in Fig. 6 against the THTR operation time of 423 fpd (full power days at 762 MW_{th}). For comparison sake, the measurements are related to the actual thermal reactor power (Bq/MW) and to the rated mass flow through the gas purification plant (0.15 kg/s). The load factor of the core and extended shutdown events (marked by arrows) are also shown in Fig. 6.

From 70 to 150 fpd of operation, the activity increased by about a factor 3. From 150 to 300 fpd only small changes occurred. After 300 fpd, a continuous long-term decrease of the activity was observed.

During the rise of the total gas activity the spectrum of the different noble gas nuclides in the coolant changed significantly. The short-lived isotopes increased much steeper than the isotopes with long half-lives. This tendency is demonstrated in Fig. 7 showing the measured fractional releases in dependence of the half-lives at 90 and 150 fpd together with the original calculation (contamination release model). The measured log R/B-log-$t_{1/2}$-slope falls from 0.30 at 90 fpd to 0.15 at 150 fpd. All evaluated slope-values are plotted in Fig. 8 against the operation time. The somewhat arbitrarily drawn average curve in Fig. 8 is obviously (negatively) correlated with the development of the coolant gas activity as shown in Fig. 6.

Any interpretation of the changing coolant gas activity in the THTR can only be regarded satisfactory, if the above tendencies of the slopes are explained consistently.

4.2 Evaluation

From the calculated model results as shown in Fig. 5 it is concluded, that increased temperatures and failed particles embedded in the matrix cannot explain the observed slope changes. This conclusion was confirmed by calculations with revised temperature distributions and substantial failed particle fractions. Furthermore, a sensitivity study of the material parameters of the contamination related release model did not provide a realistic explanation. The measured slope decrease could only be reproduced by complete neglection of the gas phase transport in the open porosity and an increase of the diffusion constant in the amorphous component by two orders

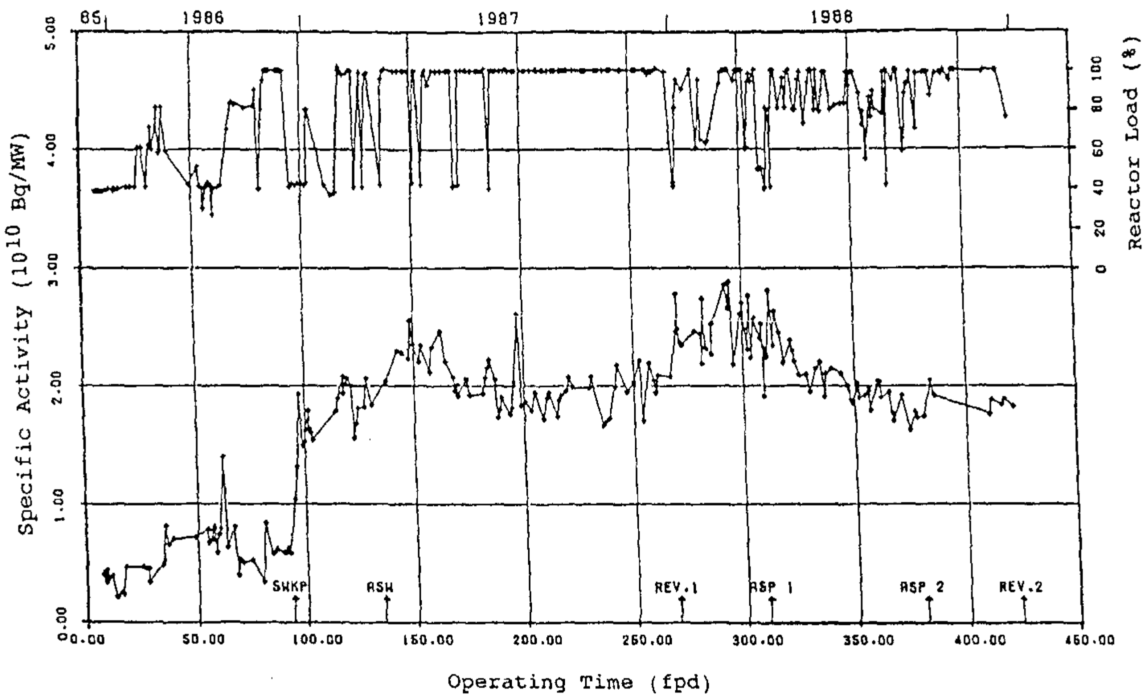

Fig. 6 THTR coolant gas activity
(sum of 9 noble gas nuclides)

of magnitude. However, there is no physical base for this fictitious parameter change. Thus, a new release mechanism had to be postulated.

The background of this model development was the THTR experience that some fuel elements have been damaged by mechanical forces from the control rods inserted into the pebble bed core very frequently under extremely adverse conditions (high sphere loading density in the core) during incommissioning the THTR. The damaged fuel elements (definition: local diameter ≤ 57 mm) are sorted out from recirculation by a special size separator at the core exit. In most of the damaged spheres the fracture surfaces are restricted to the fuel free outer shell. However, some of the damaged fuel elements have fracture surfaces right through the inner fuel zone.

It is assumed that a fraction of the coated particles situated within the fracture surfaces are also mechanically damaged, so that part of their kernel material is exposed

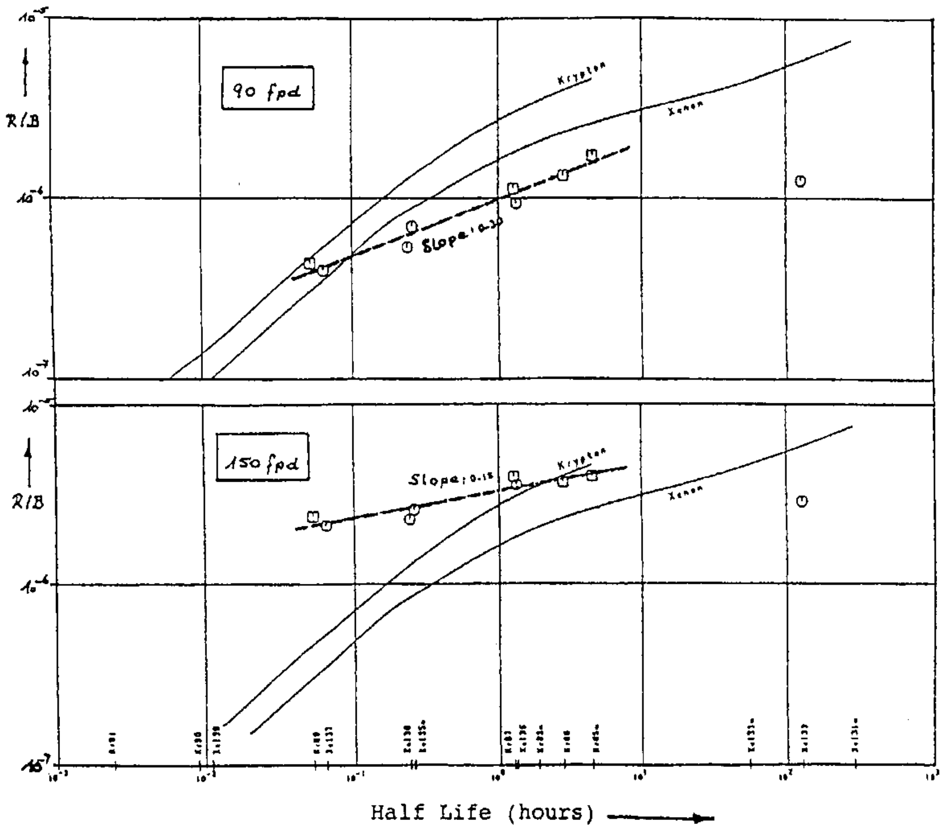

Fig. 7 Measured and calculated release fractions and log R/B-log $t_{1/2}$ -slopes at 90 fpd and 150 fpd

to the coolant. Primary fission products are recoiled into the coolant from these kernels without delay.

Because of the small remaining recoil range in the coolant of about 3 mm (at 38 bar helium), practically all fission products recoiled from the exposed kernel surfaces are stopped in the coolant gas. Thus, an additive release contribution is obtained which is independent of the radioactive half life.

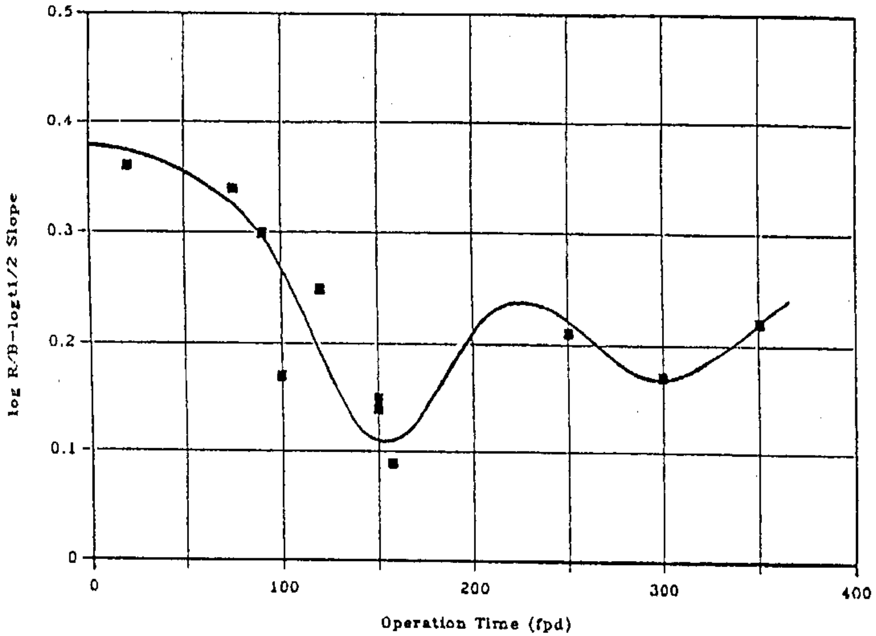

Fig. 8 Development of the log R/B-log $t_{1/2}$ slope during THTR operation

By comparison with the measured activities at 170 fpd a fractional release of 1.4 . 10^{-6} from the total core was evaluated for all gas nuclides. Fig. 9 shows the good agreement between measurement and the enlarged model. The above release fraction of 1.4 . 10^{-6} leads to an upper estimation of the fraction of exposed failed particles of 8 . 10^{-5} in the core, respectively 5 . 10^{-3} of the particles in the damaged fuel elements.

The given additive release model was substantiated by the evaluation of the fraction of damaged fuel elements in the circulating flow of spheres at the core exit as measured after the size separator. Fig. 10 shows the development of this fraction in dependence of the accumulated number of circulated spheres. The measured fraction of damaged fuel elements at the

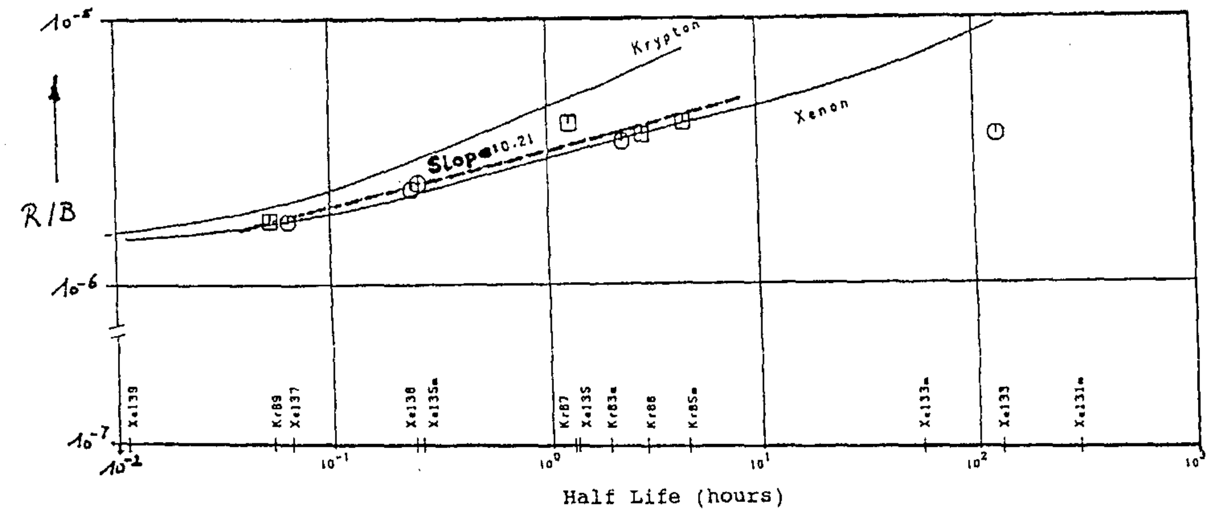

Fig. 9 Measured and calculated release fractions at 170 fpd (enlarged model with constant $1.4 \cdot 10^{-6}$ recoil release)

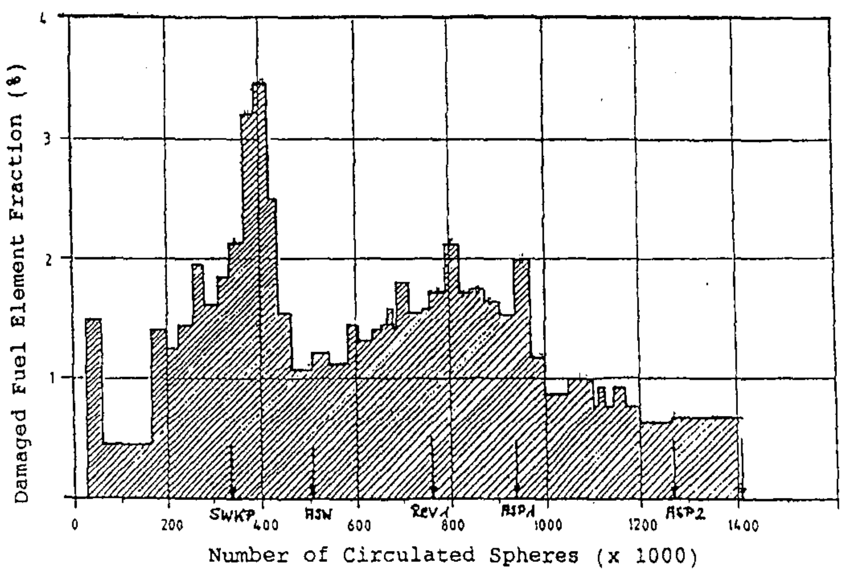

Fig. 10 Fraction of mechanically damaged fuel elements at core exit (measured after size separator)

core exit is positively correlated with the measured coolant gas activity (marked shutdown events in Fig. 6 and in Fig. 10 for orientation). The steepest rise of the activity in Fig. 6 coincides with the maximum damaged fuel element fraction (between SWKP and ASW) in Fig. 10. The long-term decline of the activity is also beginning at the same time as the steady decline of the damage fraction (between Rev. 1 and ASP 1). This correlation proves the presumption that exposed fuel kernels from mechanically damaged fuel elements caused the changes of the THTR coolant gas activity.

5. Conclusions

The model guided analysis of the THTR coolant gas activity enables a comprehensive understanding of the actual status of the fuel element performance. The unexpected activity changes are attributed to defective particles in mechanically damaged fuel elements which are exposed to the coolant. Thus, new release mechanisms hitherto not investigated in preceding experiments can be revealed by operational measurements.

It should be pointed out,that - inspite of the activity increase - the measured sum activity in the coolant agreed with the licenced expected value within a bandwith of ± 20 % (since 100 fpd). The measured activity amounted only to 4 % of the licenced design value.

Reference

/1/ K. Röllig: Nuclear Technology 35 (1977) p. 516

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Research Center Jülich
Institute for Safety Research and Reactor Technology

Report in Part

Pebble Flow Experimental Results Review

A. Kleine-Tebbe presented by Heiko Barnert

Visit of the NRC-Delegation to Germany
Mo 23. to Th. 26. July 2001

January 2001

Topic:
Safety Aspects of HTR Technology

Report Number: FZJ–ISR–RC–5025/2001
in part

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1 Flow behaviour dependence on relevant parameters

1.1 Preliminary Remarks

1.2 Filling Height of the Core

Fig.4 gives a survey of many experimental investigations, which show the transition times of first and last (99%) test balls vs. H/D - ratio (5). In this figure only experiments were considered, where the core models had a bottom with simulated cooling gas holes like THTR. This bottom roughness affects the flow behaviour, as explained in detail in chapter 1.4. The former investigations, reported in (1) to (4), were performed mainly in models with a bottom with smooth surface; this explains the differences between the results in Fig.4 and the earlier investigations just below H/D = 1.

The values in Fig.4 at H/D - ratios >2 were taken from model experiments with 3,4 or 6 outlets in the bottom, and the H/D - ratio was calculated in regard of the region of one outlet by multiplying the filling height H of the model with the number of outlets. This is a simplification, so these values in Fig.4 should be considered as an estimation of the influence of H/D on the flow behaviour.

The results of an INTERATOM experiment with a 1:6 model of the HTR - Modul (H/D = 3.1, see Fig.4) show a more even pebble bed flow than the values calculated from the multiple outlet experiments. If the Modul model had

a smooth bottom without any roughness caused by cooling gas holes, the difference can be easily explained, as pointed out above. Otherwise the calculation of H/D from the multiple outlet experiments may be to pessimistic.

The H/D - ratio of the PO - core is marked in Fig.4.

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2 Mixing of Pebbles between Inner and Outer Zone

2.1 Mixing of Pebbles of the Inner and Outer Zone due to Loading Process

2.2 Mixing of Pebbles of the Inner and Outer Zone by Interchange during Flow

Early experiments, performed in 1970 with the glass pebble model, are shown in Fig. 13 and 14. The glass pebble model was filled with glass pebbles in an immersion liquid, which had an index of refraction like glass. So the model became transparent, and the the flow lines of aluminum test balls give a visual impression of mixing during flow (10).

Fig. 15 gives the result of an experiment in a 1:6 model of the THTR, filled with graphite pebbles. Colored test balls were introduced into the pebble bed surface at 11 positions after every step of 2.5 % circulated core volume (Vc). After circulation so long, that the entire core contained test balls, the core pebbles were removed layer by layer using a vacuum cleaner and the test ball postions were recorded (11).

2.3 Conclusions from the Review of Pebble Mixing

Looking to the experimental results reported in chapter 2.1 and 2.2, the amount of mixing during loading predominates the mixing during flow.

This must not be so at the PO - reactor, because the outer and inner loading cones have the same height and therefore the possibility of the transition of fuel elements from one to the other zone will be reduced, compared to THTR (Fig.17).

The mixing area will depend also on the number of loading tubes in the outer zone, as shown in Fig.17. The lower the number of outer charging tubes is, the more the circle between inner and outer zone degenerates to a serpentine, increasing the mixing. Fig.17 shows the situation in the PO - reactor with 6 and 9 loading tubes outside.

3 Angle of Inclination of the Loading Cones in Dependence on relevant Parameters (Friction Coefficient, Pebble Density, Dropping Speed)

The first measurements of the cone inclination dependence on the density of the pebble material were reported in (1). The cone inclination varied from 29 ° (Graphite, $\gamma = 1.6$ g/cm) to 20.5 ° (Steel, $\gamma = 7.8$ g/cm).

Because these experiments were performed with a dropping speed near zero, the results are not representative for reactor conditions, where the dropping speed will be much higher, according to a height of fall of about 1m.

Further model experiments with more realistic loading conditions, regarding the dropping speed, are reported in (5) and shown in Fig.18. These experiments concerned mainly the influence of the friction coefficient, the material density varied only by a factor of 2. Looking to these experimental results, a loading cone inclination angle of 25° - 26° can be expected.

4 Flow Lines

Fig.15 shows also the flow lines of the THTR. In this figure is marked the vertical part of the flow lines, 40 pebble diameters.

From this measure in Fig.15 the curved part of flow lines near the core bottom can be calculated to about 35 pebble diameters or 2.10 m.

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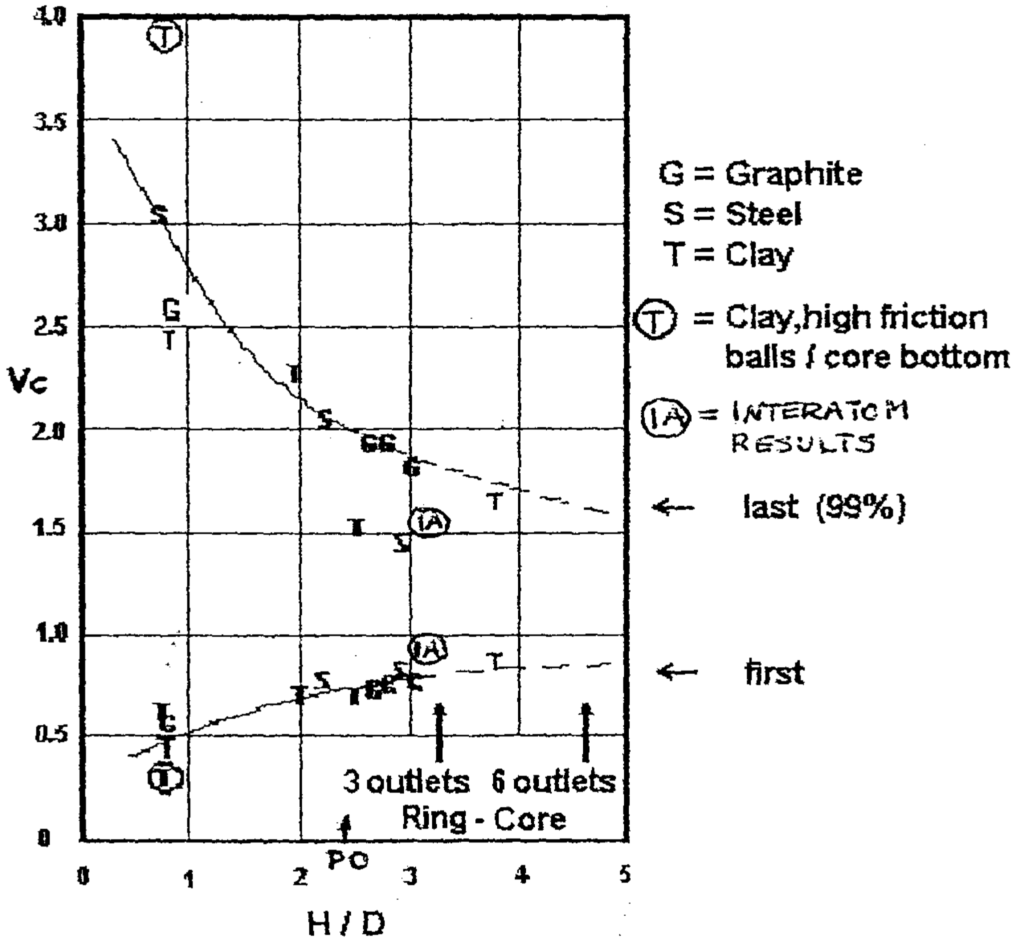
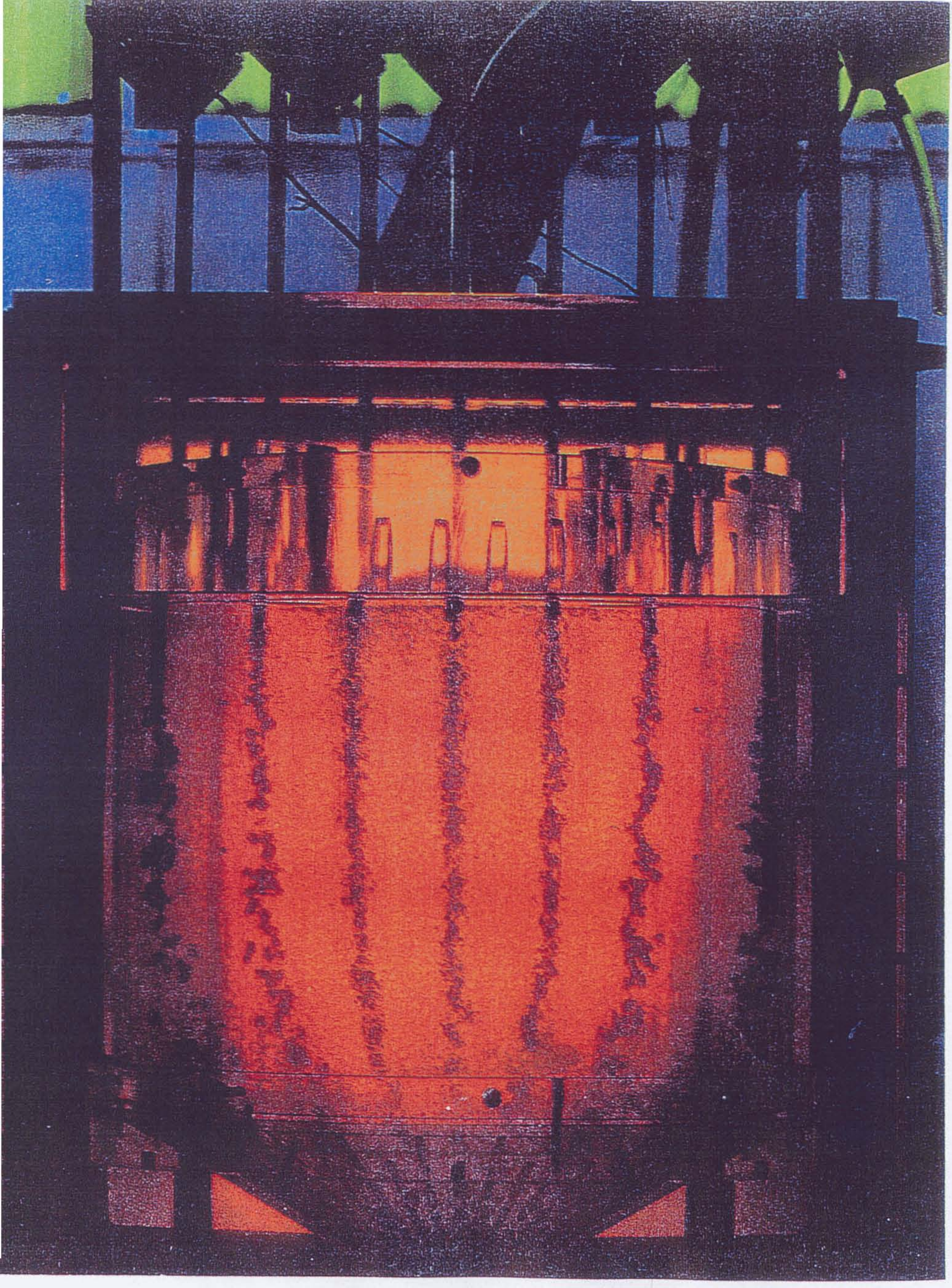


Fig.4 Transition Times of Pebbles through the Core vs. H / D



Flow lines of Test Balls in the Glass Pebble Model of AVR

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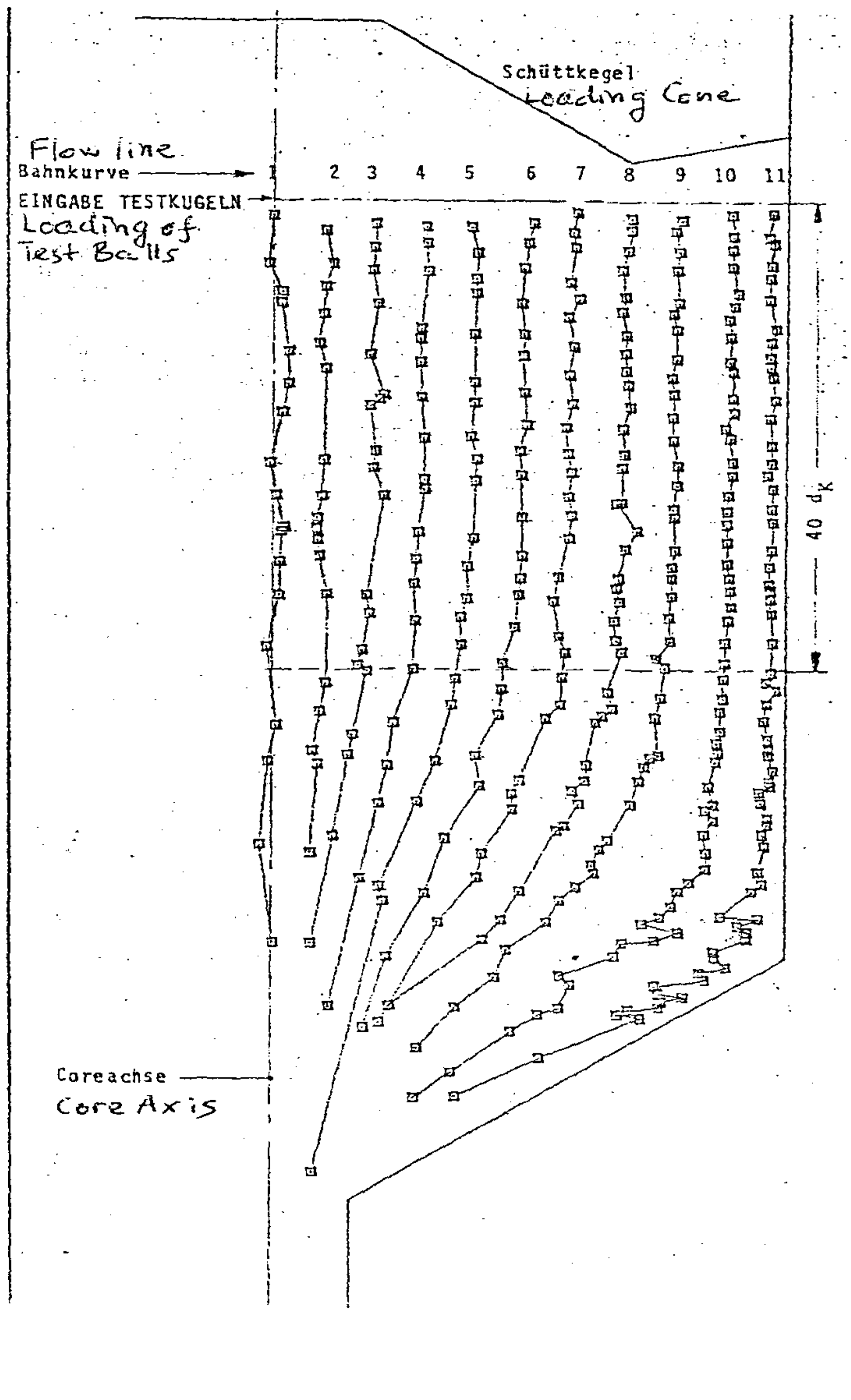

Fig.15 Flow Lines of Test Balls in 1:6 Model of THTR

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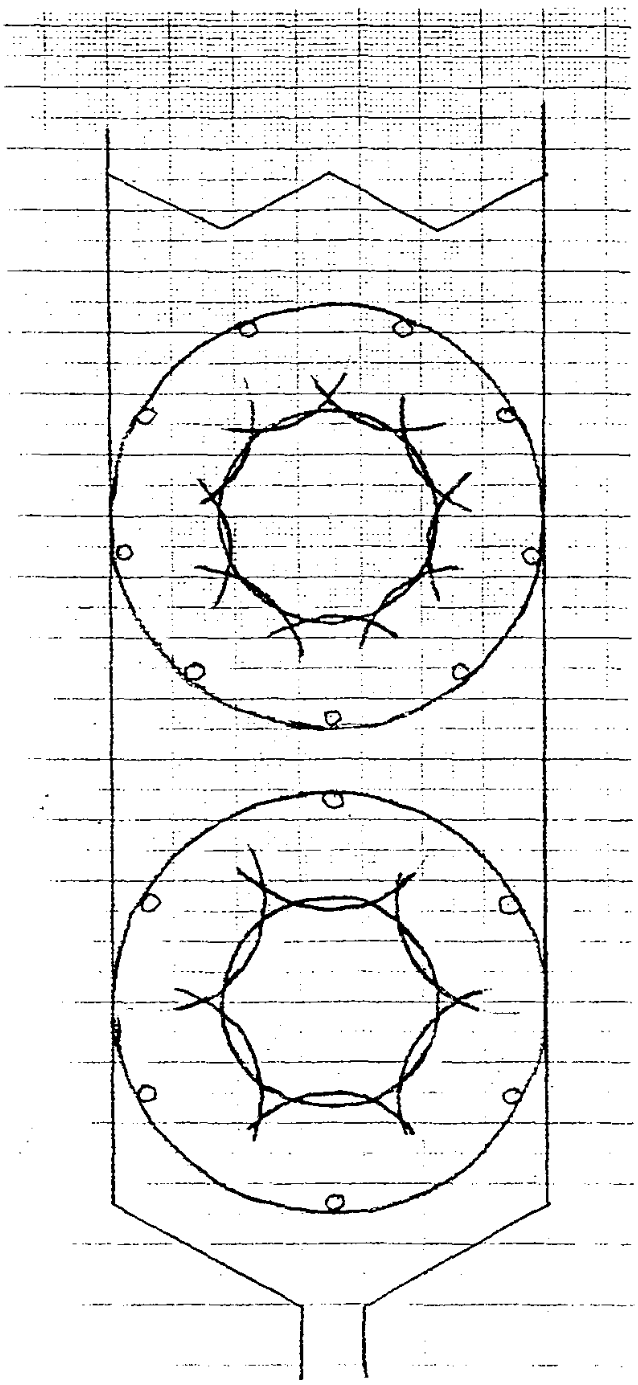

Fig.17 Geometry of PO - Core with
Two Arrangements of
Loading Tubes

DECOMMISSIONING OF THE THORIUM HIGH TEMPERATURE REACTOR (THTR 300)

XA9848059

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Abstract

The prototype Thorium-High-Temperature-Reactor (THTR 300) was decommissioned using the option of safe enclosure. Decision was made in 1989 and safe enclosure was reached in February 1997, followed by up to thirty years of operation of the safe enclosed plant.

I. Introduction

The pebble bed high temperature reactor THTR 300 was shutdown on 01.09.89 after more than 16,000 h in operation. The THTR 300 is a prototype reactor project that is jointly sponsored by the Federal Republic of Germany, the state North Rhine Westphalia and the operator Hochtemperatur-Kernkraftwerk GmbH (HKG). The public financiers of this prototype reactor and the operator could not solve the financial problems for continued operation of this technically intact plant. The decommissioning decision had not been expected at the time by the operator. This is why safe enclosure the German term for SAFSTOR turned out to be the only technical solution for quick decommissioning of the plant, apart from financial reasons and the non availability of a final repository . The plant is intended to be dismantled after about thirty years of safe enclosure, provided respective funds are available. The decommissioning was done in three steps that were mostly scheduled one after the other (FIG. 1), /1/.

II. Description of the Work

A. SHUTDOWN OPERATION

Step 1 has included the conversion of plant operation from the power mode to the shutdown regime to keep the operating costs of the plant low until the license required under the Atomic Energy Act for the core unloading has been granted.

In shutdown operation, the shutdown rods were fully inserted and locked to prevent withdrawal. Recriticality of the reactor core was thus precluded.

Owing to the long outage period, which started when the reactor was shut down for the scheduled maintenance on September 29, 1988, forced residual heat removal by operating systems was not longer required. These systems have been taken out of service by depressurization, removal of operation media, cutting off the energy supply and by blockage. These measures also apply to the prestressed-concrete reactor vessel (PCRV) with the primary system in which the helium was replaced by air/nitrogen.

This lead to a reduction in the number of yearly inservice inspections from about 4,000 to 2,000. Moreover, savings have been achieved in terms of insurance, plant security, maintenance and through labor reduction, so that the monthly operating costs of about 9 million DM in power operation could be decreased to 5 million DM in shutdown operation.

B. CORE UNLOADING

Step 2 was the core unloading, according to Section 7 (3) of the Atomic Energy Act a prerequisite for the establishment of the safe enclosure /2, 3/. For the THTR 300 this meant that about 580,000 irradiated fuel elements, which still were in the reactor core, had to be unloaded. This could only be done by the complete unloading of the core, including the absorber and graphite elements that remained there, too. A worldwide first of its kind activity to a pebble bed reactor (*FIG. 2*).

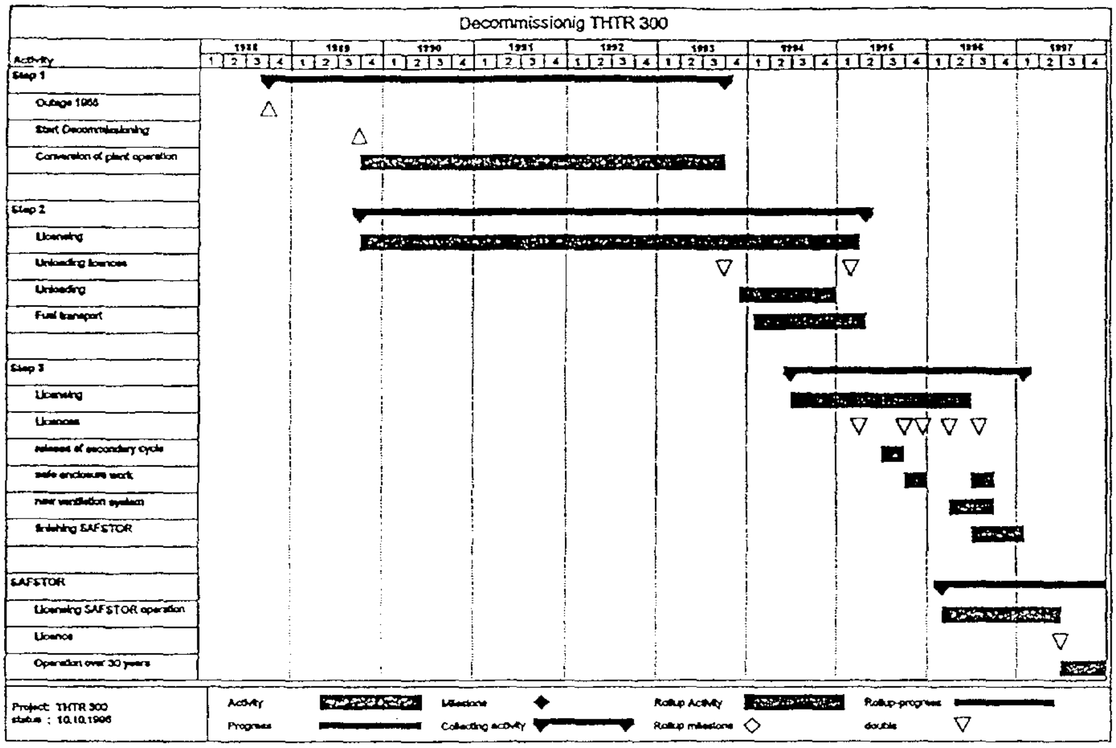


FIG. 1 *Time schedule, Decommissioning THTR*

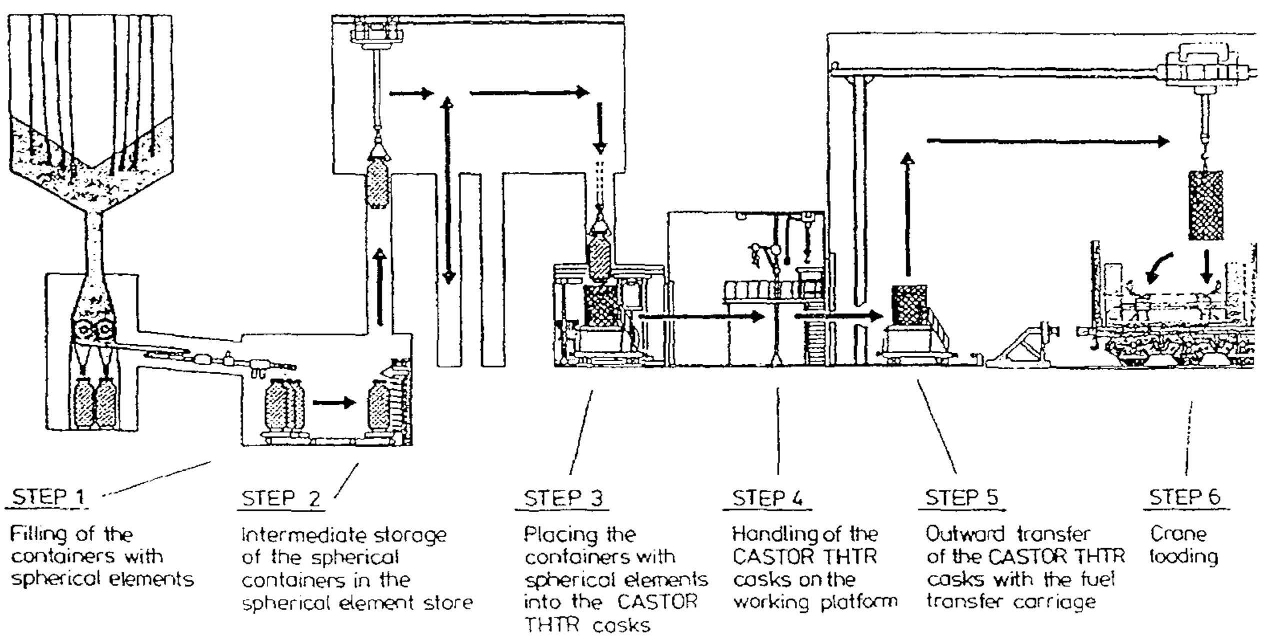


FIG. 2 *Core unloading, Management of the spherical elements*

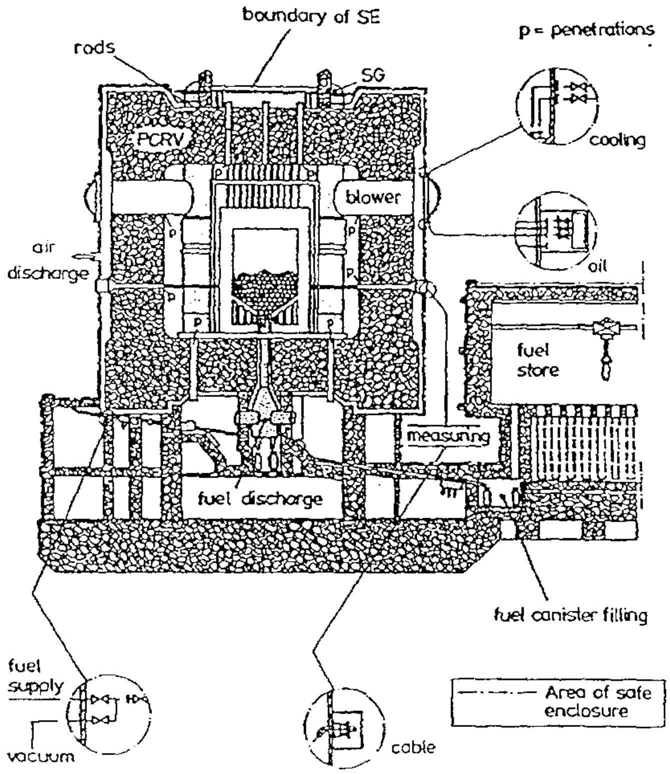


FIG. 3 *Closing scheme of penetrations through the safe enclosure*

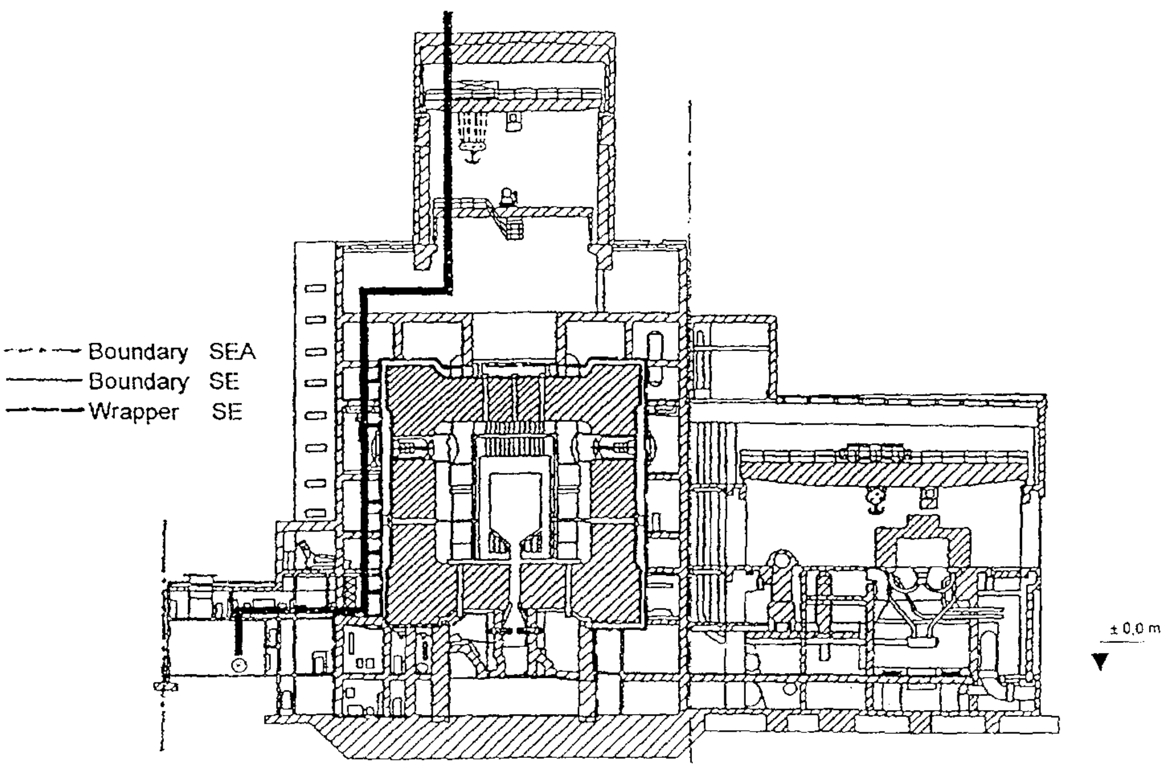
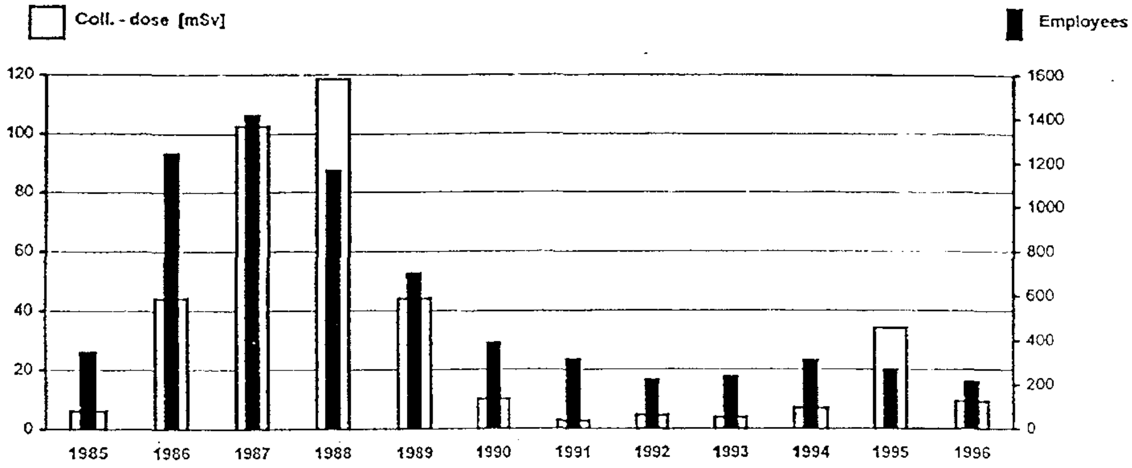


FIG. 4 *Safe enclosure concept (Reaktor building sectional drawing)*



1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	
6,1	44,1	102,5	118,46	43,98	9,89	2,54	4,5	3,71	6,89	34,19	9,38	Coll. - dose [mSv]
352	1249	1418	1172	705	392	317	224	237	311	270	217	Employees

FIG. 5 *Collective dose from 1985 - 1996, Decommissioning during 1990 - 1996*

The respective license was granted after a four years long lasting licensing procedure in October 1993 (*TAB. I*) The unloading itself was executed in a one year period from Dec. 1993 till Dec. 1994 accompanied and followed by regularly (weekly) fuel transport campaigns with CASTOR casks to the Ahaus fuel interim storage facility (in total fifty-seven transports without any rumors as unfortunately on Gorleben-CASTOR transports).

Also the nearly hands-on decommissioning of the small burn-up measuring reactor, used for distinguishing fuel absorber and graphite elements and monitoring the burn-up of fuel elements containing 3.6 kg U^{235} in form of high enriched U-Al-fuel, took place just after finishing of core unloading in early 1995.

C. ESTABLISHING THE SAFE ENCLOSURE

Step 3, the establishment of safe enclosure, was started also in 1995 after applying for in 1994/95 and granting of attachments to the core unloading license in 1995 (*TAB. I*). The main steps undertaken and finished by a general contractor even in 1995 /4/ were

- enclosing the prestressed-concrete reactor vessel by cutting and sealing all approx. 2,000 penetrations (*FIG. 3*)
- sealing all primary circuit system components
- establishing of an additional enclosure for those sealed components by using the existing vented containment as a type of air flow guidance envelope
- release of the water-steam-cycle with turbine and generator and the four emergency diesel generators from the restrictions of the Atomic Energy Act
- preparation work for the establishment of a new ventilation system tailored to the requirement for the safe enclosure operation.

In April 1996 the first part of the next license (safe enclosure establishment and pre operational tests) was granted (see also *TAB. I*) concerning mainly the erection of the new ventilation and the exhaust air measuring system. That work was finished on schedule in September 1996.

TABLE I. LICENSING, DECOMMISSIONING THTR 300

step	application	license
Core unloading	19.12.89	22.10.93
Sorting of some operating elements	14.01.94	09.02.95
Closure of PCRV, steam cycle	13.07.94	23.05.95
Closure of wrapper SE	29.06.95	02.10.95
Dismantling the He-purification	04.09.95	27.10.95
Erection of new ventilation	28.06.94/ 06.12.95	26.04.96
Establishment of SE	28.0694/ 01.02.96	15.07.96
Operation of safe enclosed plant	14.05.96	21.05.97

TABLE II. OVERALL COST, DECOMMISSIONING THTR FROM 1990-2009

	Mio. DM
Waste	253.0
Experts	55.0
Contractors	112.0
Operation 1990 - 2/1997	288.5
Operation 3/1997 - 2009	35.0
Financing	30.0
Total	773.5

TABLE III. SOLD EQUIPMENT OF THTR 300

	Mio. DM
Secondary cycle - steamturbine - generator - auxiliaries	15.0
Transformer	4.1
4 emergency diesel generator sets	3.0
Spare parts, tools etc.	1.3
Electrical-, communication-, radiation monitoring equipment	1.2
Total	24.6

The second but more important part of this license was granted in July 1996, containing the main steps for the establishment of safe enclosure and allowing to:

- dismantle the liquid waste store and evaporation system, decontamination shop and the like
- adapt the power supply
- dismantle contaminated equipment outside safe enclosure that doesn't fulfill the requirements of this area later concerning contamination limits
- adapt the drainage of the building
- decommission all other systems that are not needed for operation of safe enclosure
- install new control equipment fitting with the new operation tasks
- release all buildings of the site (except the three buildings of the safe enclosed plant: reactor hall, reactor operating and auxiliary building) from the restrictions of the Atomic Energy Act.

One important issue of this phase was the conversion of the major part ($\approx$ 80%) of the controlled area inside the safe enclosed plant into an "operational supervised" area with a dose level less than 2μSv, which can be entered for maintenance purposes without health physics monitoring. This area is the area outside the "envelope of safe enclosure" but inside the safe enclosed plant (*FIG. 4*), /4/.

This last but one part of step 3 took approximately eight months for execution and ended with the THTR 300 in safe enclosure (*FIG. 4*), comparable with the US-SAFSTOR or the IAEA passive SAFE STORAGE option at end of February 1997.

The last part of Step 3 was given on the way for licensing in May 1996. The applying documents like final safety analysis report, operating manual for thirty year operations of the safe enclosed plant and the like were checked by the experts. The license was granted on May 21, 1997.

Results

Work executed since 1990, even core unloading, resulted in yearly collective doses of personnel less than those in the years of operation (*FIG. 5*). The highest value during decommissioning occurred in 1995 due to the hands-on decommissioning of the small burn-up measuring reactor and the enclosing of the PCRV-penetrations.

Operating personnel could be reduced during step 1 + 2 only from 10 to a 8 men shift. Starting step 3 a further reduction to 5 men was allowed and at the end of second part of step 3 (safe enclosure established) only one control panel has to be checked by the site guard (24 hours a day). The personnel will then consist of the operator's plant manager plus one engineer and four additional standby service engineers on a call and contract basis. Necessary inspections will be done by specialized and certified companies on contract basis.

Then the yearly operating costs are reduced from more than 50 million DM per year during step 1, step 2 and first and second part of step 3, to 1.5 million DM per year. The overall costs of the decommissioning (1990 - 2009) sum up to 773.5 million DM and include costs of fuel transport and storage and also other waste handling and mandatory financing of final storage and financing of the project during 1990 - 2009 (*TAB. II*). The design of the THTR 300 that has for the secondary cycle a similar layout as fossil fueled power plants enabled the operator to sell many of the used components and spare parts to make financing of the decommissioning easier (*TAB. III*).

Starting into decommissioning of a nuclear power plant without chances of preplanning causes two to three years additional project execution time, equivalent to approximately 250 million´s DM in the case THTR. This is why latest schedules for decommissioning up to green field include preplanning phases of up to four years.

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/5/ N. Röhl, D. Ridder, D. Haferkamp,
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Waste Management (spent HTR-fuel elements)

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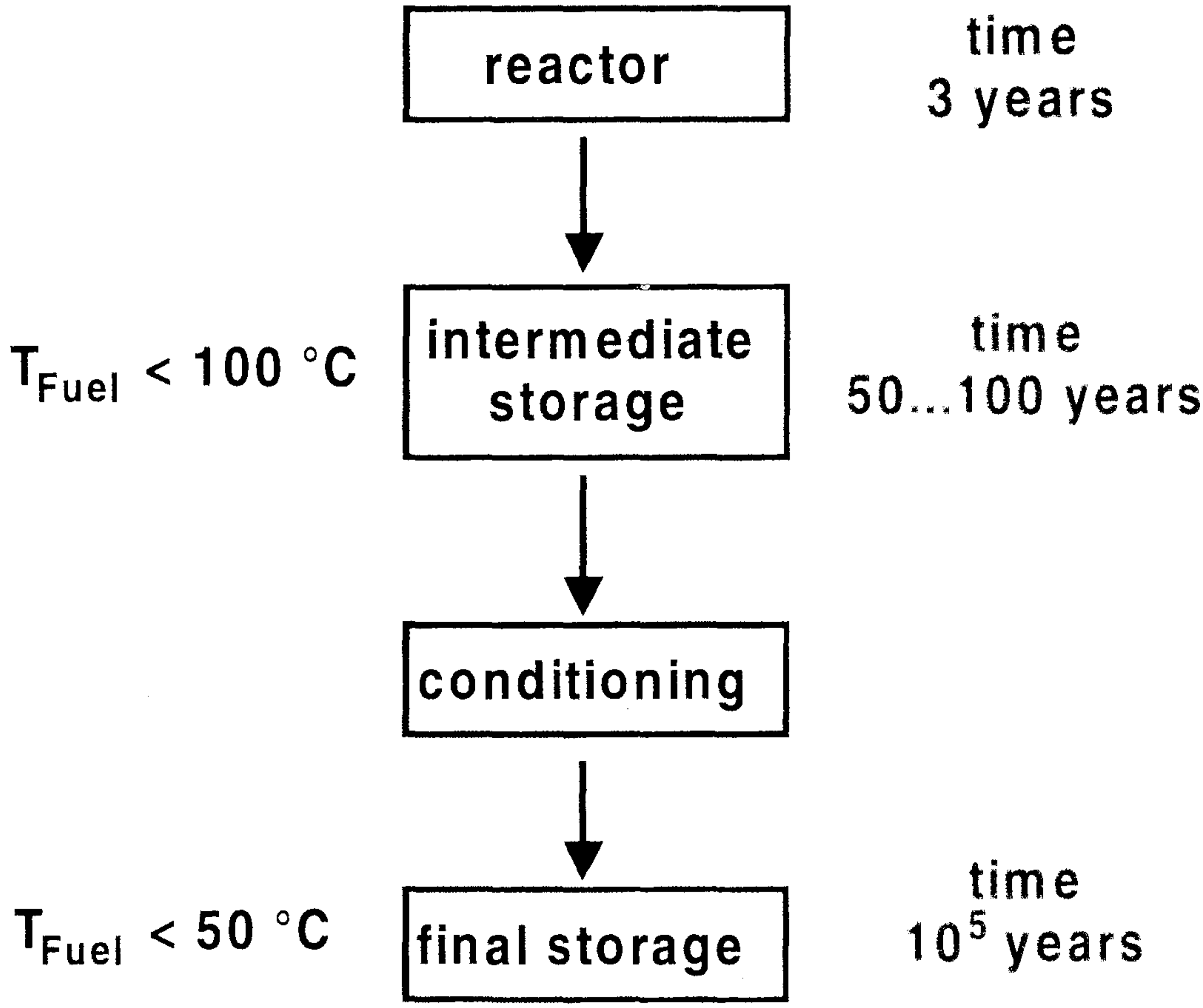
Research Centre Jülich

July 25, 2001

on the occasion of the visit of the NRC-Delegation to Germany on the Topic
Safety Aspects of HTR Technology (GRS Cologne/FZJ Jülich)
July 23 to July 26, 2001

Storage of spent HTR-fuel elements

overview:



- **long storage time (~100 years) optimal for intermediate storage**
- **ceramic fuel elements resistant in final storage**

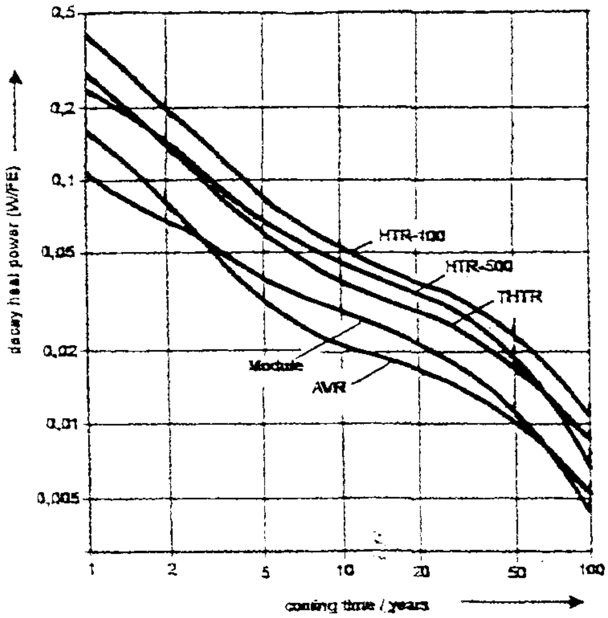


Abb. 3.11: Zeitliche Verläufe der Nachwärmeproduktion bei verschiedenen HTR-Konzepten mit kugelförmigen Brennelementen

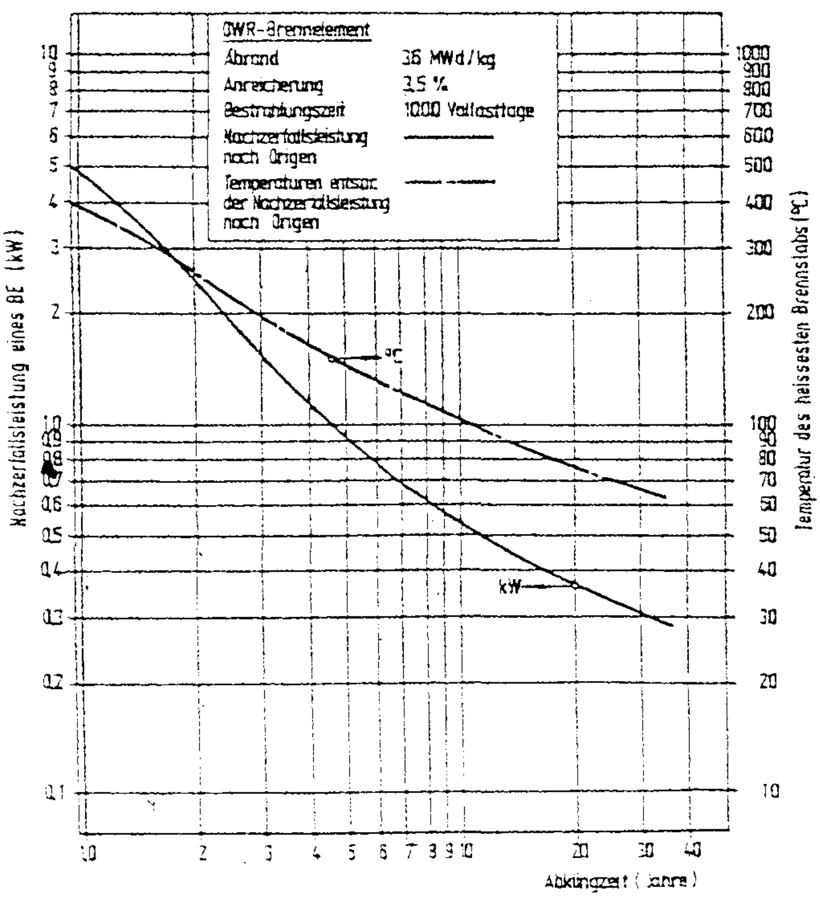
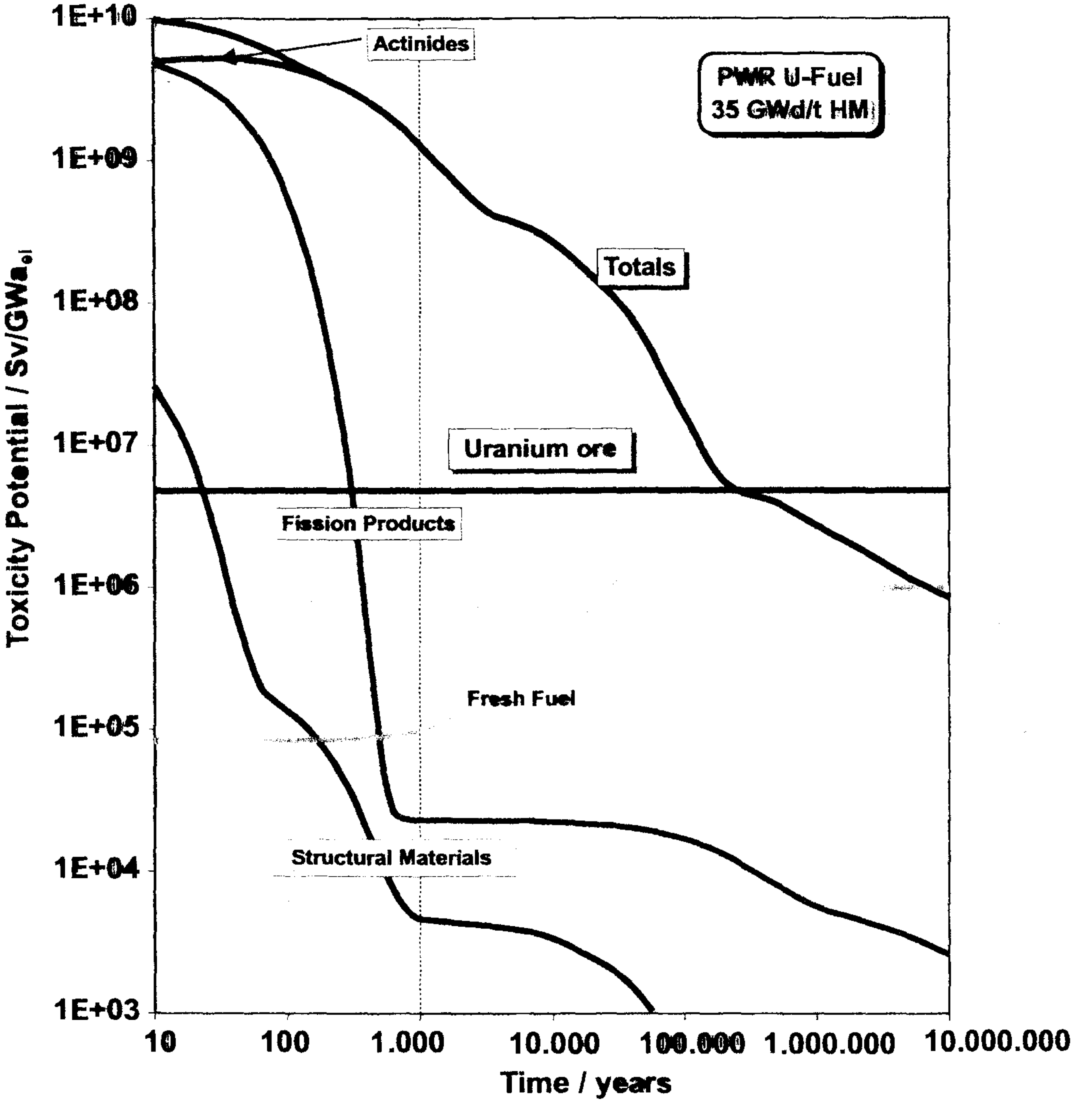


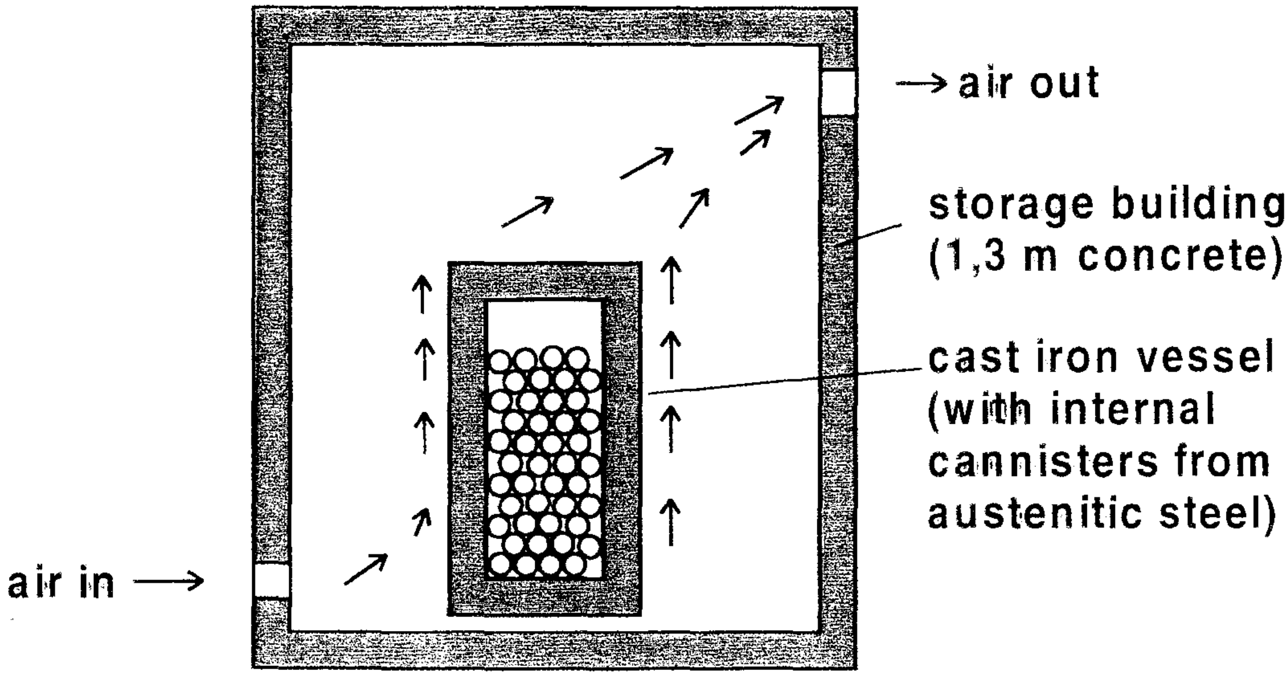
Abb. 3.12: Zeitliche Verläufe der Nachwärme und der Temperatur des heißesten Brennstabes für LWR-Brennelemente (CASTOR-IB, 4 DWR-BE, max. 25 kW, 2 t Uran



Actinides
PWR U-Fuel
35 GWd/t HM
Totals
Uranium ore
Fission Products
Fresh Fuel
Structural Materials
1E+10
1E+09
1E+08
1E+07
1E+06
1E+05
1E+04
1E+03
Toxicity Potential / Sv/GWa$_{el}$
10
100
1.000
10.000
100.000
1.000.000
10.000.000
Time / years

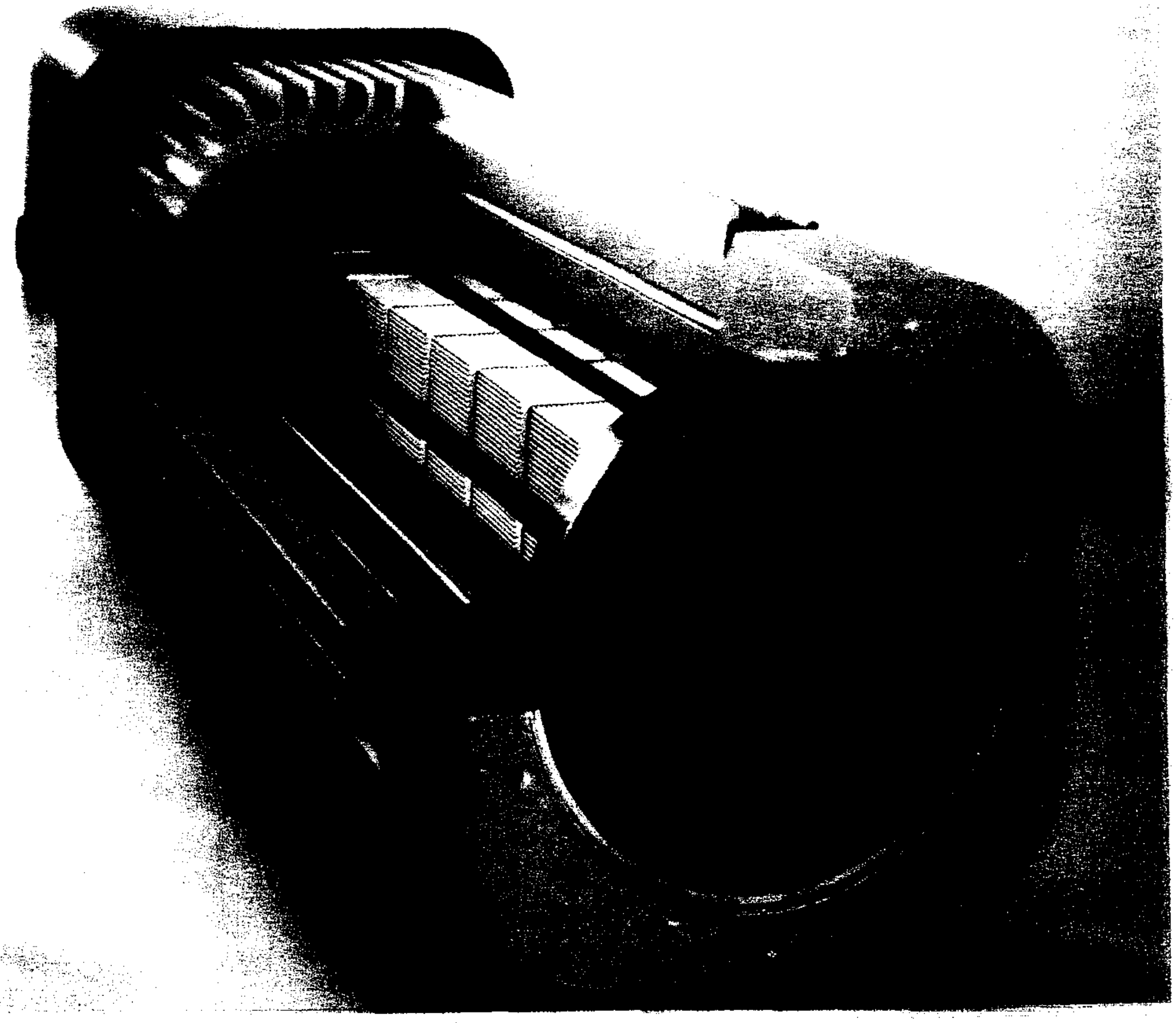
Intermediate storage of spent HTR-fuel elements

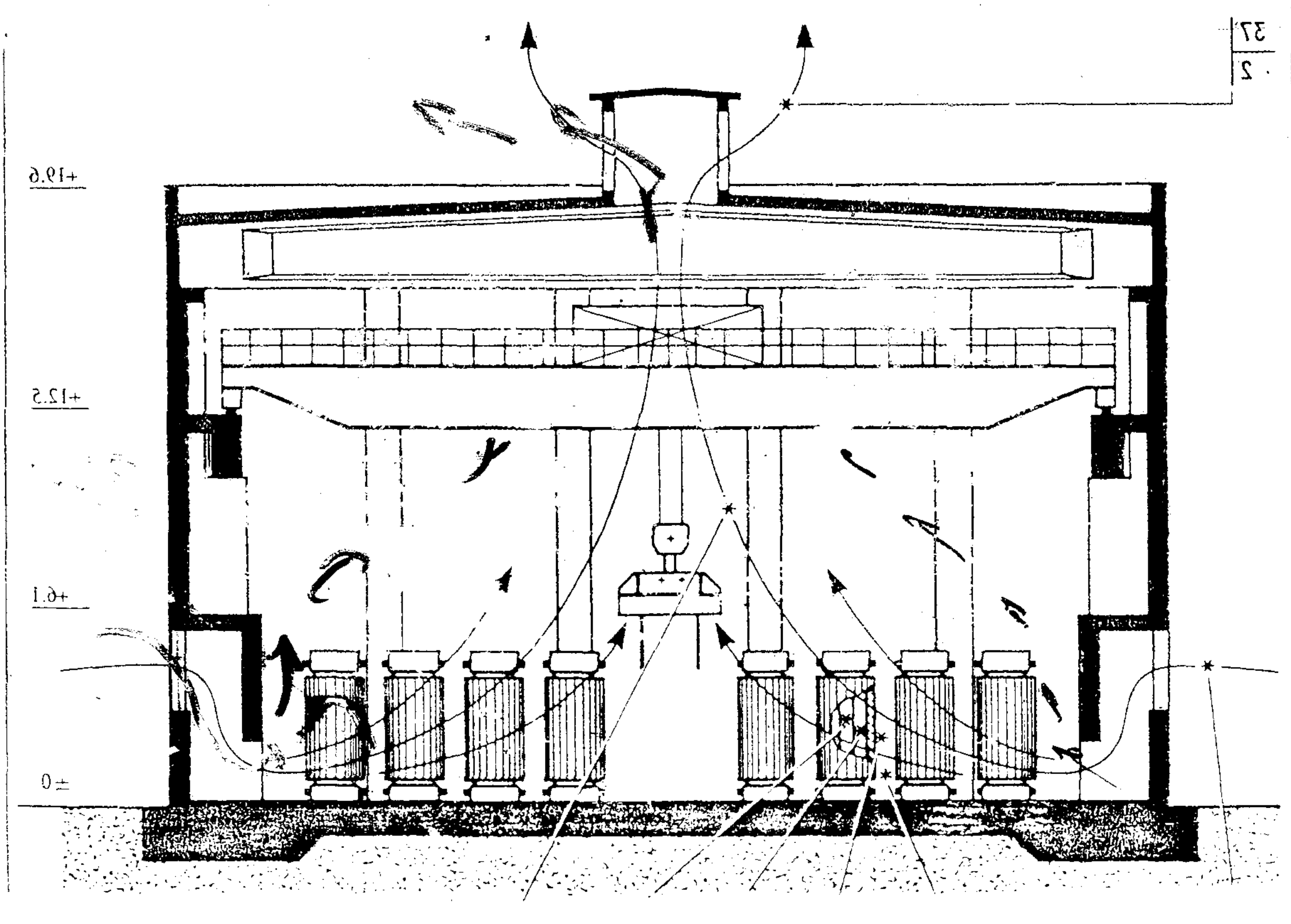
concept:

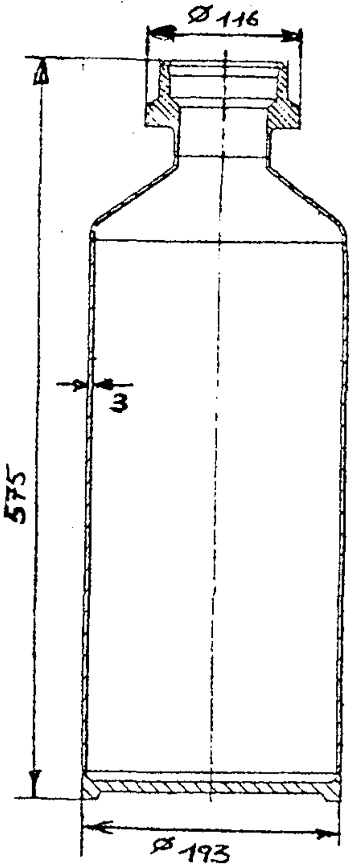


conclusions:

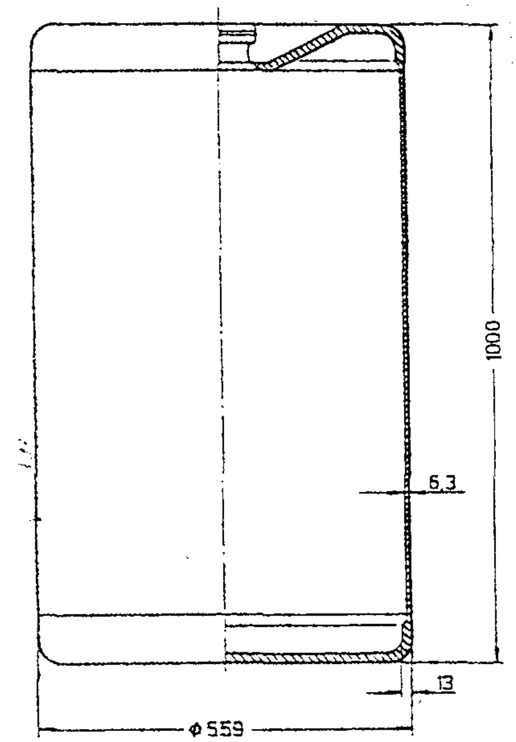
- fuel temperatures below 200 °C
- system protected against outer impact
- there are no accidents caused by internal or external reasons, which result in non- allowable release of fission products („catastrophe-free nuclear technology")
- storage time of 50 to 100 years optimal releated to total costs of spent fuel storage



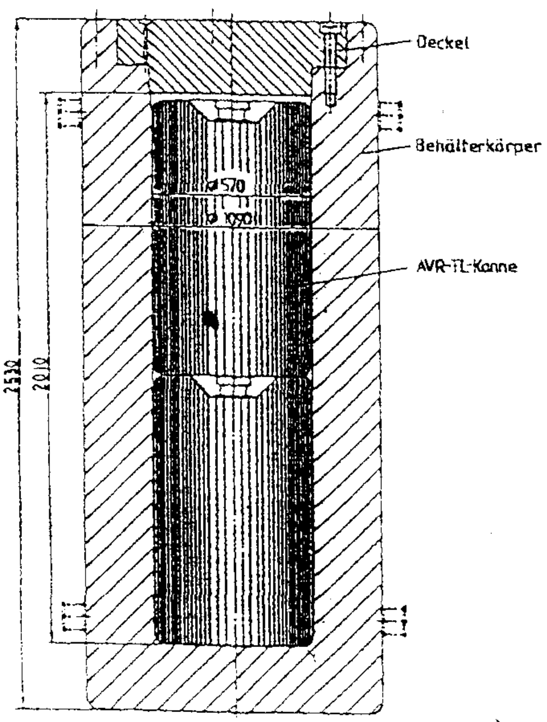

37
2
+19.6
+12.5
+6.1
±0



a)



b)



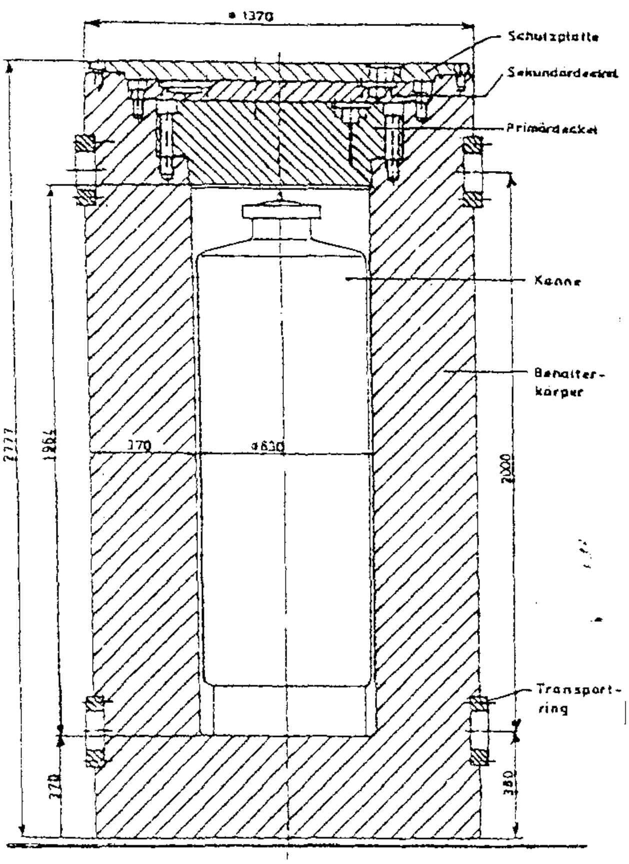
c)

AVR-Castor-Behälter:

Beladung:	2 AVR-Trockenlagerkannen
Inhalt:	420 BE
Innendurchmesser:	570 mm
Innenhöhe:	2010 mm
Wandstärke (Mantel):	300 mm
Wandstärke (Boden)	315 mm
Material:	GGG 40
Gewicht:	18,25 t
Wärmeleistung	800 W

d)

Abb. 3.7: Entnahme und Lagerung von abgebrannten AVR-Brennelementen
a) AVR-Entnahmekanne (50 BE)
b) AVR-Trockenlagerkanne (2.100 BE)
c) AVR-T/L-Behälter (AVR-Castor; 4.200 BE)
d) Daten eines AVR-Castor-Behälters

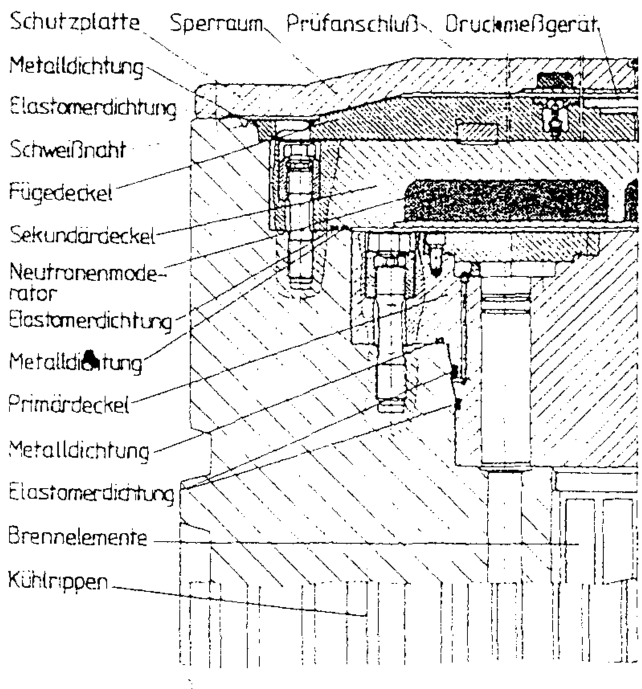


a)

Daten eines
THTR-Castor-Behälters

Zahl der Kannen/ Behälter:	1
Fassungsvermögen einer Kanne:	2100
Innendurchmesser T/L-Behälter:	630 mm
Innenhöhe:	1960 mm
Wandstärke:	370 m
Deckel:	2
Gewicht des Behälters:	t
Material:	Sphäroguß

b)

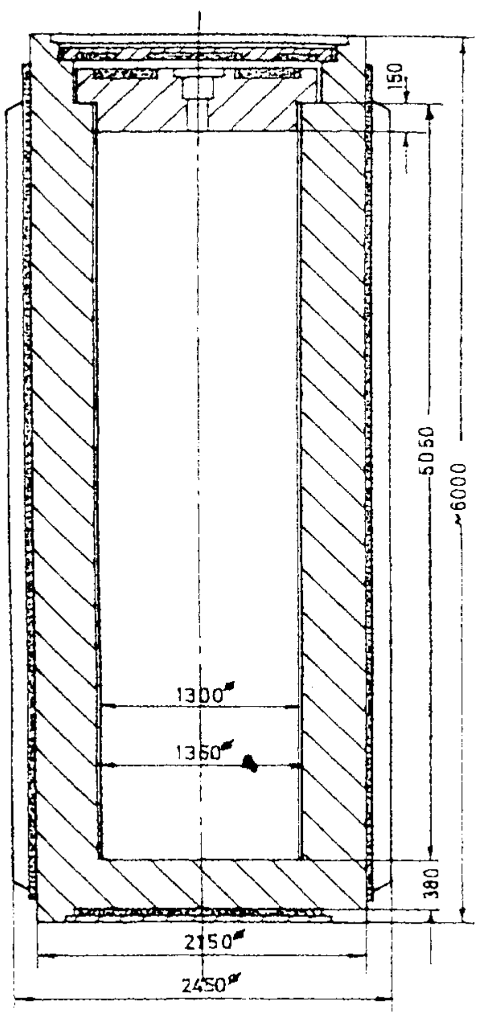


c)

Abb. 3.5: Zwischenlagerung abgebrannter THTR-Brennelemente im externen Zwischenlager
a) THTR-T/L-Behälter (THTR-Castor)
b) Daten des THTR-Castor
c) Deckelkonstruktion des THTR-Castor

Entsprechende Hilfseinrichtungen zum Positionieren der T/L-Behälter innerhalb der Halle, Überwachungseinrichtungen zum Überprüfen der Doppeldeckel sowie Einrichtungen zur Umgebungsüberwachung sind wie üblich vorhanden.

Ein hinreichend großes Freigelände umgibt die Lagerhalle und gewährleistet den üblichen Zugangsschutz und trägt dazu bei, daß die Dosisleistung am Anlagenzaun unterhalb vorgeschriebener Werte gehalten werden kann ($\dot{D}$ < 10 μSv/h).
Zukünftig evtl. noch niedrigere Werte können durch verstärkte Abschirmung oder ein größeres Gebäude realisiert werden.



a)

Brennelement:

Brennstoffzyklus:	LEU
Partikeltyp:	TRISO
Schwermetallgehalt:	7 g SM/BE
Anfangsanreicherung:	8 %
mittl. Abbrand:	80 000 MWd/tSM

Reaktor:

therm. Leistung:	200 MW
mittlere Leistungsdichte:	3 MW/m³
Lastfaktor:	0,8
jährl. BE-Menge:	105 000 BE/a

Zwischenlagerbehälter:

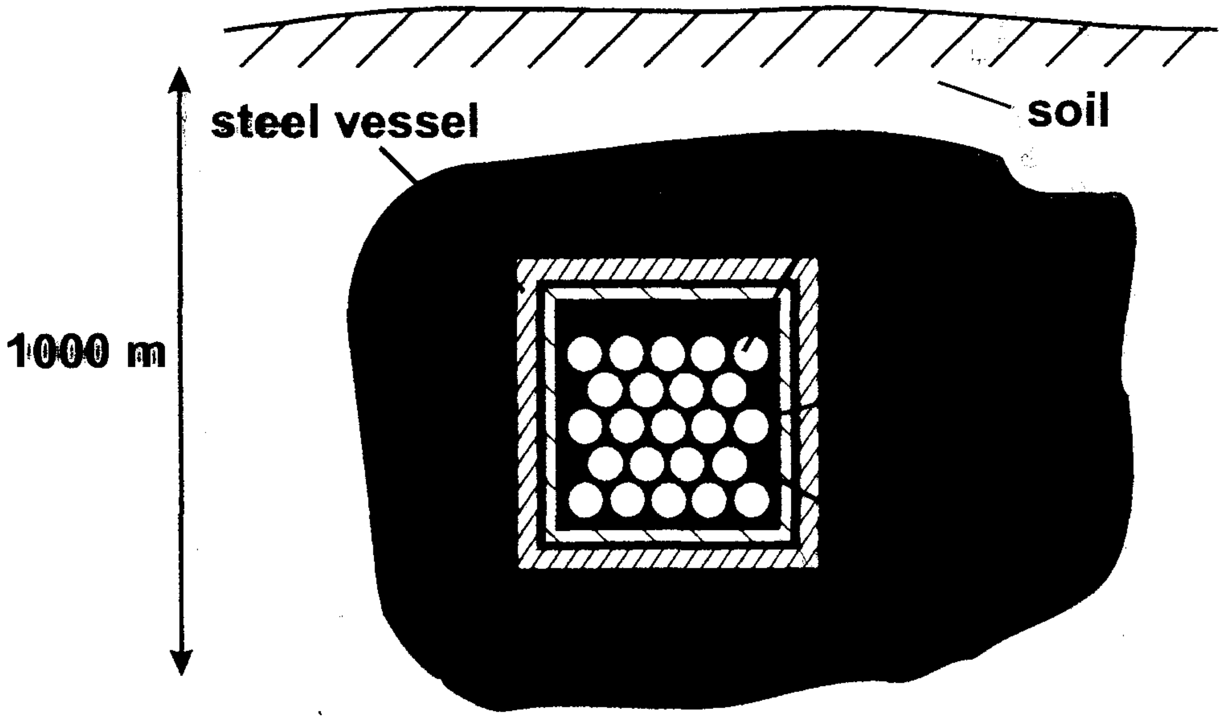
Höhe:	6 m
Innendurchmesser:	1,6 m
Wandstärke:	0,4 m
Gewicht:	113 t
Werkstoff:	Sphäroguß
Abschlußkonzept:	2 Deckel mit Dichtungen

b)

Abb. 3.1: Zwischenlagerung abgebrannter Brennelemente beim MODUL-HTR von INTERATOM/SIEMENS
a) Zwischenlagerbehälter
b) Daten von Brennelementen und Zwischenlagerbehälter

Final storage of spent HTR - fuel elements

concept:



consequences:

- fuel temperatures below 100 °C
- salt temperatures not changed
- no accidents from internal or external reasons which release non-allowable quantities of radioactivity to the environment (water ingress still requires some investigation)
- after ~ 10^5 years the radiotoxicity is in the same order as that of fresh fuel

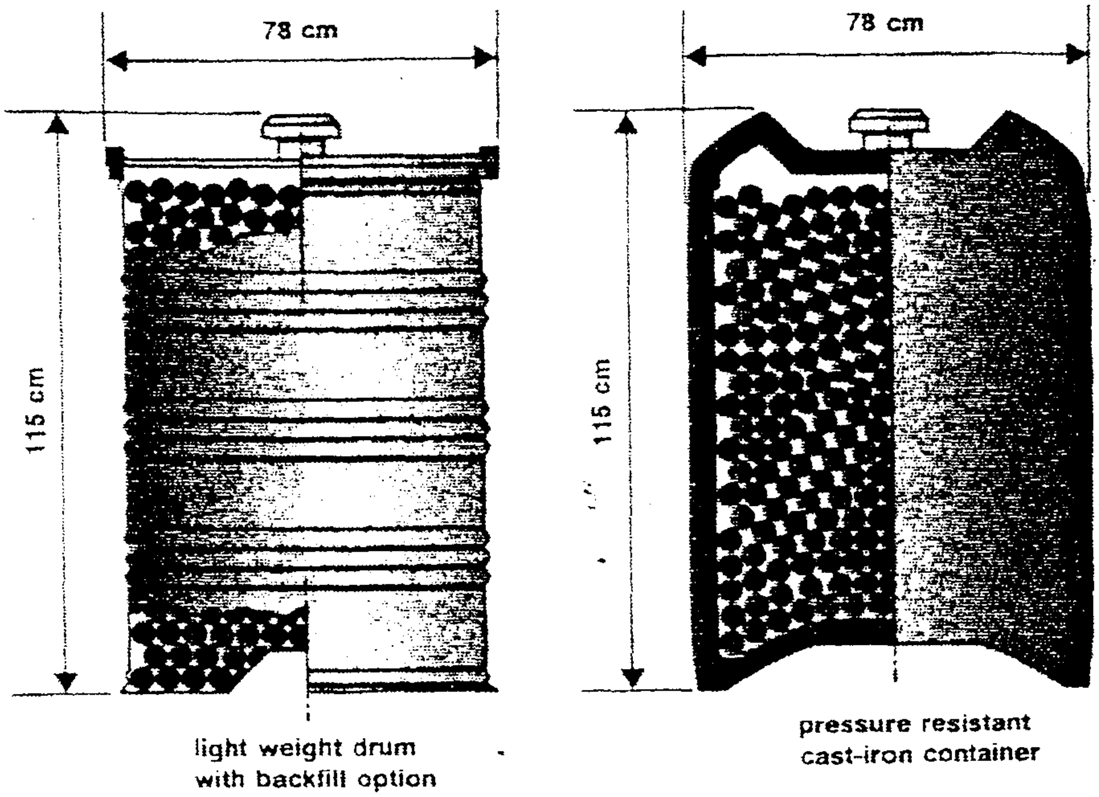
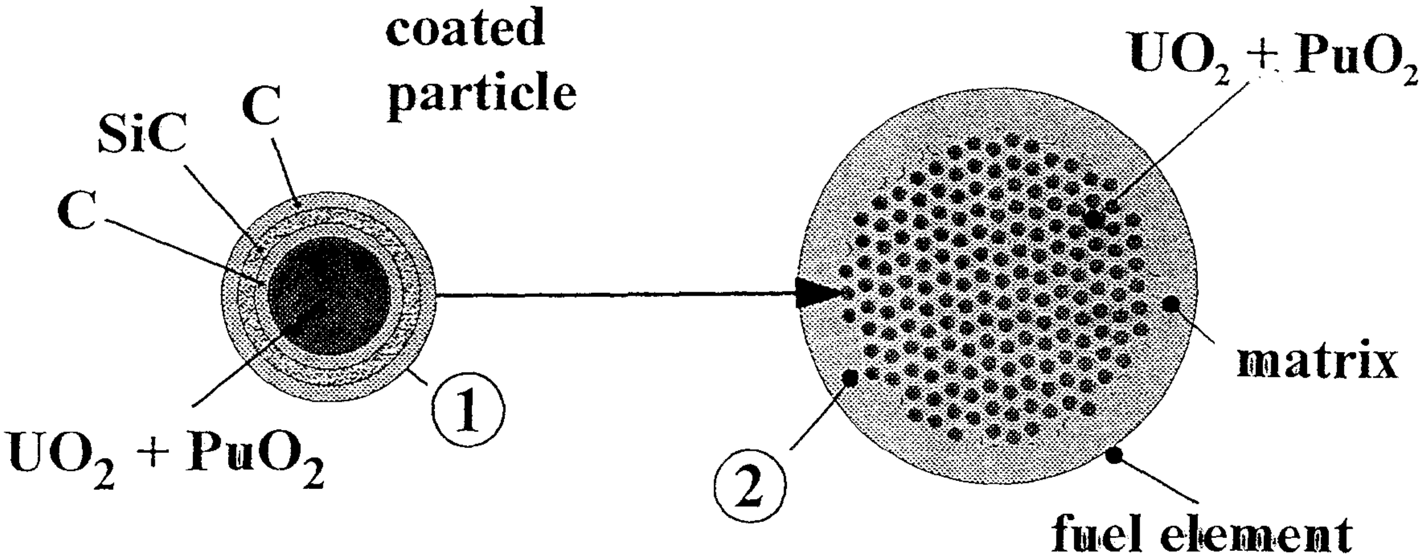


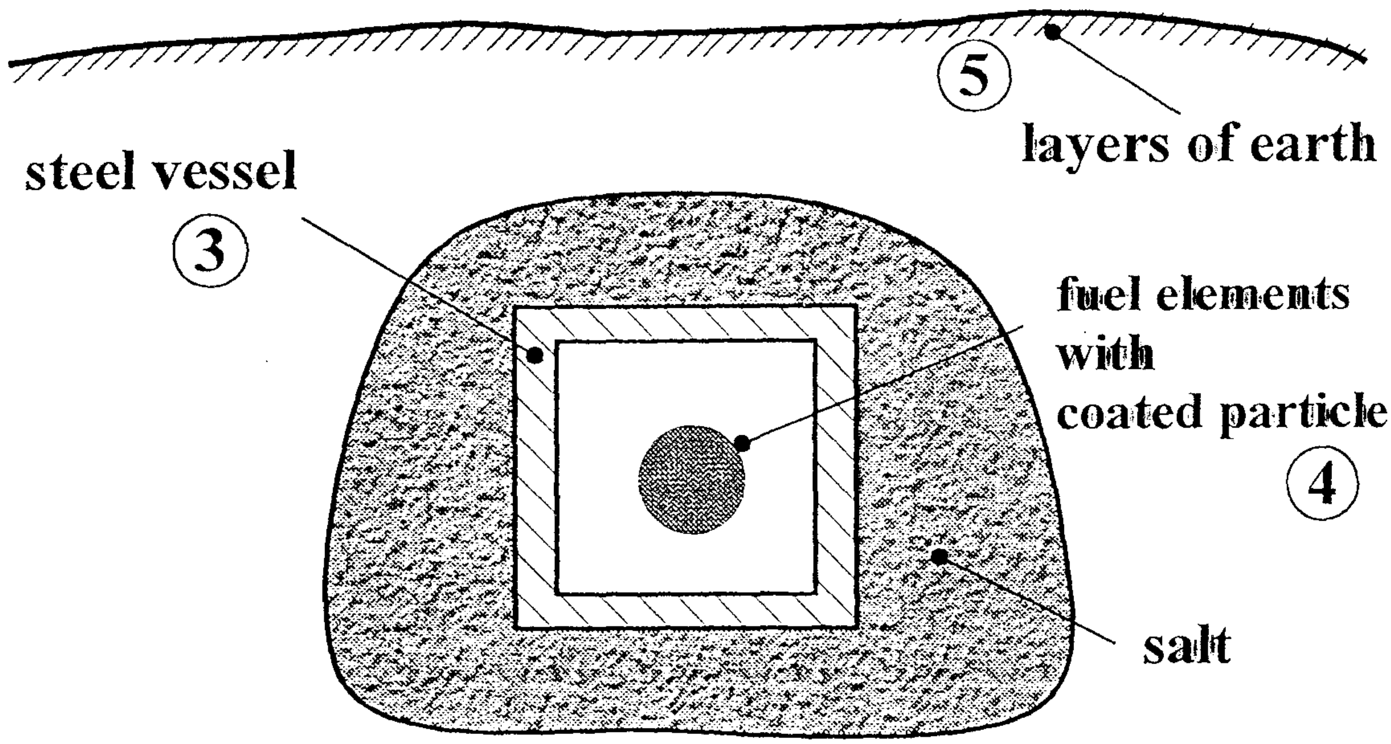
Abb. 4.2: 400 l-Faßsysteme zur Endlagerung abgebrannter HTR-Brennelemente
a) 400 Liter-Faß; b) Gußcontainer
b) druckfester Großcontainer

Volumen des Fasses	400 Liter
Zahl der Brennelemen-te/Faß	1.800
Faßhöhe	115 cm
Faßdurchmesser	78 cm
Faßgewicht	~ 500 kg
Wärme/Faß	< 50 W
Aktivität/Faß	$6 \cdot 10^{12}$ Bq

Tab. 4.1: Daten eines Abfallgebindes für abgebrannte HTR-Brennelemente (Beispiel: Stahlguß)

final storage of burned fuel elements





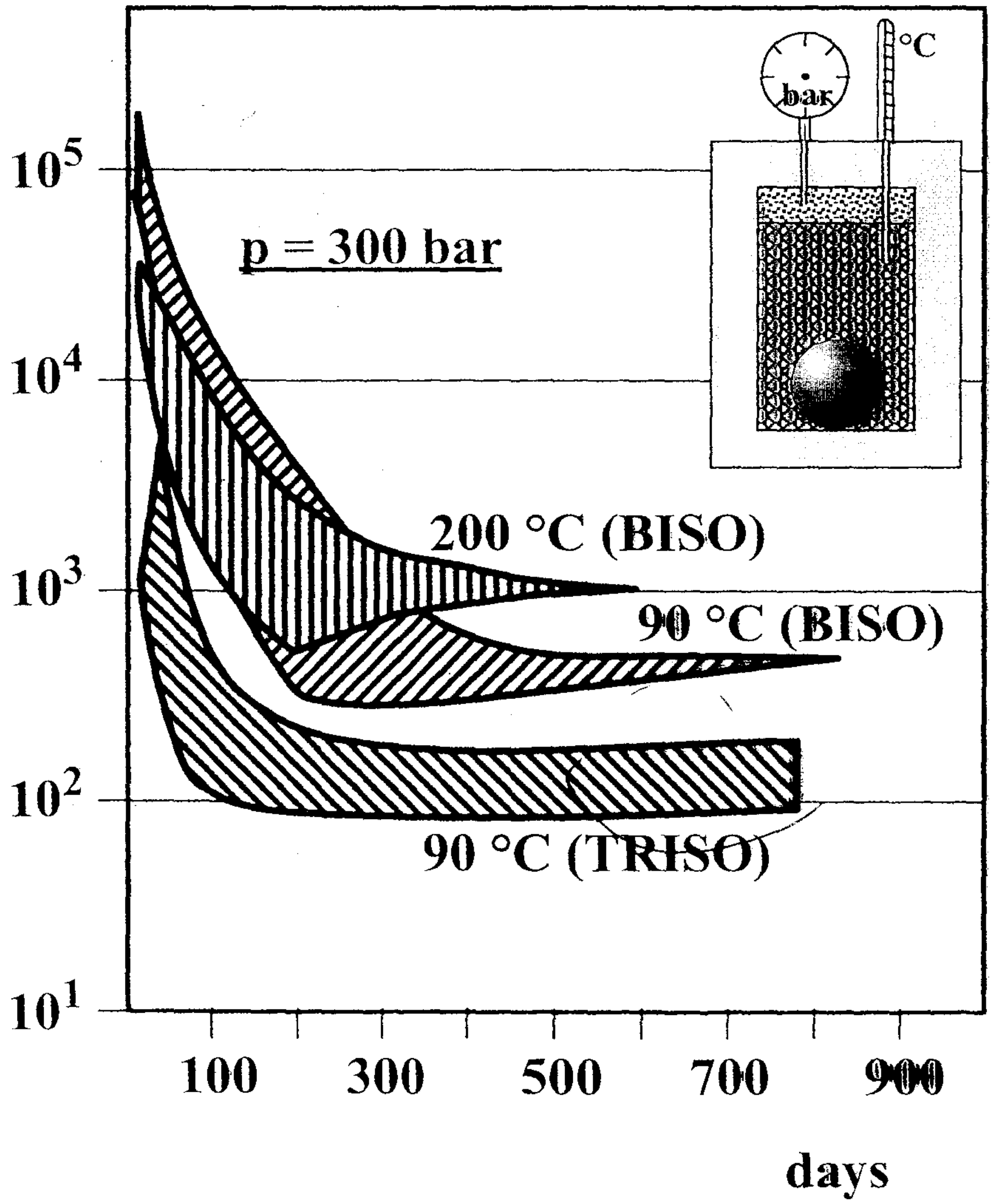
- release of radioactive material from the final storage can happen just after a very long time
- the amounts of radioactive material are relatively small
- only very small higher doses than the normal doses can be caused on the surface of earth even after extreme accidents inside the final storage system

Final storage of radioactive waste

Results of leaching experiments on HTR - fuel elements

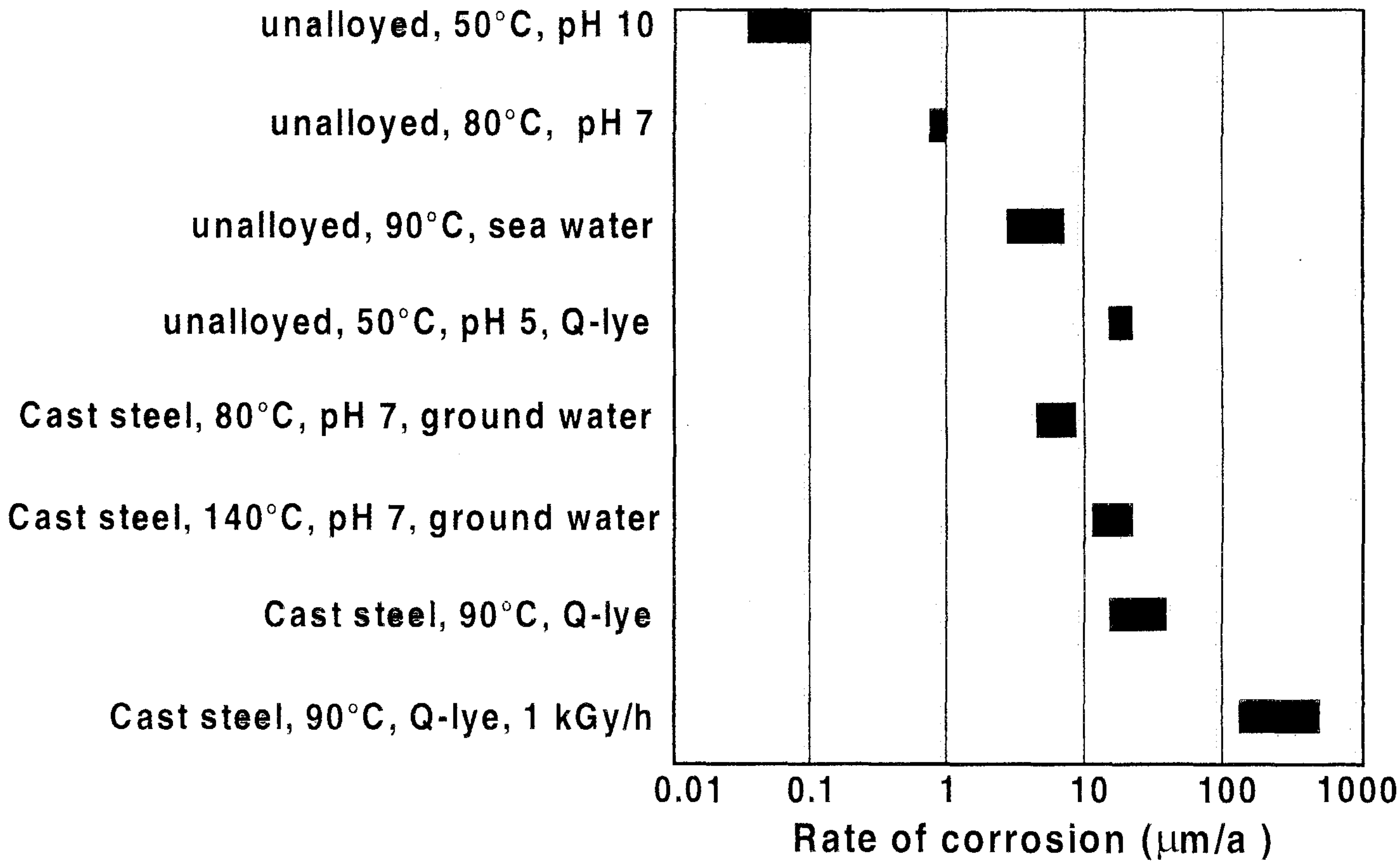
Relaeserate of Cs 137 to alkaline solution

Bq / days



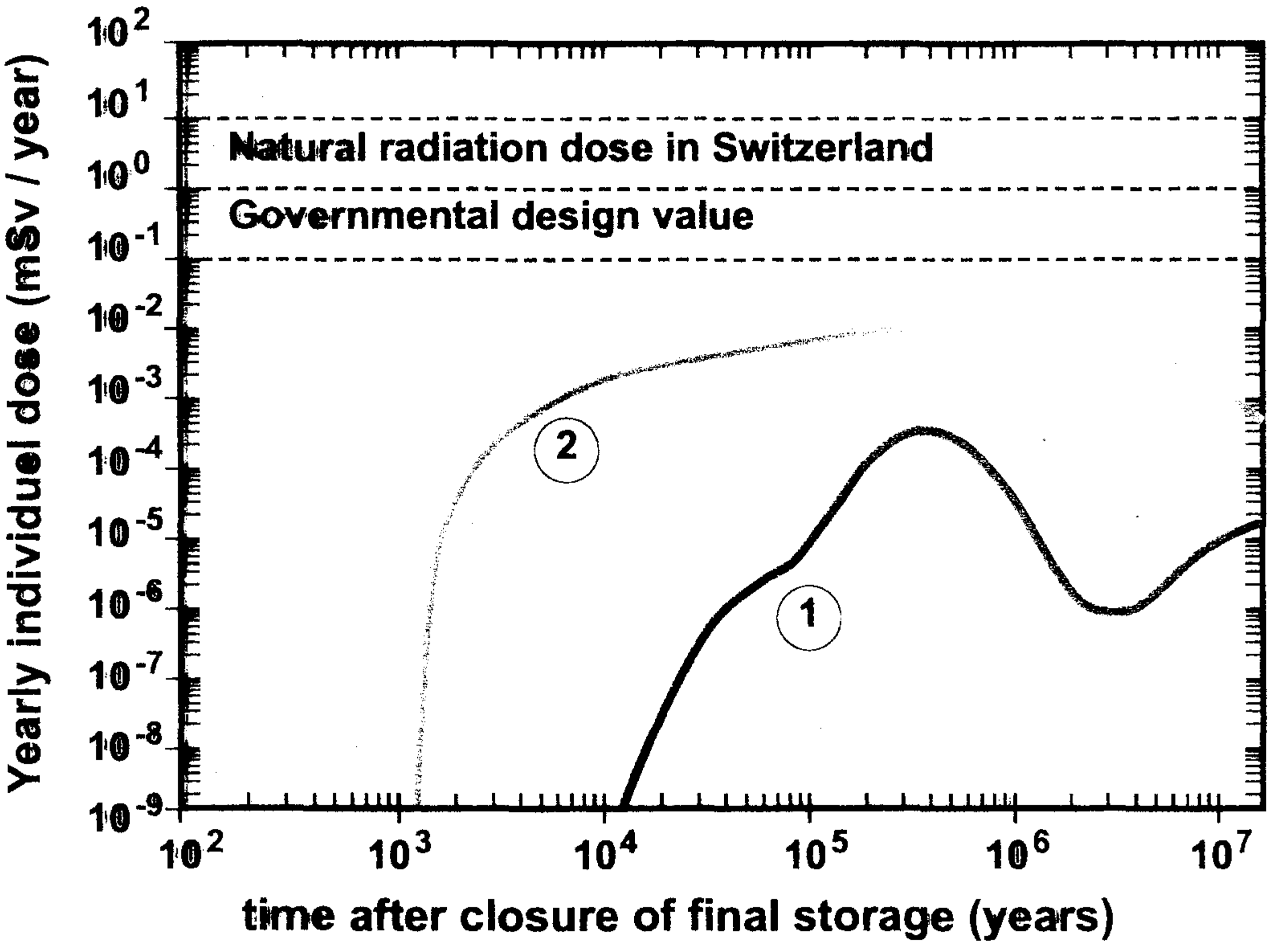
Final storage of spent HTR-fuel elements

corrosion rates of steel and cast steel in different media



Final storage of high level radioactive waste

Yearly individual doses from final storage of glass containers in granite

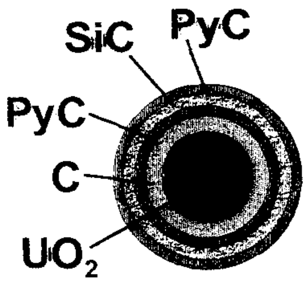


1 — Reference case (glas blocks storaged in granite)

2 — Accident assumption: water flow is higher by a factor of 100

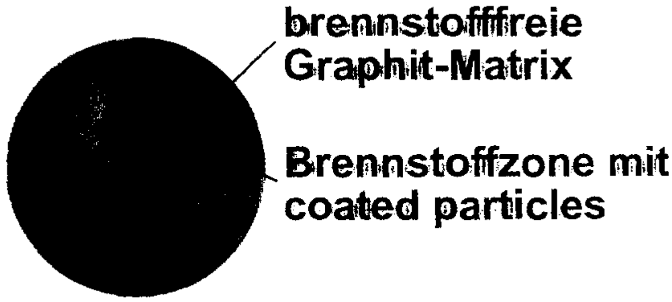
voneinander unabhängige Barrieren

1. coated particle

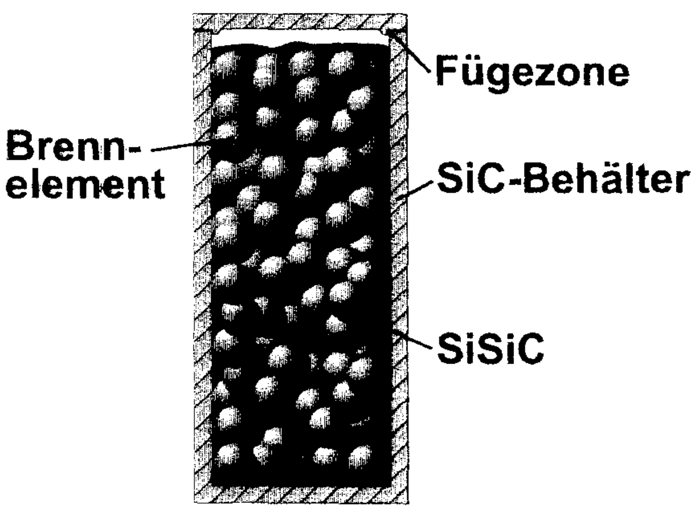


TRISO- oder BISO-coated particle

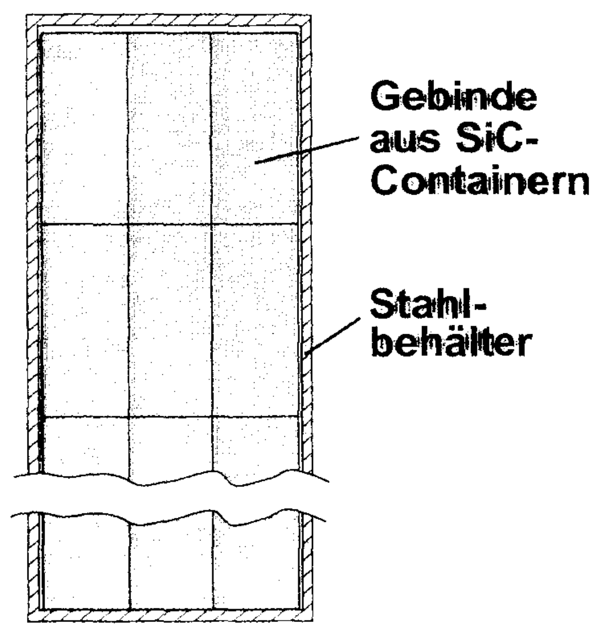
2. Brennelement



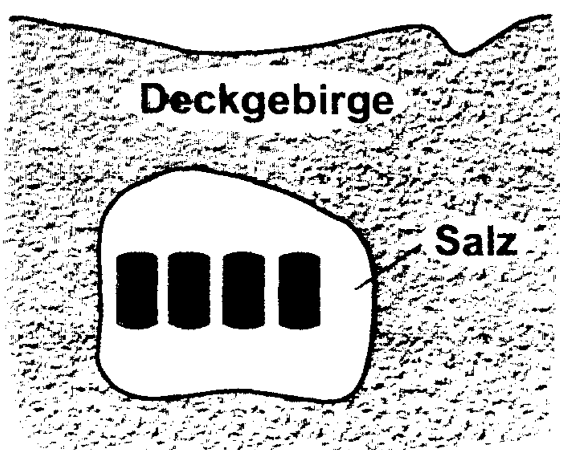
3. SiC-Container



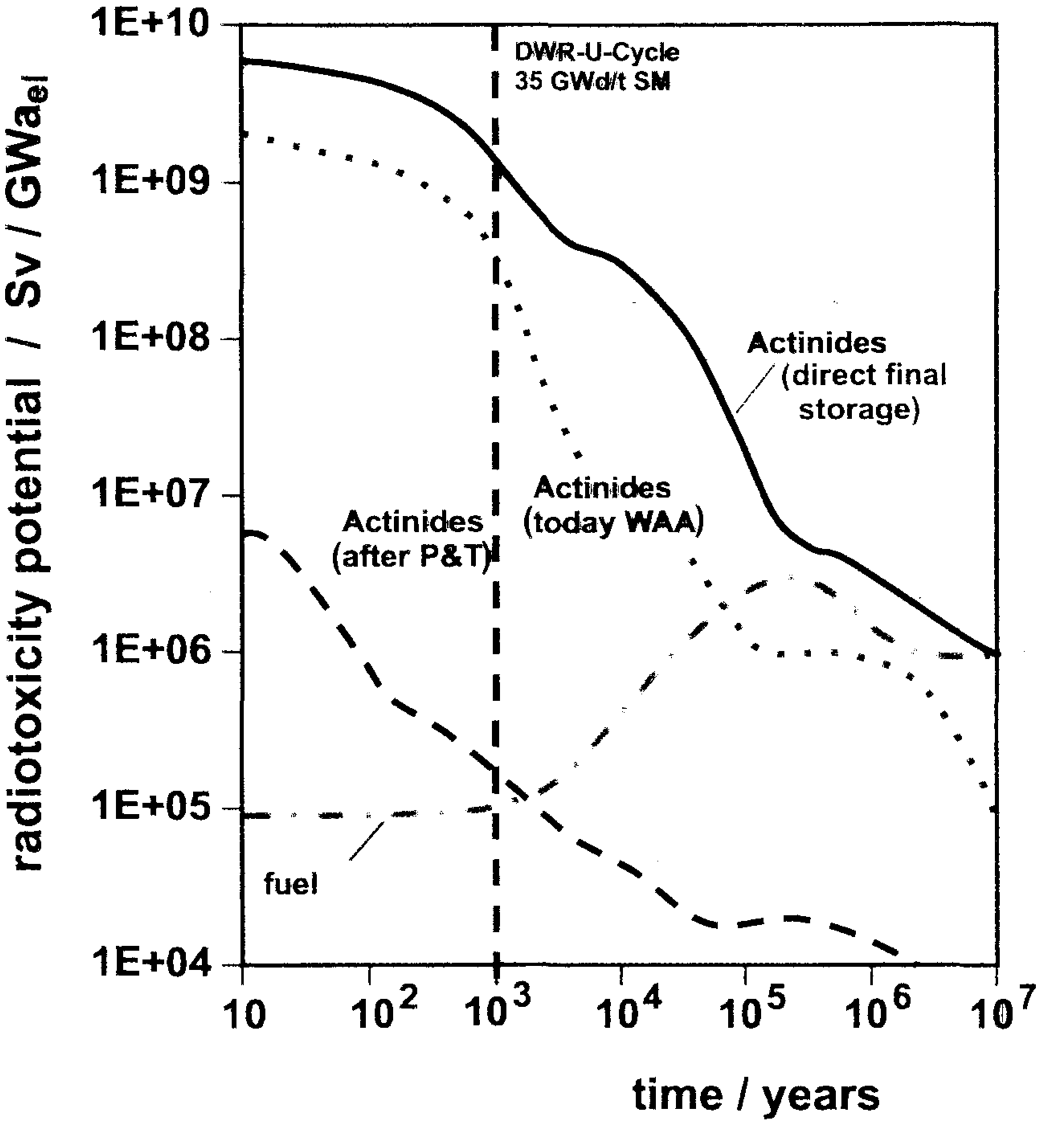
4. Stahlbehälter



5. Geologische Barriere



Radiotoxicity of waste: direct final storage of spent fuel elements compared to partitioning and transmutation

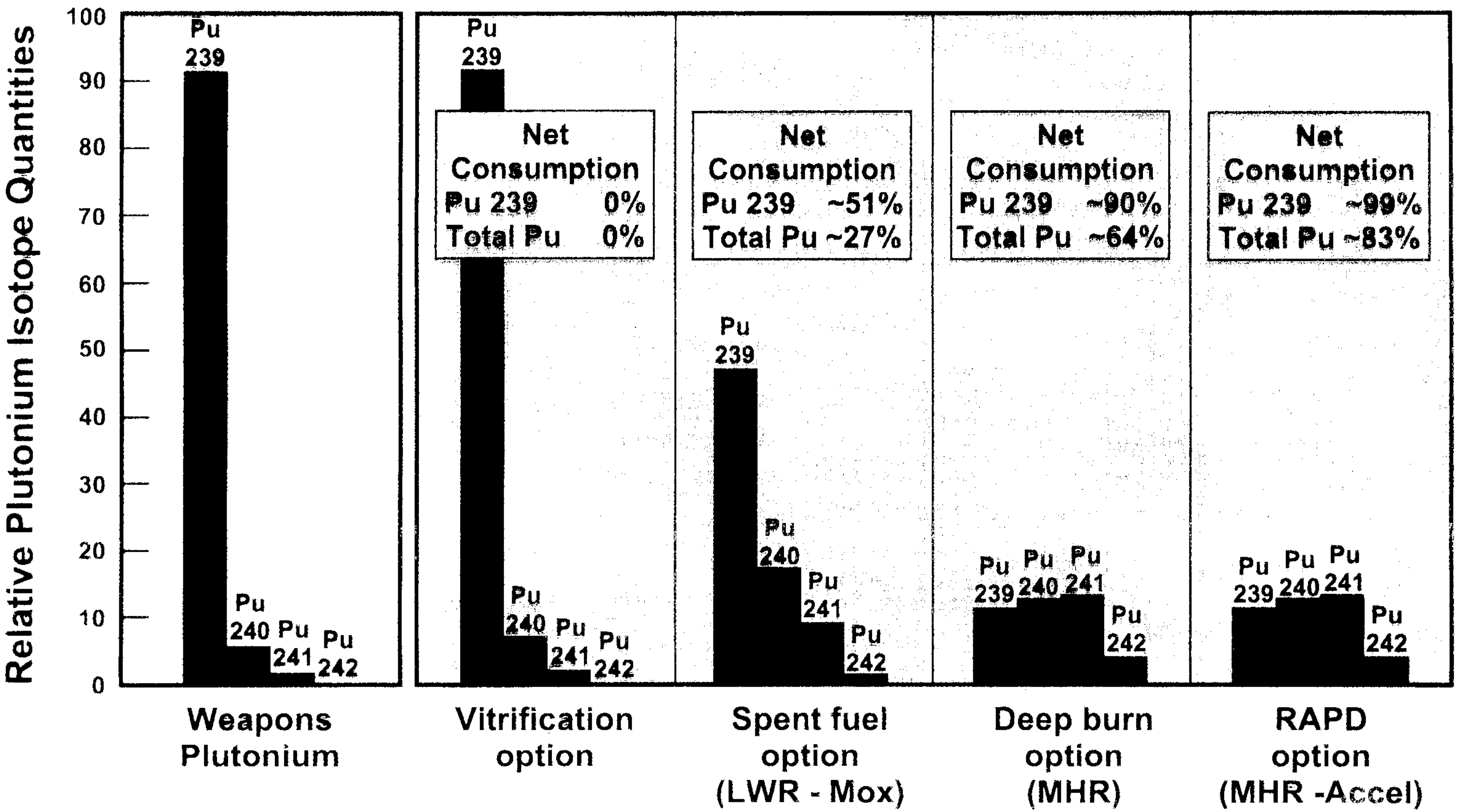


- direct final storage of burned fuel elements
  → proof of safety of storage for 10^6 years

- partitioning and transmutation
  → proof of safety of storage for 1000 years

Plutonium - ways of further handling

Comparison of disposition options to consume and degrade weapons plutonium (after General Atomics)



XA9848071

EXPERIENCE WITH THE INTERIM STORAGE OF SPENT HTR FUEL ELEMENTS AND A VIEW TO NECESSARY MEASURES FOR FINAL DISPOSAL

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Jülich, Germany

Abstract

In the Federal Republic of Germany the AVR pilot high-temperature reactor was operated successfully for more than 20 years and the THTR prototype high-temperature reactor for more than three years The reactors were shut down for decommissioning at the end of 1988 and the discharge of core inventories and packaging of the fuel, together with the temporarily stored fuel, for long-term interim storage in appropriate casks and facilities was started in 1992 and finished in 1995 for the THTR and began in 1994 for the AVR and will be completed at the beginning of 1998.
With a view to the long-term interim storage and final disposal of spent HTR fuel from both reactors many experiments have been carried out to characterize the spent fuel and to learn about its behaviour and during the operating period of the AVR reactor much experience has been gathered by remote handling, shipping and temporarily storing fuel packages at different appropriate facilities of the Forschungszentrum Jülich GmbH (FZJ). Furthermore, after starting the discharge of the AVR core more than 200 so-called AVR dry storage canisters (AVR-TLK), each containing 950 spent fuel elements have been reloaded from an AVR single shipping cask into CASTOR THTR / AVR shipping and storage casks in the hot cell facility, which is one part of the waste treatment and storage building of FZJ, and currently about 100 CASTOR casks, each containing in all 1900 fuel elements, have been prepared and stored in the AVR interim storage facility (AVR-BL), as another part of this building.

1. INTRODUCTION

In the Federal Republic of Germany the AVR pilot high-temperature reactor was operated successfully for more than 20 years and the THTR prototype high-temperature reactor for three years. During operation they were charged with several types of spherical graphite fuel elements, containing different U/Th mixtures such as coated HEU or LEU fuel particle dispersions. About 300,000 AVR and 620,000 THTR fuel elements were irradiated during the operating times. THTR spent fuel was temporarily stored on site and AVR spent fuel was temporarily stored at different hot cell and pool facilities of the Forschungszentrum Jülich GmbH (FZJ).

During the long operating period of the AVR reactor a lot of R&D work was carried out by FZJ to characterize the different types of spent fuel elements for developing interim storage and final disposal concepts /1, 2, 3/ and as part of this work much experience has been gathered by using spent fuel elements for experimental set-ups and by handling, shipping and temporarily storing fuel packages at different appropriate facilities of FZJ.

At the end of 1988 the reactors were shut down for decommissioning and discharge of core inventories and packaging of the fuel, together with the temporarily stored fuel for long-term interim storage in appropriate casks and facilties was started in 1992 and finished in 1995 for the THTR and began in 1994 for the AVR and will be completed at the beginning of 1998.

At the Ahaus facility, 305 casks, each loaded with canisters containing 2100 spent THTR fuel elements have been managed and stored by the Brennelement-Zwischenlager Ahaus GmbH. At the Jülich facility currently 100 CASTOR casks, each loaded with two AVR-TLK containing in all 1900 spent AVR fuel elements, have been prepared and stored by FZJ.

2. OVERVIEW OF SPENT AVR FUEL MANAGEMENT

By the end of 1988 about 190,000 spent fuel elements had been discharged during the reactor operating period and were packaged and shipped to FZJ. After granting the licences according to the Atomic Energy Act for discharging the core inventory for decommissioning by AVR GmbH and for handling and long-term interim storage by FZJ, work began in August 1993 on managing fuel from core discharging by means of so-called AVR cans (AVR-K), each containing 50 fuel elements, and from different FZJ facilities for fuel reloading from AVR-K into AVR-TLK as well as charging of CASTOR casks for interim storage in the AVR interim storage facility (AVR-BL).

At that time about 84,000 fuel elements packaged and sealed in AVR-K were stored in the water cooling facilities of the Hot Cell (HZ) and the Research Reactor (FR) Departments, about 106,000 HEU fuel elements enclosed in AVR-TLK were stored in the LZ storage cell of the AZ hot cell facilities which is one part of the waste treatment and storage building of the Decontamination Department (DE), and about 110,000 fuel elements were still in the AVR reactor core. The paths of the AVR fuel elements from the reactor core to the AVR-BL are shown in *Figure 1.*

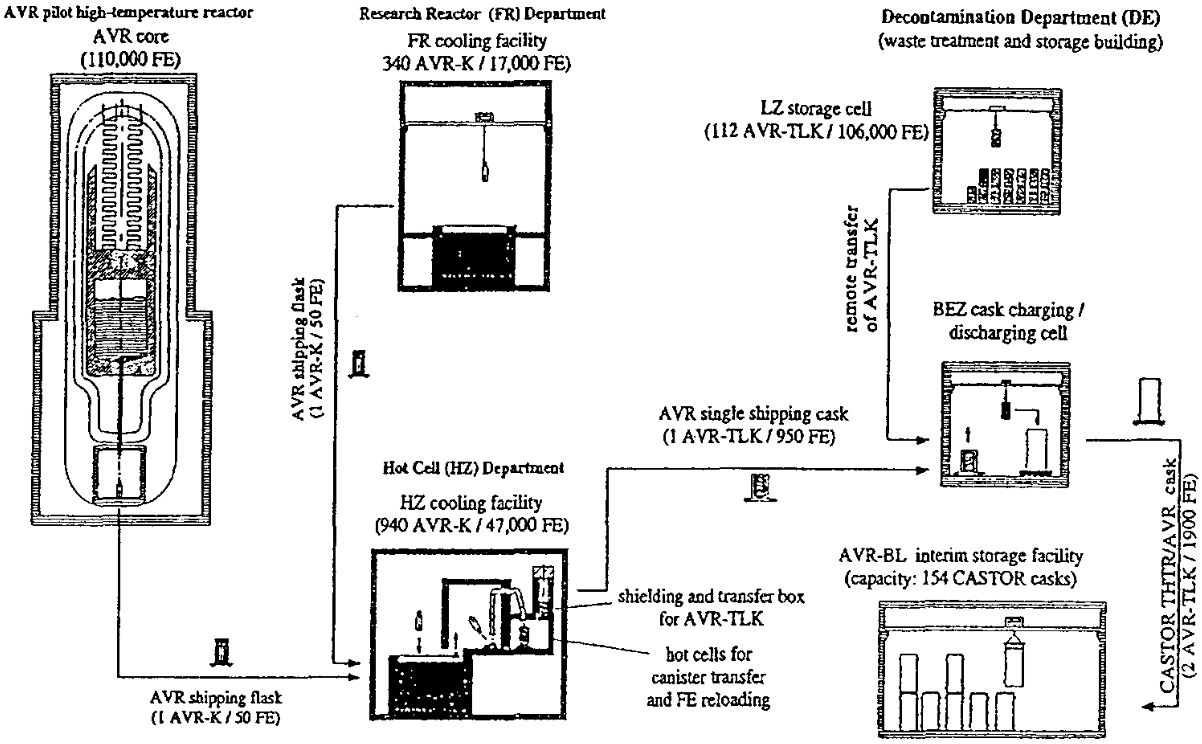

FIG. 1: Paths of the AVR fuel elements from the AVR reactor to the AVR interim store (fuel element inventories(FE) in the different facilities given by the end of 1993)

From August 1993 up to the present time about 91,600 fuel elements have been discharged from the reactor core, enclosed and shipped by means of AVR-K to HZ and reloaded there into AVR-TLK. About 65 AVR-TLK have been removed from the LZ and about 50,000 fuel elements enclosed in AVR-K have been removed from the above-mentioned water cooling facilities and reloaded into AVR-TLK so that in all 100 CASTOR THTR/AVR casks have been prepared and stored in the AVR-BL.

The delay in AVR core discharging in comparison to initial planing is caused by disturbances and failures of components from the different facilities and the equipment necessary for discharging the fuel and handling and reloading fuel packages and additionally, due to of problems, which arose at the beginning of 1995 with the liscensing procedures for LEU fuel handling and reloading in the HZ Hot Cell Department as well as handling LEU fuel packages in the DE Decontamination Department.

3. AZ HOT CELL FACILITY FOR HANDLING FUEL CANISTERS AND SHIPPING BAY FOR PREPARING AND ASSEMBLING CASTOR THTR / AVR CASKS

3.1 Preparing of CASTOR casks for charging

Preparing of CASTOR casks for charging is carried out in the shipping bay, which is part of the hot cell facility (AZ) and which covers the hot cells. Apart from a 50 Mg bridge crane for handling heavy loads, whose range of operation covers the whole shipping bay area, a 5 Mg crane is installed above the the so-called mounting area for handling CASTOR lids (*FIG. 2*).

Preparation of the sealing systems of casks, i.e. visual inspection, cleaning and if necessary, manual refurbishing of sealing groove surfaces of lids, sealing surfaces of casks as well as of the metallic gaskets, is carried out by means of the lid tilting device and the assembly station, which are installed in the mounting area. Positioning of casks onto the flat-bed cargo trailer, which is part of the assembly station, is carried out by means of the 50 Mg bridge crane.
After inspection and refurbishing work the metallic gaskets are fixed in the sealing grooves, the cask is closed with the primary lid and shipped into the BEZ cask charging / discharging cell.

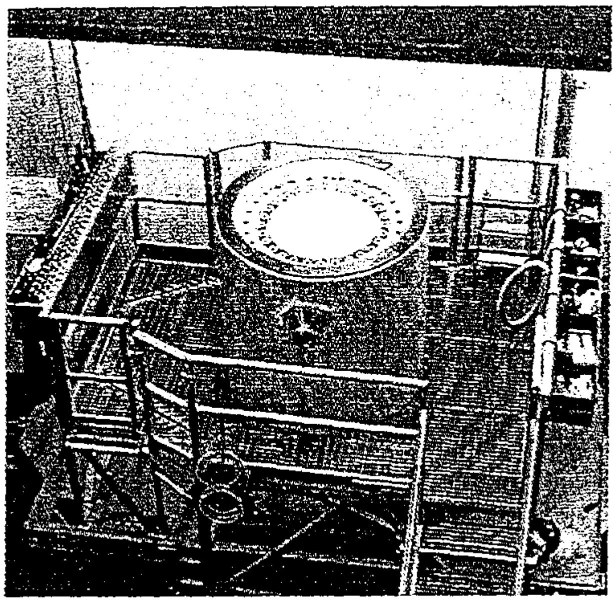

FIG. 2: View onto the assembly station with a CASTOR cask positioned on the flat-bed cargo trailer

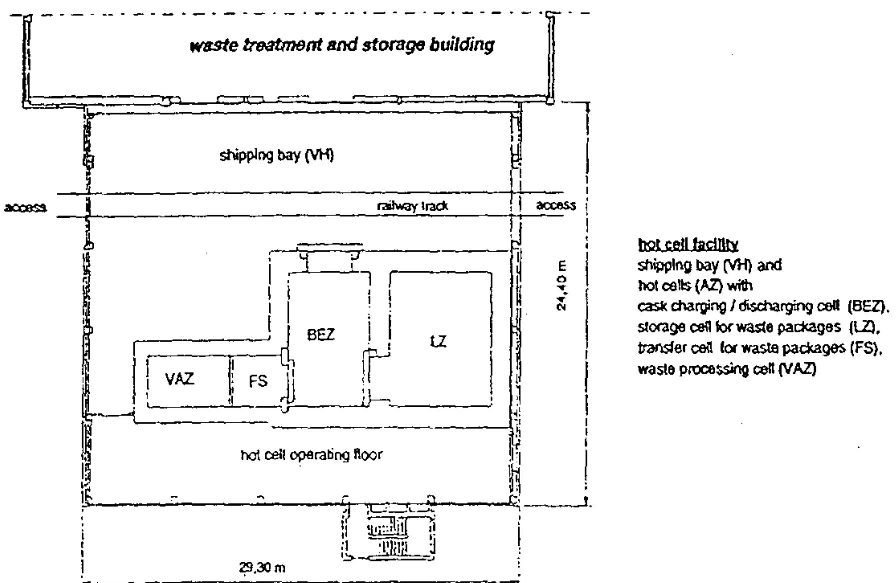

FIG. 3: Simplified ground plan of the AZ hot cell facility

3.2 Charging of CASTOR casks in the BEZ hot cell

Before remote charging of two AVR-TLK and closing of a CASTOR THTR/AVR shipping and storage cask, AVR-TLK must be shipped and discharged from the AVR single shipping cask or transferred from the LZ storage cell and lowered into lay-down positions in the BEZ cask charging / discharging cell of the AZ hot cell facility, which is accessible from the shipping hall by means of the so-called BEZ shielding gate (*FIG. 3*) . For remote handling of waste drums and heavy loads of up to 4 Mg a power manipulator with drum tongs and a hook is installed in the BEZ (*FIG. 4*).

For handling of AVR-TLK a special pintle grapple and for handling the CASTOR primary lid a coupling link can be attached to the hook. Furthermore, the manipulator is equipped with a laser positioning system for accurate lifting and lowering of the CASTOR primary lid.

During the entire handling, charging and closing procedure the CASTOR cask remains on the flat-bed cargo trailer, which is equipped with a removable scaffold framing the cask and enabling access to the top of the cask. After the primary lid is in place, the low radiation level allows opening of the BEZ shielding gate for radiation protection measures and for preliminary tightening of primary lid's screws and for shipping the cask back to the assembly station in the shipping bay

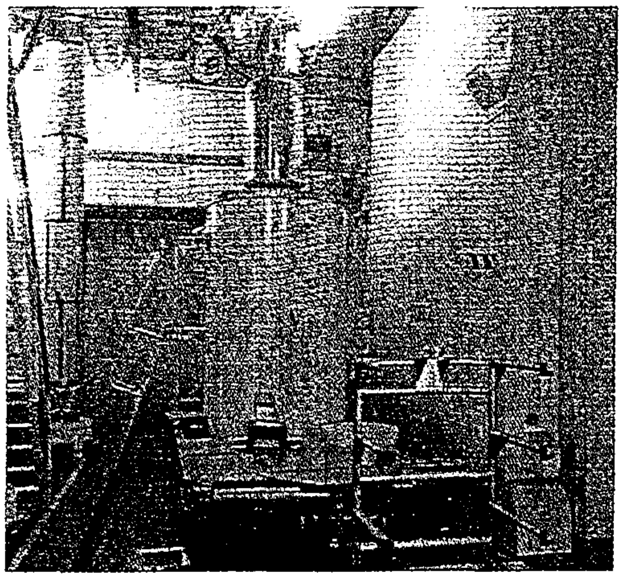

FIG. 4: Remote charging of a CASTOR cask in the BEZ cask charging / discharging cell

3.3 Assembling and leak testing procedures

After the CASTOR cask has been transferred back to the assembly station the assembling and leak testing procedures are as follows:

- tightening of the primary lid's screws
- evacuation of the cask and replacement of the withdrawn gas by an Ar /He-gas mixture
- He leak testing of the primary sealing system
- inserting the secondary lid
- tightening of the secondary lid's screws
- evacuation of the space between the lids and pressurizing the space with He gas
- He leak testing of the secondary sealing system
- mounting and He leak testing of a pressure gauge
- covering the top with the protective lid
- mounting the VACOSS seal on the protective lid
- removal of the scaffold and shipment to the AVR-BL interim storage facility

4. THE AVR-BL INTERIM STORAGE HALL

The AVR-BL interim storage facility has been built and licensed according to the Atomic Energy Act (§6 AtG) for the interim storage of spent AVR fuel elements which have been irradiated during the operating period of the AVR pilot reactor and which have to be enclosed in CASTOR THTR/AVR shipping and storage casks. Lay-out of the storage area will serve for the interim storage of 154 casks, which are stacked alternately on one and two levels (FIG. 5 and 6)

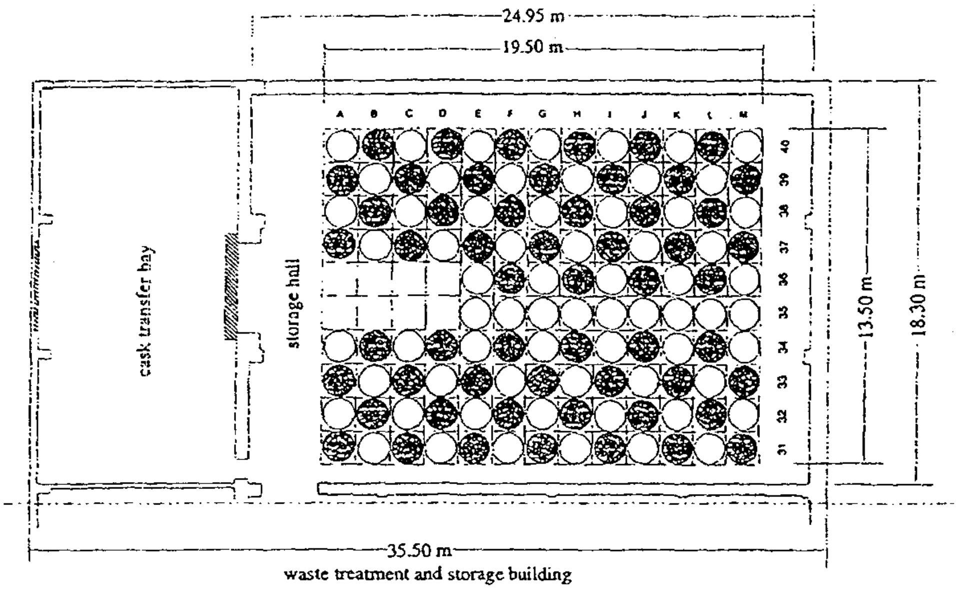

FIG .5: Scheme of the arrangement of CASTOR casks in the AVR-BL interim storage facility as another part of the waste treatment and storage building

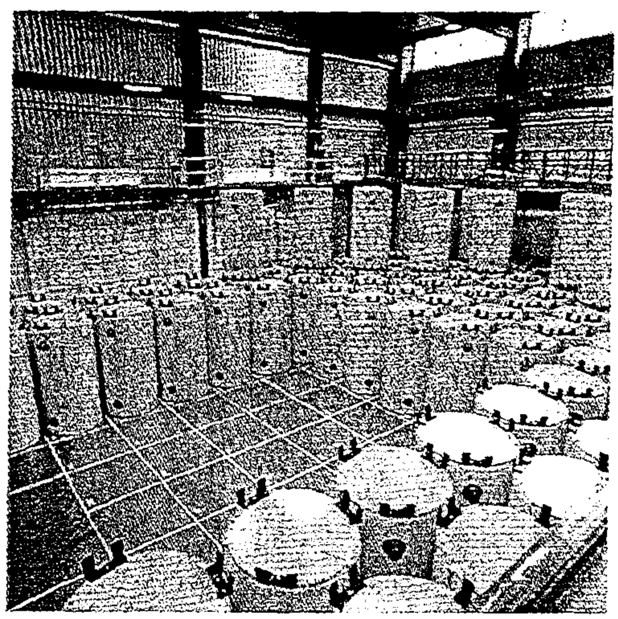

FIG. 6: View into the AVR-BL storage hall

5. SAFETY CONCEPT FOR INTERIM STORAGE

The safety concept for the interim storage of AVR spent fuel elements is based, in particular, on design requirements for CASTOR THTR/AVR shipping and storage casks as a tight enclosure so that any undue release of radionuclides is excluded both in normal operation and under conceivable accident conditions. According to design, the sealing function of both lid sealing systems is monitored during storage so that any deterioration or failure of a sealing barrier is detected and repair measures for restoring the two-barrier system can be carried out in the AZ hot cell facilities.

Within the Atomic Energy Act licensing procedures (§6 AtG) for the AVR and the BZA interim storage facility a leakage rate of $L \leq 10^{-7}$ mbar × l / s for each lid sealing sytem of the CASTOR casks is required.

Apart from the results obtained on the basis of long-term experiments with comparable lid sealing set-ups by FZJ /4, 5/ and other institutions /6, 7, 8/, which confirm the good long-term behaviour of such sealing systems with respect to the design requirements, confirmation of the required and specified tight enclosure of the spent fuel canisters has also been provided by the experience gathered from loading and preparing more than 400 CASTOR THTR/AVR transport and storage casks at the Jülich and Ahaus sites.

6. VIEW TO NECESSARY MEASURES FOR FINAL DISPOSAL

According to the plans of the Bundesamt für Strahlenschutz (BfS), solid and solidified radioactive waste forms, but in particular those with marked decay heat generation shall be disposed of in a final repository in a salt dome formation /9/. Heat-generating waste includes spent HTR fuel elements which are not to be reprocessed.

By the end of 1992 R&D work in establishing a final disposal concept for HTR fuel was focused on small 400-l fuel packages to be emplaced in 300-m deep boreholes in the final repository still to be constructed and then ultimately confined. Most of the work was discontinued at the end of 1992 /10, 11,12/.

The BfS subsequently gave preference to a final disposal concept for HTR fuel oriented along the lines of the direct disposal concept for irradiated LWR fuel elements, which is based on packaging the fuel in so-called POLLUX shipping and final disposal casks /13/.

Due to the fact that the design features, radioactive inventories and long-term behaviour of LWR fuel is completely different from those of HTR fuel further studies on the suitability of the CASTOR THTR/AVR cask to establish an adequate final disposal concept for HTR fuel have been currently initiated and will be carried out by Forschungszentrum Jülich GmbH in cooperation with Gesellschaft für Nuklear-Behälter mbH (GNB).

7. CONCLUSIONS

At the present time handling, reloading and packaging of fuel elements from AVR reactor core discharge, in addition to handling and reloading of some thousand fuel cans (AVR-K) and some hundred fuel canisters (AVR-TLK) in different facilities of FZJ as preparatory steps for charging CASTOR casks, has been conducted in a safe manner according to the requirements of the Radiation Protection Ordinance (StrlSchV).

Extensive cold testing of the equipment, training of the personnel responsible for charging and gas-tight closure of the CASTOR casks before starting hot operation and feedback of experience in addition the experience gathered by charging 100 CASTOR casks, has led to safe routine handling without the occurence of non-normal events.

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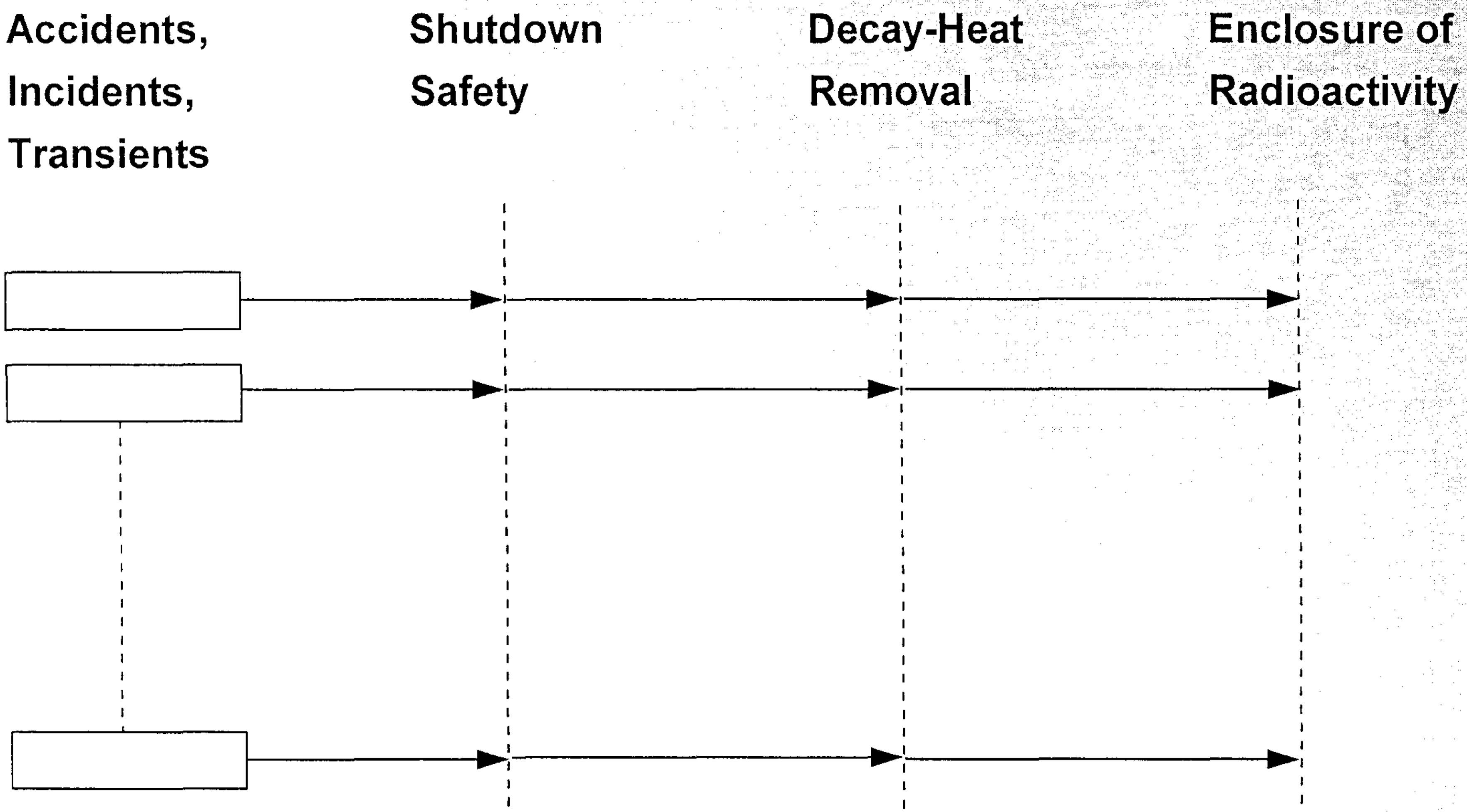
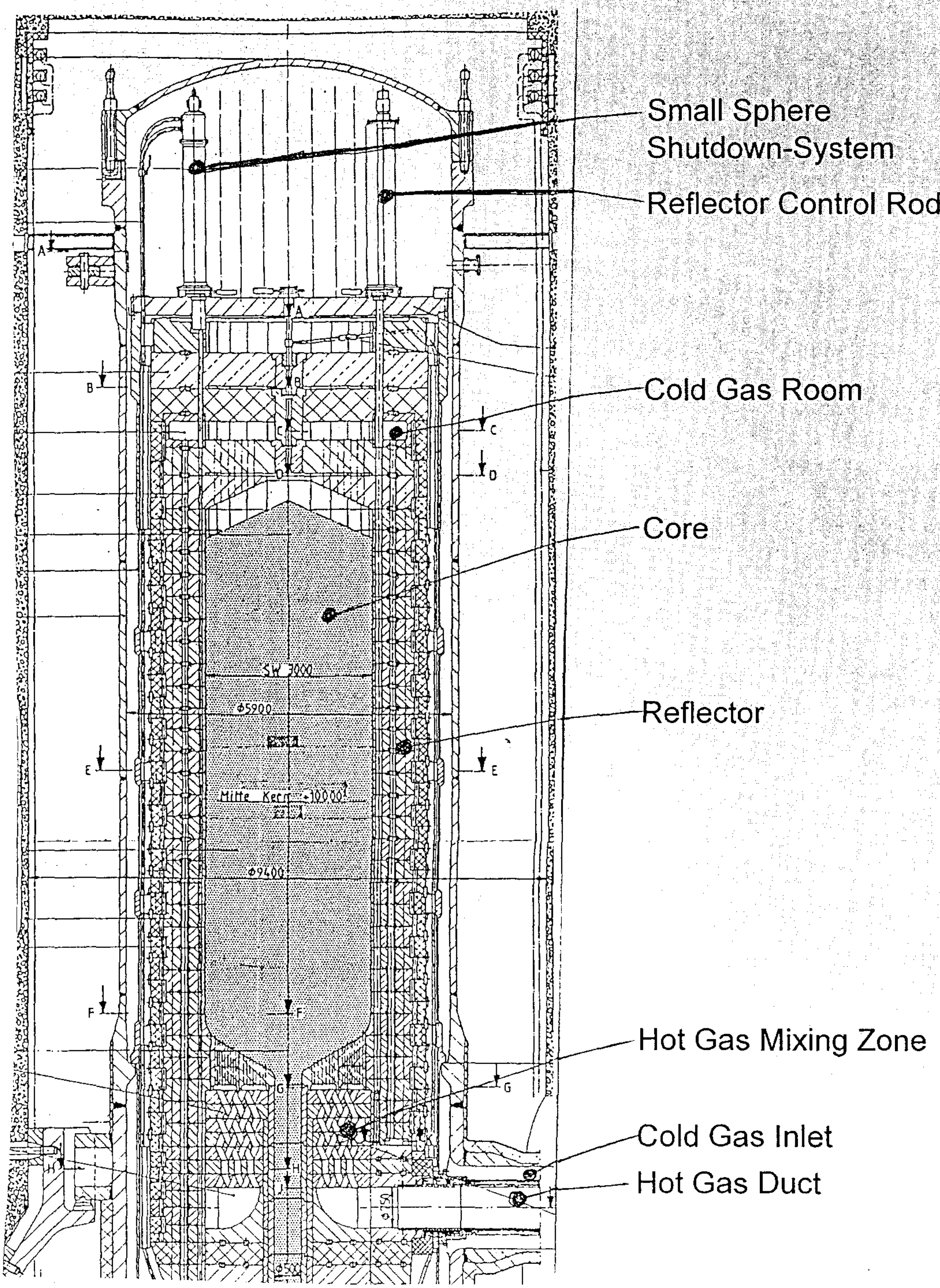
Examples of Safety Assessment

REACTOR PHYSICS

- Shutdown Margins
- Reactivity Feedback,
  Inherent Safety
- Power Density Distribution
  (Design and Control)

Design of Plant, Core and Fuel





Modular HTR-2 NPP, Reactor Physics

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Division Energy and Systems Technology

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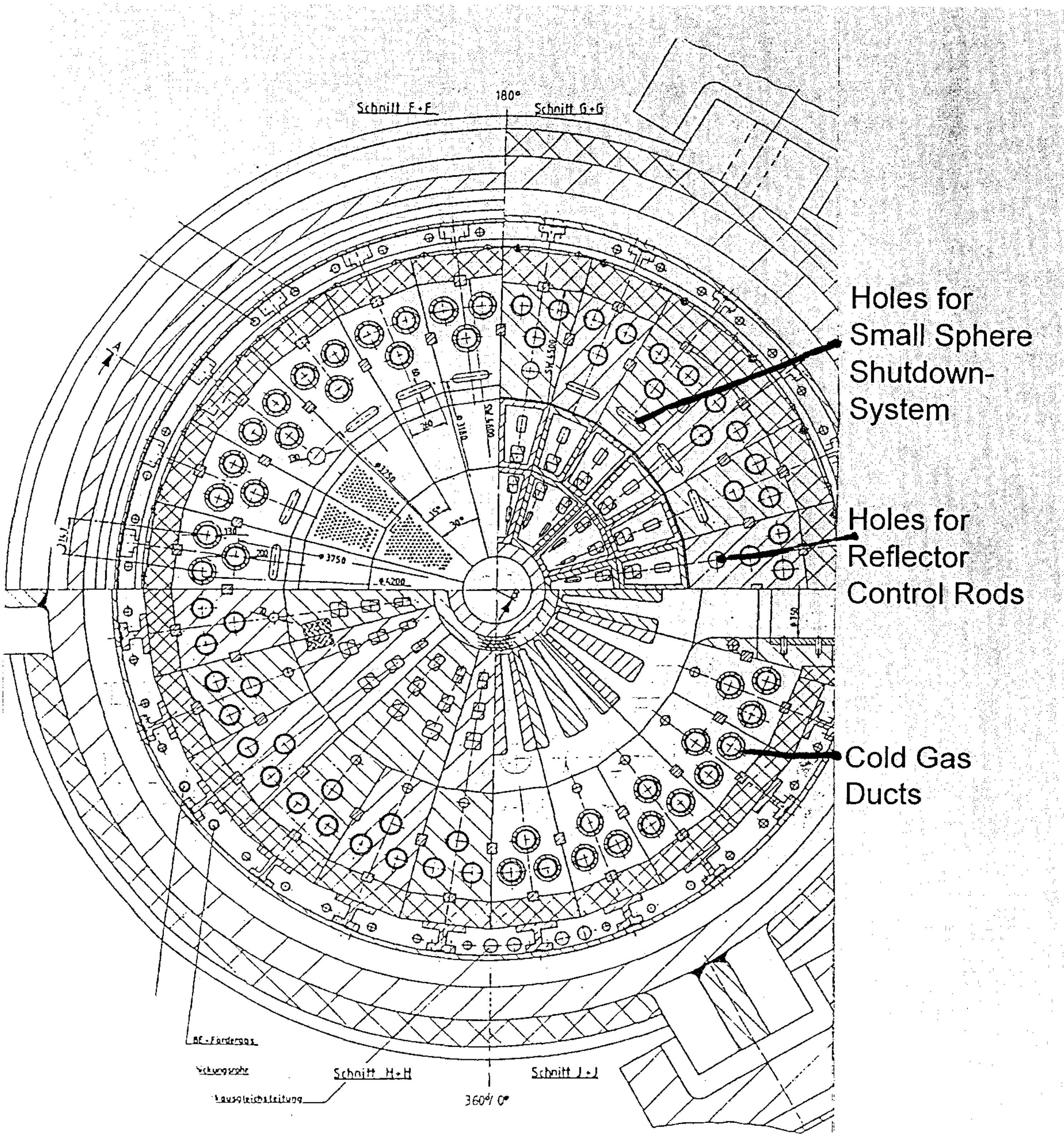

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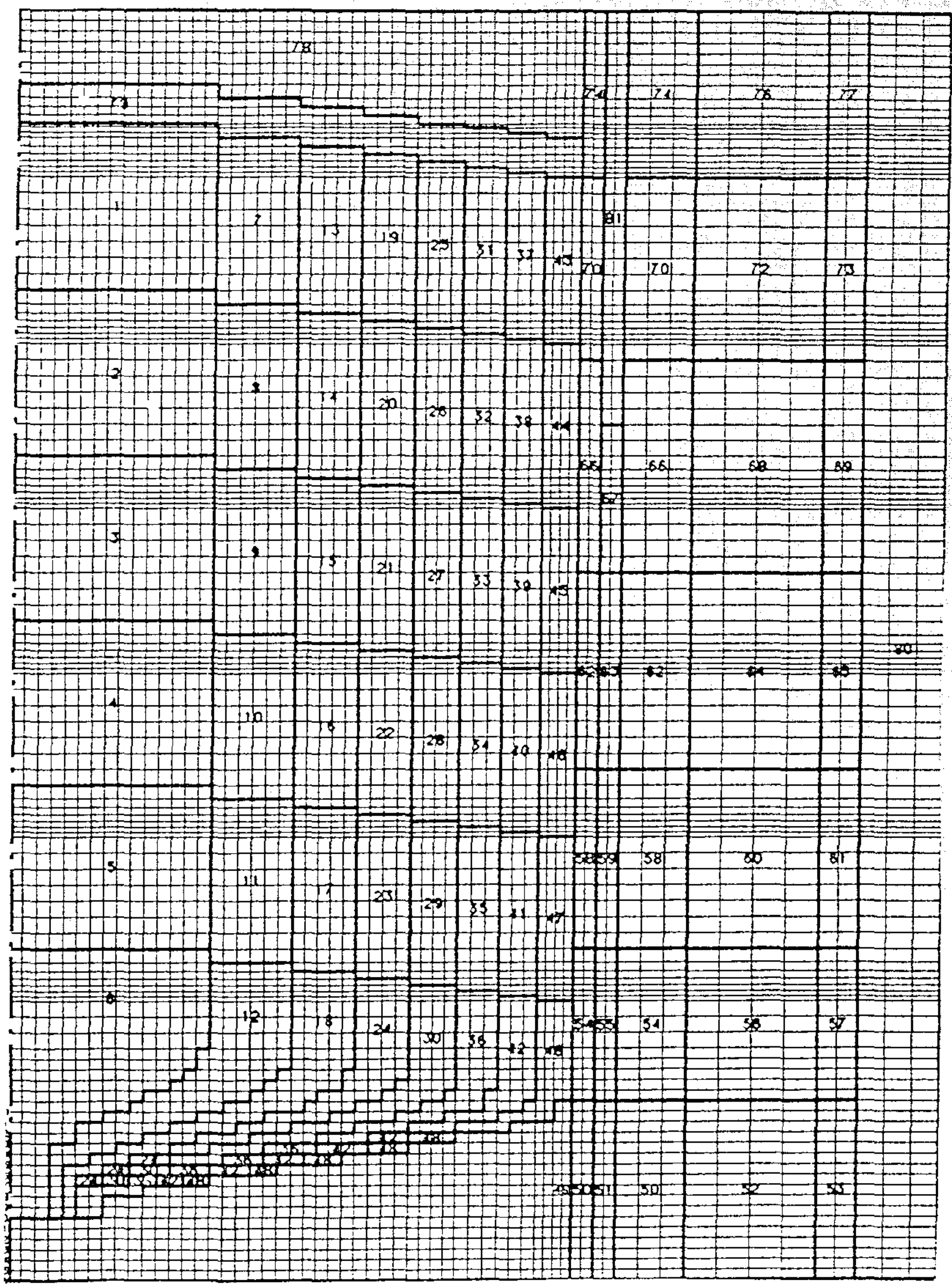
Scheme of Reactor Physics Calculation V.S.O.P (Very Superior Old Programs)

Name of the Program	Application	Result
GAM/THERMOS ZUT-DGL	Spectral-Calculation	Cross-Section of Core- and Reflector Regions
CITATION	Diffusion-Calculation	Multiplication-Factor, Power Density Map
FEVER	Burnup-Calculation	Nuclide-Densities
THERMIX	Temperature-Calculation	Temperatures of: - Fuel (UO_2) - Core-Graphite - Reflector-Graphite in all Regions

Model in R/Z-Geometry



For Example:

48 Core-Zones

(Burnup-Zones)

Calculation of Shut-Down-Margins

Rules

BMI-Kriterium 5.3

RSK-LL, 3.1.2, 22.1.1

KTA 3104

KTA 3103

KTA 3101.2

IAEA Safety Standards

IAEA Safety Guides

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Filter for HTR-Design

Additional Aspects of HTR-Design

Definition of design-basis-accident, incident and normal operation conditions

Calculation of Shut-Down-Margins
Results

Starting Core (BOC) Reactivity Balance Hot Shut Down

Control Reserve 100-90-100 %	+ 0,2 %
Reactivity Compensation for Partial Power (Xenon, Temperature)	+ 0,6 %
Accidental Reactivity Increase	+ 0,5 %
Partial Power - Zero Power	+ 0,1 %
Efficiency of 5 of 6 Rods	- 3,4 %
	- 2,0 %

Steady State Core Reactivity Balance Hot Shot Down

Control Reserve 100-50-100 %	+ 1,2 %
Reactivity Compensation for Partial Power (Xenon, Temperature)	+ 0,4 %
Accidental Reactiviy Increase	+ 0,5 %
Partial Power - Zero Power	+ 0,1 %
Efficiency of 5 of 6 Rods	- 2,6 %
	- 0,4 %

(Pictures core and profile)

Filter for and Additional Aspects of HTR-Design (Shutdown Safety)

Event, Effect	PWR	HTR	
Rod ejection	yes	no	
Uncontrolled rod withdraw	yes	yes	
Boron dilution	yes	no	
Water ingress	yes	(no)	600 kg
Uncontrolled fuel addition	no	yes	
Collapsing of holes in the core	no	yes	
Loss of Collant Accident	yes	yes	
Loss of Flow Event	yes	(no)	
Rod drop with reactivity gain	no	yes	
Core densification	(no)	yes	
Boron-loss by earthquake	yes	no	
Burn out of poison in reflector structural materials	no	yes	
Uncontrolled cooling of the core	yes	yes	
Temperature increase of the reflector	no	(yes)	
Error in excess reactivity	(no)	yes	
ATWS	yes	yes	
Changes of reflector geometry	no	yes	
Burn out of reacitivity control devices	yes	yes	
....	...	...	

Inherent Safety

Rules

BMI-Kriterium 3.2

RSK LL, 3.2

DIN 25405

KTA 3101.2

IAEA Safety Standards

IAEA Safety Guides

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HTR-Design-Filter and Additional Aspects of HTR-Design

- Negative Temperatur-Coefficient
- Negative Power-Coefficient
- Accidental Reactivity Increase limited so that the reactor can be shut down and temperature of the fuel elements remaining below a specified value

Temperature Coefficient of the fuel: Negative

Temperature Coefficient of the fuel-graphite: Slightly negative, a small positive value at Xenon-maximum

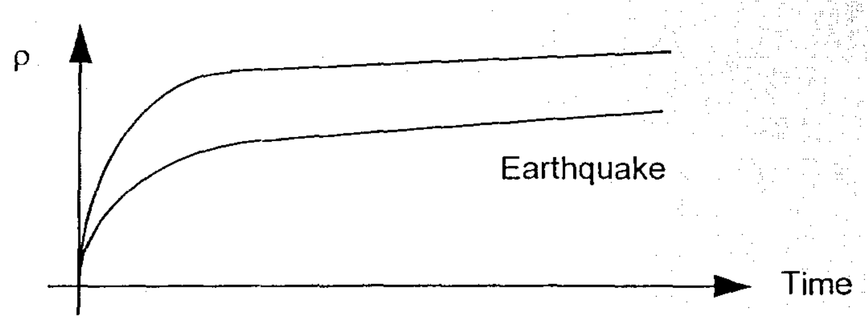
Temperature Coefficient of the reflector: Slightly positive

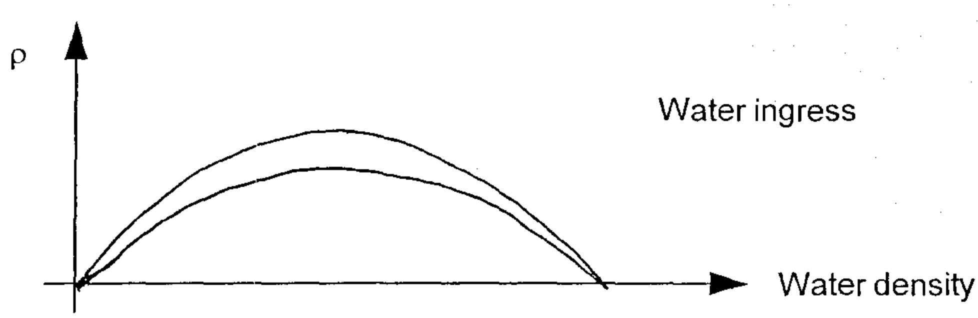
⇒ Inherent Safety confirmed

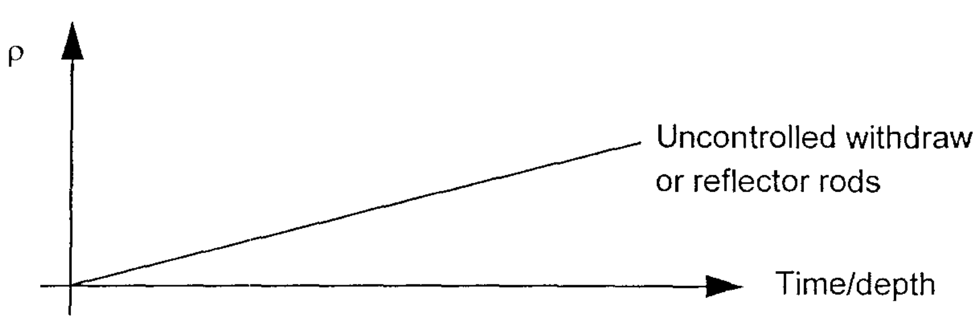
Inherent Safety

Search for possibilities of accidental increase of reactivity

- Core densification by earthquake
- Water ingress into the core
- Uncontrolled withdraw of reflector rods



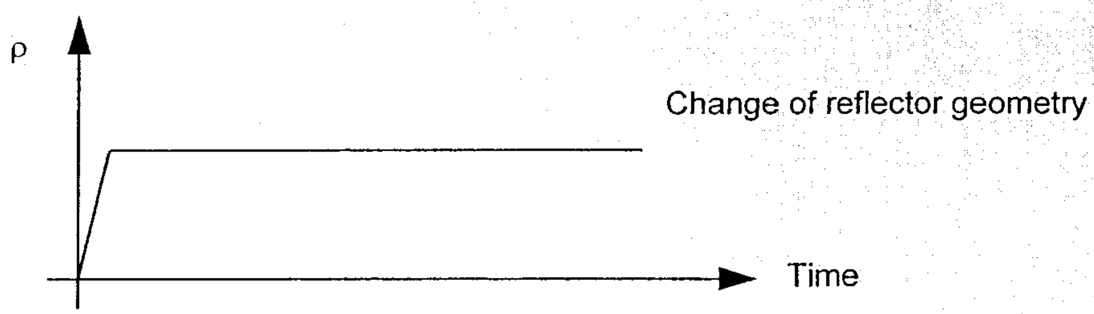


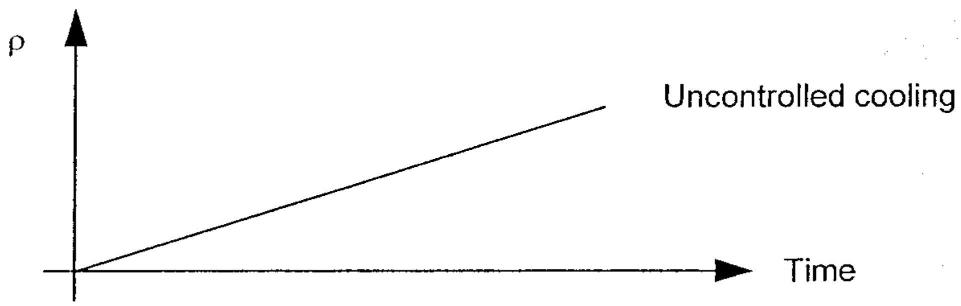


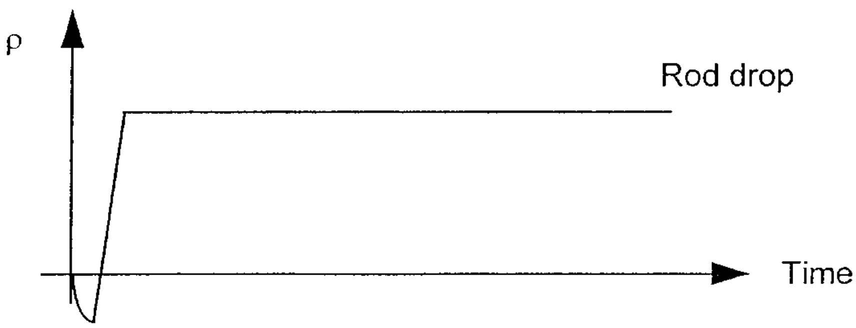
Inherent Safety

Search for possibilities of accidental increase of reactivity

- Change of reflector geometrie
- Uncontrolled cooling of the core
- Rod drop



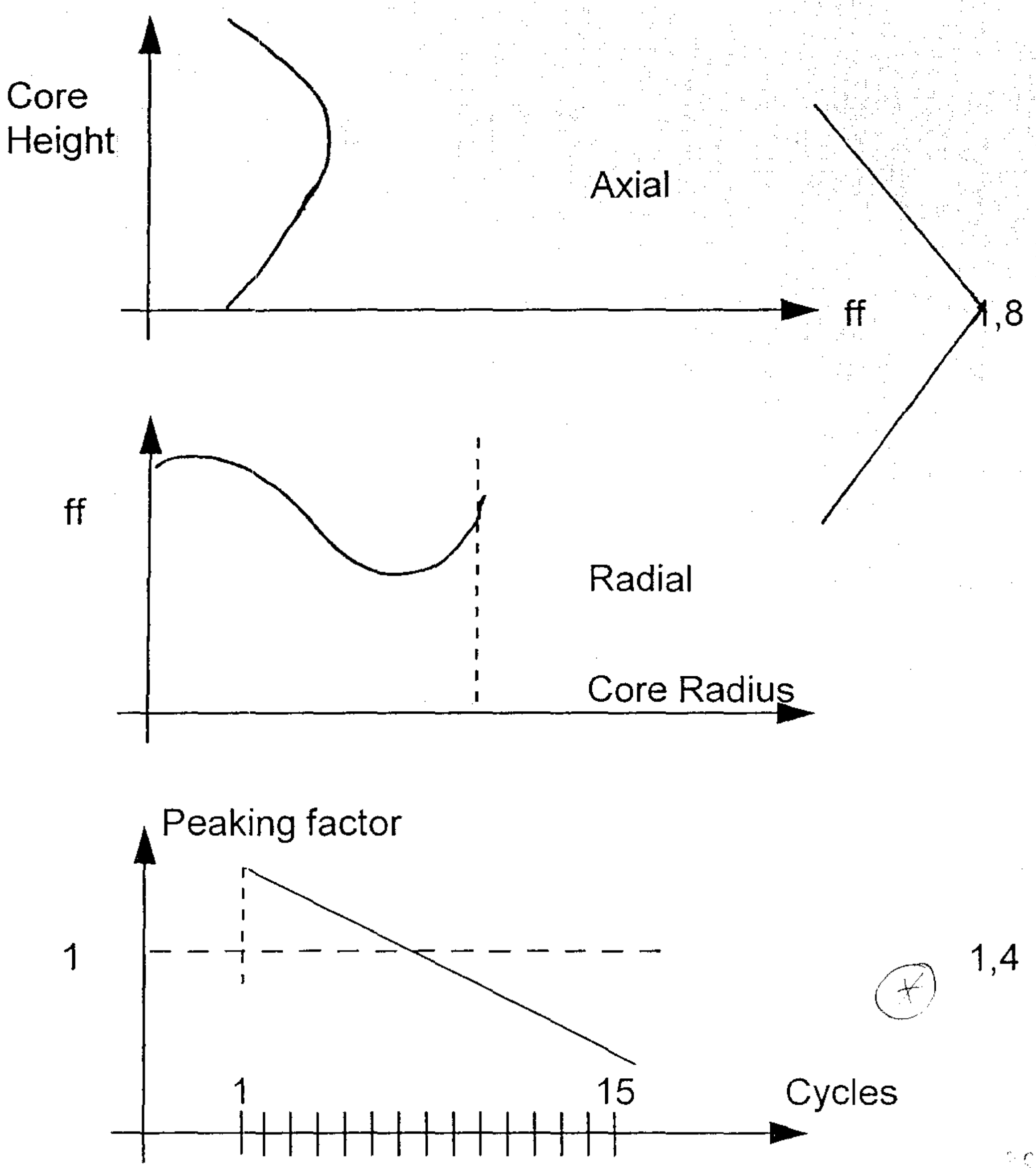




Power Density Distribution

Characterized by three formfactors:

- Axial formfactor
- Radial formfactor
- Peaking factor for fresh fuel element



Modular HTR-2 NPP, Reactor Physics

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Power Density distribution

During operation the power density distribution is kept in a proper shape by:

- Reactor Physics calculation,
- Measuring the depth of the reflector rods and by adjusting the rods to the same depth,
- Measuring the height of the columns of the small sphere shutdown system and by adjusting them to the same height.

The power density distribution is controlled by the 3 x 4 neutron flux instrumentation outside the RPV.

An indirect control of the axial shape is given by measuring the helium-inlet and -outlet temperatures.

Under safety aspects the power density distribution is important as start condition for accident analysis.

Accident Analysis

Rules for
BWR, PWR,
Fuel Handling

HTR-Filter

Additional Aspects
of HTR-Design

List of:
- Design basic accidents
- Incidents
- Transients
- ATWS
- Hypothetical Events

BMI

RSK

KTA

IAEA SG-50-D1

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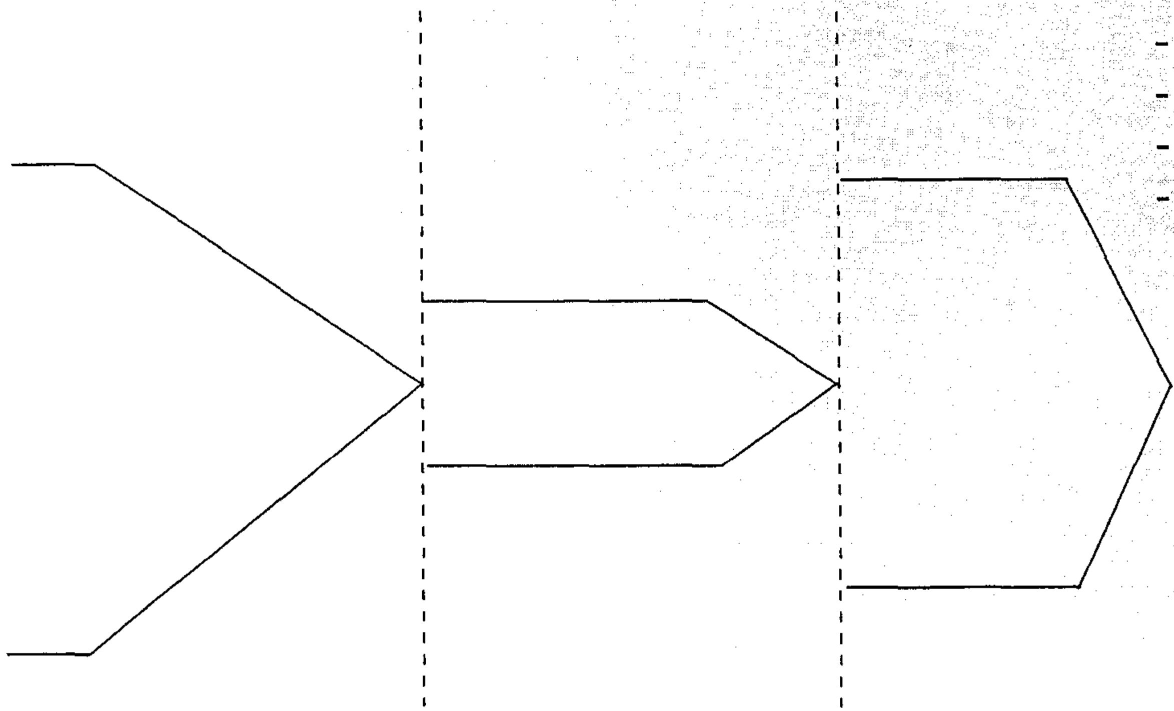

Accident Analysis

Reactivity Accident

Uncontrolled withdraw of reflector rods

Time [sec]	Power [%]	Action	Max. Fuel Temperature
0	105	---	~ 880 °C
(12	120	Scram (Power)	)
90	~ 200	Scram (Coolant Temperature)	~ 950 °C
120	< 10	---	
3700	Decay Heat	---	~ 1050 °C
~ some hours	Decay Heat	---	~ 1240 °C

Conditions:

- Conservative integral power, only area cooler system
- Skipping of first reactor protection limit

Result:

Fuel temperature well below limit: 1620 °C, also for different starting conditions like zero power, partial power.

Accident Analysis

Loss of Coolant Accident

Starting conditions:

- Reactor power 105 % (limitation system limit)
- Inlet temperature 280 °C (near protection system limit)
- Outlet temperature 750 °C (near protection system limit)

max

- 2. Scram limit ($\frac{dP_R}{dt} \geq |-20\%/\text{min}|$)
- Leak of 65 mm Ø

Calculation

Programs TAC 2D (TÜV) and THERMIX

Detailed 2-dim model with evaluation of the main tolerances (1 σ-value):

- Decay heat ± 2,8 %
- Local power density ± 5 %
- Fuel heat conductivity ± 1,15 W/m•K
- Effective core heat conductivity ± 5 %

Results for maximum fuel temperature (incl. 2 σ-tolerance)

THERMIX	TAC 2D	limit
1608 °C	1625 °C	1620 °C

(~ 1 % of the core > 1500 °C)

Modular HTR-2 NPP, Reactor Physics

Accident Analysis

Loss of Coolant Accident

Additional temperature results (THERMIX)

Position	Maximum temperature	Design limits
Reflector rods	880 °C	650 °C
Core container wall	500 °C	500 °C
Core container bottom	380 °C	500 °C
RPV wall	360 °C	500 °C
RPV bottom	350 °C	500 °C
Outlet temperature	750 °C	900 °C

The high temperatures of the reflector rods do not affect the conceptual design, because:

- The mechanical integrity of the rods can perhaps be approved by additional experiments,
- The shutdown safety is not affected by a loss of the rods during temperature increase,
- The rods can be changed.

MATRIX-METHOD FOR SEARCHING FOR POSSIBLE INCIDENTS (REACT

INCIDENT / REACTOR STATUS	ROD EJECTION	ROD DROP	ROD WITHDRAW	• •
LOADING	–	–	✓	
START UP COLD	✓	✓	✓	
START UP HOT	✓	–?	✓	
PARTIAL POWER	✓	–?	✓	
FULL POWER	✓	–?	✓	
• • •				

?: Power Distrib.-Influence
: Detectable?

MATRIX-METHOD FOR SEARCHING FOR POSSIBLE SYSTEMATIC ERRORS (FHS)

STATUS \ FUNCTION	FUEL-TRANSPORT	GRAPIT/FUEL..	BURN UP-MEASURE..	SPENT..
LOADING	△	△	—	—
UNLOADING	✓	—	—	✓
NORMAL OPERATION	✓	✓#	✓	✓
START UP CORE	✓	✓	✓?	✓
BURN-IN-CORE	✓	✓	✓?	✓
EQUIL.-CORE	✓	—	✓	✓

•

•

•

△: How?

?: Different Enrichment!

Filter for and Additional Aspects of HTR-Design (Shutdown Safety)

Event, Effect	PWR	HTR
Rod ejection	yes	no
Uncontrolled rod withdraw	yes	yes
Boron dilution	yes	no
Water ingress	yes	(no) 600 kg/m
Uncontrolled fuel addition	no	yes
Collapsing of holes in the core	no	yes
Loss of Collant Accident	yes	yes
Loss of Flow Event	yes	(no)
Rod drop with reactivity gain	no	yes
Core densification	(no)	yes
Boron-loss by earthquake	yes	no
Burn out of poison in structural materials	no	yes
Uncontrolled cooling of the core	yes	yes
Temperature increase of the reflector	no	(yes)
Error in excess reactivity	(no)	yes
ATWS	yes	yes
Changes of reflector geometry	no	yes
Burn out of reacitivity control devices	yes	yes
....	...	...

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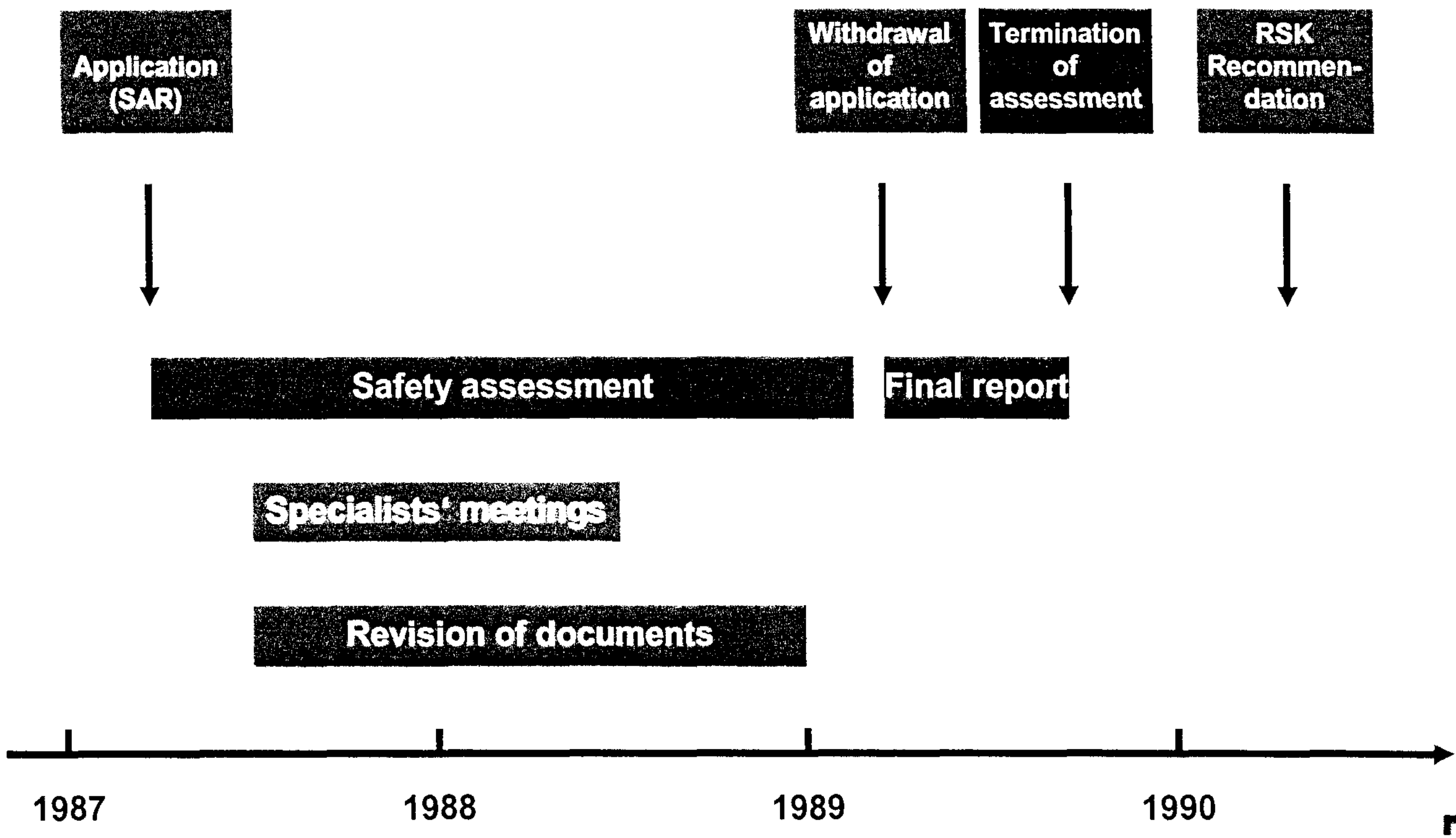
Topics - 2

Thursday, 26 July 2001

Safety Assessment of the HTR Module in Germany

- The task as defined in the contracts
- Overview of the plant concept
- The methodology applied in safety assessment of the HTR-2 NPP
- The most important results

Licensing of the German HTR-2 NPP



The Tasks as Defined in the Contracts - 1

Two different and consecutive steps in safety assessment of the HTR-2 NPP:

- Site-independent application for a preliminary license:

  ⇨ Task carried out for the Ministry of Environmental Protection in the federal state Lower Saxony

  ⇨ Compliance of the HTR-2 NPP concept with the requirements of applicable laws, ordinances and technical rules

  ⇨ Documentation of the safety assessment results in a Safety Assessment Report as a technical basis for the license to be granted

The Tasks as Defined in the Contracts - 2

- Compilation of the safety assessment results without reference to a licensing procedure

  ⇨ Task carried out for the Federal Ministry for Research and Technology

  ⇨ Compliance of the HTR-2 NPP concept with the requirements for nuclear facilities in Germany

  ⇨ Definition of topics where further development steps might become necessary

  ⇨ Documentation of the results in a report

The Tasks as Defined in the Contracts - 3

Aim of the second step: To ensure that future development steps will be carried out under consideration of the respective regulations for nuclear safety in Germany

- Consequence: The final report does not contain (formalized) proposals for licensing restrictions as usually in a safety assessment report prepared in a licensing procedure
- Instead, a number of (unformalized) recommendations is given concerning deficiencies with respect to the applicable laws,ordi nances and rules

Applicability of Nuclear Safety Regulations for the HTR -2 NPP

Class of regulation	Example	Relevance
Laws and Ordinances	Nuclear Energy Act Radiation Protection Ordinance (RPO)	Obligatory
Guidelines	BMI/BMU Criteria for NPP Event Guideline § 28.3 RPO	Partly Obligatory
Technical Rules	KTA Rules DIN/ISO Standards	Concept-Specific

Derivation of Assessment Criteria

"Filtering and enrichment" process typical for prototype plants:

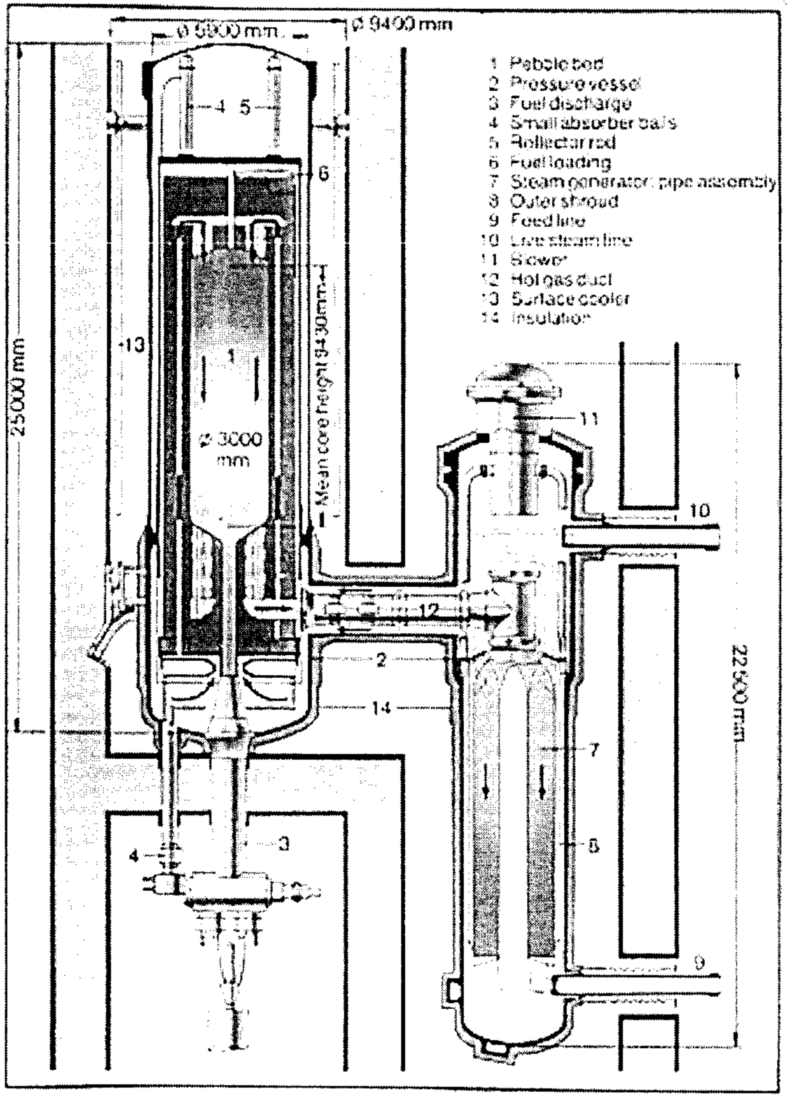
- Scanning of existing rules for HTR-2 NPP relevance
  - ⇨ Reduction of the existing requirements to those relevant for the HTR-2 NPP
  - ⇨ If applicable, consideration of concept-specific or intrinsic requirements
- Scanning of HTR-specific publications
  - ⇨ If applicable, consideration of HTR-specific published data

⇨ Comprehensive and consistent set of design and evaluation criteria applicable to the HTR-2 NPP

Design Basis of the HTR-2 NPP

Reactor concept characterized by limitation of fuel temperatures such that even in case of failure of all active cooling systems and the loss of coolant event no considerable release of radioactive fission products from the fuel elements will take place

The Pressure Vessel Unit of the HTR-2 NPP



Design Features - 1

- Modular HTR-2 NPP consisting of two reactors
- Two-circuit high temperature gas-cooled reactor:
  - ⇨ Primary circuit with helium as coolant
  - ⇨ Secondary circuit with steam turbine to generate power
- Power rating: 200 MW th; low power density
- Primary circuit and steam generator of secondary circuit housed in the Pressure Vessel Unit (PVU)

Design Features - 2

- Reactor core:
  - ⇨ Tall and slim geometry
  - ⇨ 2 independent shut-down systems: reflection rods and small spheres
  - ⇨ Core components: metallic, graphitic and carbon materials
  - ⇨ 360,000 fuel elements

- Fuel elements
  - ⇨ Spherical shape, diameter 60 mm
  - ⇨ Graphitic matrix
  - ⇨ Fuel kernels in TRISO particles
  - ⇨ Fuel design temperature: 1620 °C

Design Features - 3

- Decay heat transfer
  - ⇨ No active system required for short-term cooling after shut-down
  - ⇨ Decay heat removal passively via heat-up of core and area surface coolers outside RPV
  - ⇨ Fuel design temperature not exceeded
  - ⇨ Surface cooler to protect the reactor cavity

- Activity retention
  - ⇨ TRISO particles intact below fuel design temperature
  - ⇨ Pressurized boundary (PVU and other components)
  - ⇨ Fuel design temperature not exceeded during normal operation and after design basis events

Design Features - 4

- Safety enclosure/confinement
  - ⇨ No safety containment comparable to that of an LWR
  - ⇨ Safety confinement designed for
    - ◆ controlled and filtered activity release during normal operation
    - ◆ controlled and filtered activity release during and after small pipe break (Φ < 10 mm)
    - ◆ controlled and filtered activity release including unfiltered pressure relief of the building during and after large pipe break (Φ < 65 mm)

Design Basis Events

Listing of design basis events submitted by the applicants:

- Assessment criteria: Guideline §28.3 of the Radiation Protection Ordinance, modified under consideration of HTR-2-specific aspects
- Result: Revision and enlargement of the design basis event catalogue
- Action by the applicants: Inclusion of revised listing into the revised Safety Analysis Report

Basic Assumptions of the Event Analysis - 1

Basic assumptions of the event analysis needed to be modified in the event analysis, e.g.:

- Failure of the first criterion to activate the reactor protection system
- Consideration of a single failure and repair fault in systems crucial for event management
- Non-consideration of non-safety-related systems in the event analysis

Action by the applicants: Revision of the initially submitted event analysis in the revised Safety Analysis Report

Basic Assumptions of the Event Analysis - 2

Effect of revision of the event analysis:

- Introduction of additional safety measures, e.g. control rod insertion limitation and additional criteria to activate the reactor protection system
- Modification of limiting safety-relevant data, e.g. increase of the fuel element design temperature from 1600 to 1620 °C
- Additional analyses to demonstrate the acceptability of the revised design

Shut-Down Margin

Requirement introduced by TÜV: Shut-down in all operation modes to a core temperature below 50 °C

Results of safety assessment:

- Necessary measures: insertion limitation of the control rod system and filling height limitation of the small spheres shut-down system
- Necessity of an initial start-up measuring programme causing a modification of the loading strategy

Reactivity Events

Confirmation by TÜV that withdrawal of all reflector rods with maximum speed during full load operation is the relevant design basis reactivity event

Results of safety assessment:

- Power limitation to a maximum of 105% of the rated reactor power necessary
- Precautions against an unintended blower start at high helium temperatures
- Further investigations concerning the compaction of the pebble bed due to an earthquake

Disturbed Heat Removal Without Loss of Coolant - 1

Analysis of the applicants: Failure of the external mains feed simultanuously to loss of auxiliary power due to failure of the emergency power generation system for the cases:

- Short-term loss of auxiliary power (< 2 hours)
- Medium-term loss of auxiliary power (< 15 hours)
- Long-term loss of auxiliary power (> 15 hours); not included into basic design by the applicants

Disturbed Heat Removal Without Loss of Coolant - 2

Results of safety assessment:

- Confirmation that no limiting data (e.g. fuel temperature, radiation exposition) exceeded for low- and medium-term loss of auxiliary power
- Exclusion of long-term loss of auxiliary power from basic design only acceptable, if adequate QA measures foreseen ensuring a short-term repair of the emergency power system

Loss-of-Coolant Events - 1

Analysis of loss-of-coolant PRIMARY and SECONDARY loss-of-coolant events with respect to:

- Maximum temperature of fuel and components
- Resulting loads, e.g. differential pressure loads on the reactor building or graphite corrosion
- Radiation exposition

Design basis for PRIMARY COOLANT COMPONENTS:

- Exclusion of a failure of the pressure vessel (basic safety)
- Rupture of a large connection pipe (Φ < 65 mm)
- Rupture of a small connection pipe
- Rupture of pipes with primary coolant outside the reactor building

Loss-of-Coolant Events – 2

Results of safety assessment for PRIMARY loss-of coolant events:

- Break exclusion confirmed on the basis of fracture-mechanical analyses, but investigation programme and inset probes necessary to determine the neutron-induced embrittlement of the RPV
- Main steam nozzle: Mixed welding seam to be shifted from the high-temperature area
- Modification of the thermal insulation of the RPV necessary
- Filtering concept for medium size ruptures to be modified
- Confirmation that the rupture of a large pipe covers the consequences of all postulated ruptures
- Fuel design temperature not exceeded
- Radiation exposition limits not exceeded

Loss-of-Coolant Events – 3

Results of safety assessment of SECONDARY loss-of-coolant events:

- Modification of the main steam lock-off system to ensure pressure values within the reactor building below limiting values
- Differential pressure values close to the rupture position higher than the design value; to be considered in building design (not concept-relevant)
- Ingress of water into the primary circuit:
  - ⇨ Leakage quantity restricted to 600 kg
  - ⇨ High-quality initiation of countermeasures required
  - ⇨ Reactivity gain within acceptable limits
  - ⇨ Corrosion attack on fuel elements within acceptable limits

External Events

Results of safety assessment:

- Seismic design of some components and structures needed to be improved
- Probabilistic method to determine the seismic data acceptable
- Aircraft impact and shock after a detonation not to be considered as design basis events due to their low frequency to occur
- Plant design suitable to confirm the precaution of sufficient risk-reducing measures for events beyond design basis

Summary

- Consequent and consistent plant concept characterized by pronounced features of inherent safety
- Concept is suitable to meet the requirements of the design basis as well as the regulatory requirements
- Proposals of the TÜV for further investigations or modifications of the detail design are not concept-relevant, instead they confirm the logical consistency of the basic safety concept of the HTR-2 NPP
- There is no doubt that with respect to safety a license could have been granted

TÜV Hannover/Sachsen-Anhalt e.V.
Division Energy and Systems Technology
Certified according to DIN EN ISO 9001

TÜV NORD GRUPPE

Hanover, June 1998

Safety Assessment of the Design of the Modular HTR-2 Nuclear Power Plant

- Summary -

May 1990

Translated from *"TÜV Hannover • HTR-2-Modul-Kraftwerksanlage • Sicherheitstechnische Konzeptbeurteilung • Kurzfassung • Mai 1990"*

The task documented in this report was carried out on behalf of the Federal Ministry for Research and Technology and assigned to the identification no. 03 TVH 201. The responsibility for its contents is with TÜV Hannover e.V.

OPBZ8028

Table of Contents

1 Introduction

1.1 Extent and Course of the Task

In April 1987 the Siemens and Interatom companies applied for a provisional decision according to § 7a of the German Nuclear Energy Act concerning the design of a modular HTR-2 Nuclear Power Plant for combined generation of electrical power and process steam or heat for district heating, respectively. The application was submitted to the Ministry for the Environment of the German state of Lower Saxony.

TÜV Hannover/Sachsen-Anhalt e.V. (at that time TÜV Hannover e.V.) was contracted by the aforementioned ministry to assess the safety of the modular HTR-2 NPP in the licensing procedure. TÜV Rheinland e.V. was subcontracted for specific tasks by TÜV Hannover/Sachsen-Anhalt e.V..

The basis of our first preliminary investigations was the applicants' safety analysis report dating from April 1987, which they had submitted to the licensing authorities as addendum to the application. During a sequence of specialists' meetings taking place in the second half of the year 1987 and in the first half of the year 1988 we substantiated our request for further, more detailed reports concerning the design of the modular HTR-2 NPP and we proposed several modifications of the safety concept to be carried out by the applicants.

As a result of our proposals and requests the applicants modified the HTR-2 design and submitted further technical reports for our safety assessment. Additionally, the safety analysis report /1/ was revised. The task to update and complete the documents was finished early in 1989.

In April 1989 the application for a previsional decision was withdrawn; the licensing procedure was terminated by the Ministry of the Environment and the contract with TÜV Hannover/Sachsen-Anhalt e.V. was cancelled.

At that time the task to assess the safety of the modular HTR-2 NPP was in an incomplete, but largely advanced state. For this reason TÜV Hannover/Sachsen-Anhalt e.V. offered to the Federal Ministry for Research and Technology to perform an assessment of the HTR-2 NPP safety concept - in this case not related to a licensing procedure - and to evaluate, if this concept complied with the general safety requirements on nuclear installations in Germany and which conceptual features should be modified. This task aimed at ensuring that possible future research and development activities were compatible with the German safety standards concerning nuclear facilities.

TÜV Hannover/Sachsen-Anhalt e.V. was entrusted with this task and submitted its safety assessment report /2/ of roughly 900 pages in October 1989. In the following we summarize the relevant results of the aforementioned report and we demonstrate by several examples, which modifications of the NPP design methodology resulted from the safety assessment. *(Remark for comprehension: Although the tasks for the Lower Saxony Ministry of the Environment and the Federal Ministry for Research and Technology differ markedly as the second task does not fall within the scope of a licensing procedure, we consistenly use the term "applicants" for the Siemens and Interatom companies in this report to avoid possible misunderstandings.)*

1.2 Assessment Criteria and Design Basis Requirements

The German Nuclear Energy Act /3/ and the Radiological Protection Ordinance /4/ form a general legal basis for planning, construction, operation, and decommissioning as well as supply and waste management of nuclear facilities and thus define the framework for safety assessment of these installations. They are not restricted to specific plant concepts and technical construction details. The requirements imposed by the Nuclear Energy Act and the Radiological Protection Ordinance have to be met under any circumstances.

The legal basis is supplemented by a series of technical rules and guidelines, which largely relate to technical concepts and which had been elaborated continuously accompanying progress of nuclear technology. They aim at defining the state of the art and shall enable the manufacturer of a nuclear plant as well as the expert to apply uniform evaluation criteria in their tasks. Most of the rules and guidelines apply to LWRs. Thus the German rules and guidelines in nuclear technology mainly relate to the physical and technological features of LWRs and especially to those of pressurized water reactors, which are characterized by a nuclear core design with high power density. For this reason the rules and guidelines cannot be transferred to reactor concepts differing substantially from the LWR without modification.

Gas-cooled high temperature reactors are characterized by pronounced features of inherent safety against reactivity transients and disturbed heat removal. This applies especially to units with low power and low power density in the reactor core. Thus it would have been inadequate to transfer the unmodified requirements on LWRs to the modular HTR-2 NPP, particularly as in the latter case a reactor concept had been developed by consequent exploitation of low-power HTR safety characteristics, which even in the case of failure of all active cooling systems and in the loss-of-coolant accident will limit the fuel temperature in such a way that no relevant release of radioactive fission products will occur. This property reaches far beyond technologically realized standards, which implies that even the rudimentarily existing German rules and guidelines for HTRs had to be adapted before applying them to the modular HTR-2 NPP.

Therefore we applied in our assessment of the HTR-2 safety concept - apart from the concept-independent legal basis - the technical rules and guidelines only to such an extent as they are compatible with the concept itself. In our report to the Federal Ministry for Research and Technology we have justified in detail our procedure to deviate from technical rules and guidelines, when necessary. It was, however, beyond the scope of our task to perform a complete assessment of the applicability of all rules and guidelines in nuclear technology to the modular HTR-2 NPP design.

According to the Nuclear Energy Act the protective measures against damage from nuclear energy have to comply with the state of science and technology. This means for a prototype plant that additionally to the modified technical rules and guidelines the current state of research and development must be taken into consideration. For this reason we referred extensively to results presented in publications.

1.3 Assumed Site Characteristics

The conceptual design of the modular HTR-2 NPP was developed without referring to a concrete site of the planned NPP. For this reason site characteristics were assumed in such a way that they are representative for a large number of potential sites. Site-independent characteristics, referring e.g. to aircraft impact, were assumed according to the applicable technical rules and guidelines.

In our safety assessment we verified the site characteristics given by the applicants in their safety analysis report only with respect to completeness and conclusiveness. Further we verified if the chosen procedures to generate the site characteristics complied with the applicable German rules. We mainly paid attention to those characteristics, which could affect conceptual design and operation of the plant according to our general experience with nuclear facilities.

The result of our evaluation was that the assumed site characteristics, given by the applicants in their safety analysis report, take into account all relevant loads and design requirements to be imposed on nuclear facilities site-independently and cover to a large extent the properties of potential sites. As, on the other hand, the assumptions cannot replace all realistic characteristics, it will be necessary to evaluate a concrete site for the modular HTR-2 NPP for its suitability and compatibility with plant design. Therefore we have to state that the results of our safety assessment are valid only for those sites, which are compatible to the assumed site properties. As an example we based our evaluation of the emergency power supply concept on reliability data, which are typical for Germany.

2 The Modular HTR-2 NPP

2.1 Basic Safety Design Features

The nuclear core design of the modular HTR-2 NPP is such that even in the case of an assumed long-term failure of all active installations for heat removal and a simultaneous failure of the scram system the fuel temperature will not exceed its acceptable maximum value (fuel design temperature), which amounts to 1620 °C. This inherent safety feature is mainly caused by the large difference between fuel temperature under normal operation conditions and fuel design temperature, which will cause an intrinsic shut-down of the reactor, as well as the low power density of the reactor core in conjunction with its slim geometry and the design of the surrounding structural components, which will enable passive heat removal from the core.

The aforementioned scenario is a hypothetic combination of two different incidents and extends beyond design basis assumptions. It demonstrates, however, the inherent safety features of the modular HTR-2 NPP, which form the basis of its safety concept and which have to be taken into account in our safety assessment. In the following we will deal with the question, how the dominant protective requirements to grant the necessary safety of nuclear power plants - i.e. the required shut-down safety margin, decay heat removal and retention of radioactive substances - are met by the modular HTR-2 NPP, which is characterized by its inherent safety features.

Shut-down Safety

Due to its design the reactor core can be shut down already by insertion of neutron absorbers in vertical openings of the side reflector. There are two different measures, which are called the "first" and "second shut-down system".

The first shut-down system consists of the reflector rod system. According to the safety analysis report this system is designed in such a way that it will shut down the reactor sufficiently fast and keep it in the state "hot, below criticality" both from normal operation conditions and in design basis accidents. The system design takes into account the assumed failure of the most reactivity-efficient reflector rod. For thermal decoupling of the reactor and the steam generator as a prerequisite to keep the reactor in the state "hot, below criticality" the primary circuit blower must be shut off, too, i.e. both the first shut-down system and the primary circuit blower are shut down automatically and simultaneously by the reactor protection system. Due to its task to keep the reactor in the state "hot, below criticality" the first shut-down system is also called "hot shut-down system".

The second shut-down system consists of the small-sphere shut-down system. This system is designed such that it will shut down the reactor from normal operation in those cases, which do not require fast reactivity changes, and keep it continuously below criticality at 50°C, the lowest operation temperature. The small-sphere shut-down system is also called "cold shut-down system". This system will be initiated by manual action and not automatically.

According to the aforementioned inherent safety properties of the reactor, it can also be shut down by interrupting the primary coolant flow, which means the primary coolant blower is turned off. This action would not cause an immediate interruption of heat generation; however, as a result of disturbed heat transfer to the secondary circuit the core temperature will rise and the reactor will stabilize below criticality on a higher temperature level due to the negative temperature coefficient of the reactivity. The applicants did not take into account this effect in their shut-down concept, as shut-down of the blower and insertion of the reflector rods are always initiated simultaneously.

Decay Heat Removal

Due to the intrinsic property of the modular HTR-2 reactor to remove the decay heat after shut-down passively by heating up the surrounding structural components, there is no need to integrate the coolant circuits, especially the secondary circuit, into the safety concept, as the fuel temperature will be below fuel design temperature under any circumstances. Thus the main protective task of the active coolant systems is to limit the temperature of the concrete structures surrounding the reactor core and of the reactor pressure vessel including its components to acceptable values.

This protective task is valid for the concrete structures also under normal operation conditions, as the heat due to energy losses of the reactor pressure vessel will heat up the reactor cavity. For this reason the reactor cavity is equipped with a surface cooler. The supporting components of the reactor pressure vessel and the socket integrated into the vessel bottom to join the fuel removal pipe are cooled separately. If heat transfer to the main heat removal system is interrupted, the surface cooler will be active to prevent excess temperatures due to decay heat after shut-down of the reactor.

The applicants have performed analyses to demonstrate that there is no need for short-term availability of the active cooling systems, as the design temperatures of the reactor cavity concrete structures will only be exceeded after 15 hours.

Retention of Radioactive Substances

The spherical fuel elements of the modular HTR-2 NPP contain the fuel in multiply coated particles (TRISO-particles), which are embedded into a graphite matrix. Due to these features the fission products - which form the bulk of all radionuclides in the reactor core - will be enclosed nearly completely, provided that the fuel design temperature is not exceeded. Thus the intrinsic limitation of the fuel temperature and of mechanical loads to values below design limits due to its inherent safety features is one of the basic properties of the modular HTR-2 NPP.

The radioactive substances released to the coolant originate either from only few defective coated particles or from fission or activation of uranium traces in the graphite matrix of the fuel elements. It has been demonstrated successfully both by experiments and by operation experience that under the given circumstances the resulting coolant activity in the primary circuit will be very low.

In a loss-of-coolant accident the fission gas activity of the coolant and part of the plate-out activity on the primary circuit surfaces would be released to the reactor building and via the ventilation stack to the environment. According to different analyses the resulting radiation exposure in the environment would be far below the accident dose limits laid down in the German Radiological Protection Ordinance. For this reason the design of the modular HTR-2 NPP does not include a gastight containment.

2.2 Technical Design

In this section we give exemplary results of our safety assessment, which refer to important components and systems of the modular HTR-2 NPP. For our summary we have selected those systems and components as well as those results, which are relevant for the design of the modular HTR-2 NPP and its operational and accident behaviour. The complete and consistent results of our investigations are compiled in our safety assessment report of the modular HTR-2 reactor design.

2.2.1 Nuclear Steam Generation System

The modular HTR-2 NPP consists of two reactors, each of which being designed for a thermal power of 200 MW. For this reason two independent nuclear steam generation systems are foreseen, which to a certain extent are connected to common auxiliary systems.

Each steam generation system has the task to transfer the heat generated by fission in the reactor core to the feedwater-steam circuit in the steam generator. The coolant in the primary system is highly purified helium. Under normal operation conditions the heat transport occurs by forced convection, where the coolant flow is maintained by the primary circuit blower. If the blower is turned off, the decay heat will be removed from the core by radiation and natural convection via the reactor pressure vessel to external cooling systems (surface cooler).

The nuclear steam generation system consists of the following main components:

- reactor pressure vessel (RPV) including core, core components and shut-down systems,
- connecting pressure vessel including the hot gas duct,
- steam generator pressure vessel including its components, the primary circuit blower and the blower isolation valve.

These three main components form a common pressure vessel unit. The pipes to adjoining systems are connected to sockets and are led out of the primary cell by means of wall passages. All valves in these pipes are arranged outside the primary cell. This applies to the primary circuit isolation valves, too. The steam generator pressure vessel is shifted to a lower level with reference to the reactor pressure vessel and arranged at the side of the latter. Thus the hot gas duct will be very short. Due to the low positioning of the heat sink as compared to the heat source, any interruption of forced convection will be followed by only poor natural convection. This complies with the intention that the decay heat shall only be removed from the RPV walls to the surface cooler.

the effects of incidents caused by failures in the secondary circuit, especially water ingress into the primary circuit, are limited distinctively by arrangement and design of the primary circuit components. The feedwater and steam pipings are arranged on a comparatively low level and decoupled from the RPV. Thus a direct contact of the reactor core with water in its liquid state can be excluded for all relevant design basis accidents; only a contact to a helium/steam mixture has to be taken into consideration. All investigations concerning fuel element and material corrosion can be based on this boundary condition.

The arrangement of the primary circuit components in the modular HTR-2 NPP is advantageous for operational reasons and especially for safety aspects. On the other hand the primary circuit design as a basis-safe pressure vessel unit means high requirements on design, construction, materials, weldings and the support concept of the components.

Fuel Elements

The reactor core consists of about 360,000 spherical fuel elements, which are arranged within the ceramic and metallic core components in the form of a pebble bed. The spheres have an outer diameter of 60 mm. The fuel is embedded within an inner zone of the spheres (50 mm ∅) and enclosed in particles consisting of the fuel-containing kernels

(0.5 mm ∅), which are surrounded by several pyrolytically deposited carbon layers and a layer of silicon carbide (SiC). These coated fuel particles are distributed evenly within a graphite matrix.

The task of the spherical fuel elements is to generate heat by fission of the nuclear fuel. Loads resulting from handling, operation and possible accidents and affecting both the fuel elements and the adjoining components have to be limited in such a way that the dominant protective requirements can be met. For this reason the fuel elements have to be designed such that the

- fission gas release will be limited to acceptable values,
- strength of the spherical fuel elements will be sufficient,
- fuel element corrosion will range within acceptable limits,
- dimensional stability will be sufficient.

In their documents the applicants have described the planned fabrication methods of the fuel elements for the modular HTR-2 NPP and indicated, which quality assurance methods are foreseen to grant an even and sufficiently high quality level in fuel manufacture.

The existing manufacture and operation data concerning the foreseen fuel type are mainly based on the experience made with the AVR reactor. Additional extensive experience with different fabrication steps stems from the manufacture of THTR reactor fuel elements. Furthermore fuel elements and fuel samples were manufactured for a great number of irradiation experiments in material test reactors.

Ensuring the product properties - especially fission product retention - is of conceptual importance in safety assessment of the modular HTR-2 NPP. According to our opinion the applicants have demonstrated successfully by referring to extensive manufacturing experience made with fuel elements and fuel samples for irradiation experiments and the AVR reactor that fuel elements with properties required for the modular HTR-2 NPP can be manufactured. We further expect that the quality assurance system and the correspondig manufacture and inspection methods are suitable for transition to a large production scale and thus ensure the required fuel properties.

For our results concerning fission gas release from the fuel please refer to section 4.2.

Core Components

The relevant construction feature of the ceramical core components consisting of either graphite or carbon material is to divide the structure into single blocks, as has been demonstrated succesfully in the high temperature reactors AVR and THTR. The metallic core

core components form a support structure and are designed to bear the loads due to normal operation and under accident conditions. The ceramic core components, which form the upper, side and bottom reflectors, are made from graphite materials and will due to their neutron-physical properties reflect the neutrons escaping from the core to the pebble bed again. The outer part of the reflector will be made from carbon material, which has a lower heat conductivity as compared to graphite and thus will protect the adjoining metallic components against excessive temperature loads. The bottom layer and the neighbouring layer of the side reflector contain boron as a neutron absorber to reduce the neutron irradiation of the reactor pressure vessel, the metallic core components and the hot gas duct.

In our safety assessment we have verified by performing design calculations that thermally induced displacements of the core structure components will not affect shape and dimensions of the core enclosure and of the openings for the control rods and shut-down eqipment as well as for the coolant gas flow.

Shut-down Systems

In section 2.1 we have already described the shut-down concept. In the following we will summarize the results of our evaluation concerning both shut-down systems of the modular HTR-2 NPP with respect to their construction and functional safety.

According to the shut-down concept of the modular HTR-2 NPP the reflector rod system is the first shut-down system and serves - in conjunction with the simultaneous blower shut-down - as a scram system, which is designed to reduce the reactor power from normal power operation and under design basis accident conditions and to keep it in the state "hot, below criticality" as long as necessary, even if the most efficient reflector rod has failed.

Apart from this safety-related task the reflector rods are designed for reactivity control of the reactor under power operation conditions.

The reflector rod system corresponds in its relevant properties to that of the THTR reactor. It consists of six reflector rods and their drive and control units. The reflector rods are suspended by connecting them to their drives by chains and are moved up and down in the openings of the side reflector. The six control rod drives are arranged above the the so-called thermal upper shielding within the reactor pressure vessel.

The rod insertion limitation shall ensure the shut-down reactivity required to scram the reactor by the reflector rods at any time and from all operational modes.

The strength design of the reflector rod components and of their drives must be compatible with the loads to be expected, and their operational behaviour has to be verified experimentally and by evaluation of operation experience made with comparable systems to ensure an undisturbed and highly reliable shut-down behaviour of the reflector rods, even after long-term operation.

For this purpose the applicants have planned an evaluation of already performed prototype experiments with THTR-specific components as well as experiments to be carried out with a prototype specific for the modular HTR-2 NPP. In our view the foreseen experiments, inspections and calculations will be suitable to demonstrate that the detailed design and construction will meet all requirements of the KTA-rules to be applied to the first shut-down system.

The initiation concept of the scram system has been realized in comparable nuclear power plants and has proved to be reliable.

The small-sphere shut-down system is the second shut-down system and shall shut down the reactor from normal operation in those cases, which do not require fast reactivity changes, and keep it continuously below criticality at 50°C, the lowest operation temperature. In case of a scram it will in conjunction with the reflector rod system keep the reactor in the long-term state "cold, below criticality". Apart from this safety-related task the small-sphere shut-down system is involved in reactivity control of certain normal operation modes.

The small-sphere shut-down system contains 18 units, which are independent from each other. One unit consists of the following main components:

- Storage container with container lock and cyclone separator,
- removal container,
- transportation pipe and transportation gas return pipe,
- small-sphere shut-down elements.

Shut-down of the reactor is achieved by dropping the neutron-absorbing small-sphere elements from the storage containers into openings (oblong holes) in the side reflector.

The design of the small-sphere shut-down system planned by the applicants is new. According to the state of science and technology the functional safety of shut-down installations, which have a safety-related task, has to be demonstrated. The relevant basis for such a demonstration will be a suitability test carried out with a prototype unit of the newly developed shut-down system, if the applicant cannot refer to positive operation experience.

The safety function of the small-sphere shut-down system requires to open reliably the trail for the spheres to drop from the storage containers into the oblong holes in the side reflector. Opening the trail is achieved by opening the container lock, which is designed in a very simple way and consists of only few movable parts. The relevant requirements on the function of the container lock are:

- Reliable opening,
- suppression of arching during sphere discharge,
- discharge of partial lots without damage to the spheres due to closing the lock.

We expect that these requirements can be met by the design planned by the applicants and that the necessary functional safety of the container lock can be demonstrated successfully by the foreseen suitability tests.

As the small-sphere shut-down system is safety-relevant apart from its importance in normal operation, a level limitation of the sphere columns in the oblong openings in the side reflector, correlated to the level indication of the storage containers, is required to ensure the necessary shut-down reactivity.

The small-sphere shut-down system shall be initiated by manual operation. This design feature complies with the results of the accident analysis.

Pressure Vessel Unit

The pressurized walls of the pressure vessel unit will be made from the heat-resistant, fine-grained and heat-treatable steel 20 MnMoNi 55. This material has been qualified for operation in nuclear facilities at temperatures up to 375 °C and is suitable for the planned application range.

Nevertheless, additional investigations are required to gain material data relevant for material embrittlement due to neutron irradiation in the temperature range of about 250°C, the foreseen operation temperature of the HTR-2 reactor pressure vessel. These investigations shall be performed within an irradiation program preceding operation and simulating long operation times. The aim of this program is to gain a relation between the relevant material data and neutron fluence at temperatures typical for the HTR. Additionally the applicants are planning to compare irradiation results at low and high neutron flux densities to demonstrate that simulating long operation times by high neutron fluence densities will lead to conservative results. The applicants will carry out material tests of the samples after irradiation to verify the shift of the brittle fracture transition temperature, upon which the fracture-mechanical design analyses have been based. If a higher value results as compared to design calculations, further fracture-mechanical samples will be irradiated. To verify the progress of embrittlement due to neutron irradiation, material samples stemming from the original material of the reactor pressure vessel will be irradiated during operation of the modular HTR-2 NPP. As for plant-specific reasons these irradiation investigations cannot be carried out applying a lead factor, the sample number and the discharge frequency will be increased as compared to the requirements of the KTA-rules to predict reliably the expected increase of irradiation embrittlement between two discharge steps. The planned positions of the sample holders close to the reactor pressure vessel wall will ensure that temperatures and fluences at the sample locations are representative for the vessel material.

The pressure vessel unit is designed for the assumed failure of the largest pipe (65 mm ∅) joined to the vessel, i.e. the occurence of larger leaks is not postulated. This assumption determining the design of the modular HTR-2 NPP is justified by the applicants by referring to the planned design and surveillance measures that will exclude a failure of the pressure vessel unit during the lifetime of the plant. In their opinion especially the

- high ductility of the material,
- stress limitation by meeting the requirements of the KTA-rules,
- monitoring of possible flaws in the components after manufacture and after each operation cycle

justify the exclusion of fractures penetrating the vessel walls. Thus they exclude leakages for any wall area of the pressure vessel unit. Additionally the applicants carried out fracture-mechanical investigations of the pressure vessel unit to justify the exclusion of fractures.

According to our assessment the applied methods to justify the exclusion of cracks are suitable. We have verified the results of the applicants' investigations by performing independent calculations with our computer codes. In our opinion the applicants' reasons to exclude fractures of the pressure vessel unit are well-founded. This result is supported by the fact that the area of fracture, which can be determined by assuming a very long duration of operation extending the lifetime of the plant, but can be excluded to occur during its lifetime, will be below the area of fracture represented by the pipe with a diameter of 65 mm joined to the vessel. The fracture length occurring on the basis of these assumptions will be far below the critical fracture length, i.e. critical failure can be excluded.

Based on our evaluation of component experiments, which aimed at demonstrating safe operational behaviour of the welding between ferritic and Incoloy materials at operation temperature of 530 °C, and our crack propagation calculations we came to the result that for the designed socket of the steam duct located at the steam generator we cannot confirm the exclusion of fractures.

Therefore the applicants have redesigned the socket of the steam duct avoiding consequently weldings between ferritic and Incoloy materials in this area characterized by design temperatures above 350 °C.

The measures taken in redesigning are suitable to minimize the loads on the components of the socket, especially by avoiding stress effects due to inhibited thermal expansion of the different materials at temperatures above 350 °C. The stress calculations performed by the applicants for the most important operational modes covering the loads to be expected meet the requirements to be imposed on fracture mechanical calculations. We have verified these caculations by performing independent calculations with our computer codes and found that for the design lifetime of the plant the occurence of leakages and fractures can be excluded, as even the propagation of large initiating flaws will be reduced both in depth and longitudinal directions to such an extent that no wall-penetrating fracture will occur.

Further investigations on fracture propagation demonstrate that only after operation intervals extending by far the design lifetime of the plant the occurence of a leakage becomes possible. However, this will not induce a rupture of the socket of the steam duct. The possible maximum leakage cross-sections - not to be expected during the lifetime of the plant - has been determined in a sufficiently conservative manner and is covered with respect to its consequences by the design leakage cross-section. The design leakage between the primary and secondary circuit is the double-sided rupture of a steam generator heating tube. According to our safety assessment this assumption is justifiable, taking into account design, materials and planned quality assurance measures during fabrication.

In our opinion a failure of the steam generator heating tubes in the course of an assumed rupture of the feedwater or steam ducts has not to be postulated, as the heating tubes are designed for the loads from these events. We further agree with the applicants that the double-sided rupture of a heating tube has not to be postulated to occur simultaneously to the rupture of the steam duct, as only reduced pre-damaging of the HTR heating tubes as compared to the LWR will occur due to the effect of the precautionary measures in water chemistry; further the foreseen moisture measurement equipment will be able to identify very small leakages already. However, in contrast to the applicants we see the need for regular operational non-destructive tests to verify the state of the heating tubes during operation.

The design to support the pressure vessel unit provides both for bearing the operational and accident loads and for facilitating uninhibited heat extension. The support components will be designed such that on the one hand the pressure vessel unit will keep its position in the building under the influence of outer forces and on the other hand they will not inhibit radial and axial displacements due to temperature expansion.

In our view the planned design to support the pressure vessel unit is suitable to bear all loads in horizontal and vertical direction resulting from the weight of the vessel unit as well as from operation and accidents. It further facilitates a mostly uninhibited extension of the pressure vessel unit due to temperature influence. For this purpose the applicants will provide the plant with proven and reliable sliding bearings, which are characterized by a careful finish of the sliding surfaces and well-adapted selection of the lubricant, thus avoiding an unacceptable restraint. Further the planned arrangement of guide units on the different support levels will facilitate the vessels to displace as required and to avoid unacceptable compulsive forces in those areas, where they could become relevant.

We have assessed the need to regularly inspect the support components during operation referring to their safety relevance. In our view regular operational visual inspections combined with surface inspections and gauging will be necessary, whereas the applicants plan to inspect the shock absorbers only.

It is the aim of the visual inspections to verify the unrestrained moveability of the support components and the shock absorbers as well as the absence of any deformation, damage or corrosion effect. Gauging will be applied to verify the adjusted cold and warm clearances to avoid unacceptable restraints in displacing the pressure vessel unit during operation. Surface inspections, as dye penetrant and magnetic particle tests, will be applied, if surface damage is to be assumed.

2.2.2 Secured Intermediate Cooling System Including the Surface Cooler

Exclusively the following coolant systems of the modular HTR-2 NPP are designed for safety-related tasks:

- the secured intermediate cooling system including the surface cooler and
- the secured service cooling water system.

Both systems together form a two-train redundant sequence of coolant systems. The secured coolant system is connected to the

- surface cooler,
- support of the pressure vessel unit,
- socket in the reactor vessel bottom (fuel element discharge pipe),

which shall remove the radiation and convection heat originating from the unisolated part of the reactor pressure vessel, the support of the pressure vessel unit and the fuel element discharge pipe to protect the concrete structure against excess temperatures.

Furthermore the surface cooler shall maintain decay heat removal after failure of the main heat sink to protect not only the concrete structure against excess temperatures, but also the reactor components, especially the reactor pressure vessel. This must be achieved after external events, too.

Due to the inherent safety properties of the modular HTR-2 NPP active heat removal is not required to keep fuel temperatures below fuel design temperature, but to limit the temperatures of the reactor pressure vessel and of the concrete structures forming the primary cavity. With respect to plant design active heat removal may be interrupted for 15 hours.

Our safety assessment has confirmed the design criteria applied for the active cooling systems. Thus we judge the operation modes of the active cooling systems to be adequate and to comply with the applicable safety requirements.

The surface cooler will be arranged about 10 cm above the inner wall of the reactor cavity; its distance to the reactor pressure vessel will be about 1.5 m. It consists of eight bolted sections, which will be suspended in a way that no compulsive forces due to heat extension will occur. The sections will consist of vertical tubes, which are connected by welded bars. The tubes are assigned in an alternating pattern either to redundance 1 and 2 of the secured intermediate cooling system or to the nuclear intermediate cooling system. According to its assignment each cooling tube is connected in its lower part to one out of three headers.

The surface cooler of the modular HTR-2 NPP has to remove about 400 kW during normal operation and about 850 kW after failure of the main heat sink. Each of the three planned cooling trains is designed for the aforementioned requirements.

The coolant flow and reflux can be interrupted by valves arranged within the reactor building. Between the components to be cooled and the isolation valves connecting sockets for fire hoses are foreseen.

The secured intermediate cooling system has to fulfil its safety-related task also during maintenance and simultaneous occurence of a single failure. This means for the two-train redundant system that it must be acceptable to interrupt decay heat removal for a sufficiently long time and to finish the maintenance activities before limiting design values are exceeded.

The modular HTR-2 NPP is characterized by the necessary inherent safety-related features required for the aforementioned procedure; according to our safety assessment active heat removal is not required to avoid excess temperatures above fuel design temperature and cooling of the concrete structures as well as of the reactor components can be interrupted for 15 hours without damage. Thus a twofold redundant system will be sufficient, provided that both redundant trains will not fail due to the same cause. We have taken into account the latter requirement by investigating the mutual interaction between both trains.

After failure of the intermediate cooling systems the system pressure can rise from 5 bar to approximately 20 bar within 15 hours due to heat-up of the surface cooler. Under these circumstances a damage of the surface cooler encompassing both redundant trains of the cooling system due to an assumed failure of one of the surface cooler headers cannot be excluded completely.

In our view this combination of events has an extremely low probability to occur, as the surface cooler including its headers will be designed for at least 40 bar and thus distinctly above system pressure in the discussed incident. Furthermore the initiating incident - simultaneous failure of all intermediate cooling trains - is classified as a design basis accident having a very low probability to occur.

2.2.3 Safety Enclosure

The safety concept of the modular HTR-2 NPP does not provide for a pressure-resistant and gastight containment to enclose released radioactive nuclides. Instead, the planned safety enclosure shall facilitate activity control in accidents accompanied by radioactivity discharge to the environment and serves to minimize the radiation exposure. The safety enclosure consists of

- the reactor building,
- the installations for building relief and ventilation system isolation,
- a subatmospheric pressure ventilation and filter system.

This design provides for separate procedures to meet the dominant protective requirement. For this reason the applicants have designed the safety enclosure referring to two different assumed courses of event, which are described in the following.

During a primary circuit leakage with a cross-sectional area corresponding to a measurement pipe having a maximum diameter of 10 mm, no excess pressure within the reactor building as compared to the environment will build up. The accident will be indicated by the room activity monitoring system. The applicants are planning to ventilate the building by the subatmospheric pressure ventilation system equipped with aerosol and activated charcoal filters.

In the case of a postulated rupture of a pipe (∅ 65 mm), which is joined to the primary circuit and cannot be closed, an excess pressure will build up in the reactor building. To reduce pressure the primary coolant will be lead to the environment via relief channels and the ventilation stack. Under normal operation conditions the relief channels are isolated from the ventilation stack by ventilation valves, which open at a pressure of 1.1 bar and close automatically after pressure balance will have been established. Additionally each relief channel is equipped with an isolation valve, which is open under normal operation conditions and will be closed by manual initiation after a pressure relief accident, if the automatic relief valve does not close. Thus a controlled air flow within the reactor building can be maintained by means of the subatmospheric pressure ventilation system. The applicants are planning to operate this ventilation system and its filter equipment also during heat-up of the reactor core after the rupture of a 65-mm-pipe to minimize radioactivity discharge to the environment.

For justification of the safety enclosure concept the applicants state that due to the favourable activity retention properties of the fuel, core and reactor design as well as utilization of helium as coolant no specific requirements concerning the safety enclosure appear to be necessary. They emphasize that the results of their radiation exposure calculations demonstrate that even an unfiltered ventilation of the reactor building during core heat-up after rupture of a 65-mm-pipe will not cause excess doses above the dose limits according to § 28 sec. 3 of the Radiological Protection Ordinance.

We agree to the applicants that the design of the safety enclosure should take into account the favourable safety properties of small high temperature reactors. The most important safety feature in assessing the safety enclosure of the modular HTR-2 NPP is the low activity release from the fuel under normal operation conditions and during accidents. With reference to the results of our safety assessment we confirm that the radiation exposure in the environment after accidents with radioactivity discharge to the environment will be below the dose limits according to § 28 sec. 3 of the Radiological Protection Ordinance, even if the effect of the planned filters is not taken into account. Thus it can be justified to abandon a pressure-resistant containment; therefore, filtering of the activity released during core heat-up will only be important with respect to the minimization principle of the Radiological Protection Ordinance.

The reactor building is designed for the pressure loads accompanying the postulated events. The rate of flow of the subatmospheric pressure ventilation system is designed to maintain the required depression as compared to the environment after pressure balance has been established, taking into account the specified reactor building leakage. As a prerequisite to minimize the radiation exposure in the environment by controlled activity discharge pressure relief must occur reliably in accidents with excess pressures in the reactor building. Otherwise the activity would be discharged near the ground passing through building leakages in such an event. According to present design the pressure relief valves will open in all accidents causing a pressure of at least 1.1 bar in the reactor building. However, in our view pressure discharge acciddents due to medium-size leakages appear to be possible, where pressure in the building will be below the initiation limit of the relief valves. For this reason we see the necessity to analyse this class of incidents in system detail planning and to demonstrate which measures will be taken to prevent radioactivity discharge near the ground.

2.2.4 Electrical Installations

Under normal operation conditions the electrical installations of the modular HTR-2 NPP have to supply the electrical equipment of the systems as well as the instrumentation and control devices with power and to lead the electrical power generated by the plant to the high voltage grid.

In the course of accidents the electrical installations have to supply the safety installations with power required for accident management. For this reason the electrical equipment of the safety installations is connected to the emergency power supply system, which consists of two separated trains as the cooling systems. Apart from this the applicants are planning a further one-train emergency power supply system in the emergency control room of the reactor building.

The components connected to the emergency power supply system will be supplied with power predominantly by the auxiliary power system of the plant. After failure of the auxiliary power system two emergency diesel engines, each of them assigned to one train, are available for power supply. Each of the diesel engines is designed to supply the electrical components required for accident managment with power.

The design of the emergency power supply of the modular HTR-2 NPP complies with the requirement that after failure of the auxiliary power supply and simultaneous non-availability of both emergency diesel engines the plant must be kept within its design limits for up to 15 hours. The safety-relevant instrumentation and control equipment and the control room will de supplied with power by the batteries of the 220-V-DC system for up to 2 hours. The measuring devices and other electrical components connected to the emergency control room will be supplied by the 24-V battery installed in the emergency control room for up to 15 hours.

The applicants classify an even longer grid failure and simultaneous non-availability of both diesel engines as a hypothetic event due to its low probability to occur. However, they will equip the plant with an external connection for power supply of the emergency control room to reduce the remaining risk in this event beyond design basis.

Accident management for at least 15 hours to keep the plant within its design limits in the case of a complete failure of all active safety installations - even neglecting possible power supply by the emergency diesel engines - will be sufficient to cover the longest breakdown durations of the public grid.

In our view this safety property of the modular HTR-2 NPP is the relevant feature to justify an emergency power supply consisting of only two diesel engines. From a deterministic point of view a two-train emergency power supply would be considered to fail taking into account the single failure criterion and a potential repair fault. With respect to the well-known breakdown durations typical for the public grid the aforementioned combination of events is not safety-relevant for the modular HTR-2 NPP due to its response to power supply failures.

In our opinion, however, an estimated very low probability of incidents with grid failure times of more than 15 hours does not justify to consider them as exclusively hypothetical events. From a probabilistic point of view we conclude that the emergency power supply must be available with sufficient reliability not later than 15 hours after begin of the event to ensure compatibility with overall plant design. This means for the diesel engines and other required equipment that their reliability required for accident management is not determined by the need for an immediate availability, but that they can be repaired or taken into operation within a relatively large time interval. As a prerequisite to proceed as described, we see, however, the need to observe the relevant quality assurance requirements of the KTA-rules in planning and inspecting the emergency diesel engines and the emergency power distribution gear.

If the aforementioned aspects are considered sufficiently and correctly, we will agree that the long-term failure of the emergency power supply due to the simultaneous occurence of the incidents "grid failure" and "long-term unavailability of the emergency power supply" can be classified as a hypothetical event.

2.2.5 Reactor Protection System and Emergency Control Room

The control room is located in the switch gear and emergency supply building and contains the operation, information and signaling equipment required for operation and surveillance of the plant. After failure of the control room surveillance of the plant will be monitored in the emergency control room located in the reactor building, which is designed for all external incidents. In the emergency control room all data are signaled and recorded, which characterize the safety state of the plant and provide the information required for the necessary steps in accident management. Further radiological and meteorological data are signaled to indicate a possible radioactivity discharge to the environment and to determine the activity propagation conditions.

The emergency control room can be entered through a separate external entrance, if access from the switch gear and emergency supply building is impossible.

Apart from the small-sphere shut-down system no further system can be initiated from the emergency control room. If the control room is not available, the need for possible manual actions can be recognized evaluating the data signaled in the emergency control room. The necessary actions can be initiated locally by means of control devices.

The emergency control room including the measurement equipment is supplied with electrical power by a one-train emergency power grid. To ensure the required power supply after a possible failure of the emergency power supply the emergency control room is equipped with a battery designed for operation up to 15 hours. After this time interval power supply can be maintained by a mobile emergency power generator using a planned cable connection, until grid power supply is reestablished. It is assumed that the mobile emergency power generator can be supplied by an external organization as the fire brigade.

The initiation of the small-sphere shut-down system being the only active measure to be carried out from the emergency control room results from the safety concept of the modular HTR-2 NPP: Due to the inherent safety features of the reactor automatic actions are not required for accident management, after the reactor protection measures have been initiated at the beginning of the incident. Even after an external event destroying the switch gear and emergency supply building partly or completely the initiation of the small-sphere shut-down system is not required for immediate accident management, but for transition of the reactor from the state "hot, below criticality" to the long-term safe state "cold, below criticality".

Although according to the present status of design further actions initiated in the emergency control room are not required, we will not exclude the need for such actions emerging in the course of detailed planning. These possible necesseties would not affect the overall concept.

The reactor protection system is part of the instrumentation equipment and belongs to the safety system of a reactor; it is designed to initiate automatic actions in the course of accidents. These actions shall ensure that the dominant protective requirements defined in the safety criteria for nuclear power plants are met and that the plant will be kept within its design limits, until manual actions will facilitate long-term accident management. For the modular HTR-2 NPP this overall task reduces to initiate the planned protective actions only once after the accident has been detected. Active measures aiming at system control during the course of accident, e.g. to ensure decay heat removal or to perform level control in coolant vessels, are not required.

Within its task the reactor protection system has to determine specific safety data, to link safety parameters deduced from these data and to develop initiation signals, which have priority over any other control signal. The accident-specific data, the initiation criteria based on them and the initiation signals developed by the reactor protection system are based on the results of accident analyses.

The reactor protection system determines the safety data

- neutron flux,
- hot gas temperature,
- cold gas temperature,
- moisture content in the primary circuit,
- pressure in the primary circuit,
- pressure in the secondary circuit,
- primary coolant flow,
- feed water flow,
- steam flow.

They are applied either directly or after linking them in calculation circuits to gain safety parameters, from which initiation criteria are developed after well-defined limits have been exceeded.

All initiation criteria will trigger the following protective actions:

- Insertion of the reflector rods,
- shut-off of the primary circuit blower,
- isolation of the steam generator.

Further protective actions depending on kind and course of accident are

- primary circuit isolation,
- steam generator relief.

The steam generator relief valves are closed without being triggered by the reactor protection system after pressure balance has established between primary and secondary circuit.

Normally the accidents are detected by determining at least two physically diversified safety data. If only one of them is available, the KTA-rules require the initiation level to be designed more sophisticatedly. The applicants are planning such a design for detection of steam generator heating tube leakages and steam duct ruptures.

The planned redundant arrangement, the local separation and the constructional details of the equipment to initiate protective actions will ensure sufficiently that even in the case of failure-initiating incidents within the plant or system as well as external events the necessary protective actions to control an accident will be initiated reliably.

The constructional details of the reactor protection equipment and the local separation of redundant components to turn off the primary circuit blower and to insert the reflector rods are of dominant importance with respect to possible partial damage of the switch gear and emergency supply building due to aircraft impact or external shock wave. In our view the planned design - characterized by multiple initiation and initiation-directed failure behaviour of the logic module of the reactor protection system as well as of the switching equipment - is suitable to ensure the required initiation safety even in the case of partial damage of the switch gear and emergency supply building.

The three-train reactor protection system is connected to the two-train power supply system. This means that already the simultaneous occurence of a single failure-initiating incident and an accidental failure can interrupt the power supply of the complete reactor protection system. Due to the initiation-directed failure behaviour of the reactor protection system all reactor protection actions will be initiated, i.e. the reactor will be shut down safely.

2.2.6 Buildings

In our safety assessment of the buildings for the modular HTR-2 NPP we have verified, if the safety requirements are met, which are defined e.g. in the safety criteria for nuclear power plants and the RSK-guidelines and exceed those valid for conventional buildings. Specific requirements apply to buildings, which are necessary either - e.g. due to their constructional design, shielding or barriers - directly or - e.g. due to installation and load design of safety-relevant systems - indirectly

- to shut down the reactor safely and to keep it in the shut-down state,
- to remove the decay heat and
- to ensure safe enclosure and shielding of the radioactive inventory.

The following buildings are classified as safety-relevant:

- Reactor building (UJA),
- reactor building annex (UJH),
- reactor auxiliary building (UKA),
- switch gear and emergency supply building,
- secured cooling cells (URB),
- cable tunnels from UBR to UJA.

We have verified, if these buildings are designed sufficiently to bear the loads resulting from normal operation of the modular HTR-2 NPP and from accidents.

Based on our safety assessment we have recommended inter alia that the ventilation stack to be erected on the reactor auxiliary building should be designed for seismic loads, provided that subsequent damaging of the latter building endangering its stability required for accident management cannot be excluded. We have already discussed our results concerning the reactor building as part of the safety enclosure elsewhere.

Further we have verified, if the arrangement of the buildings described in the safety analysis report is such that damage can be excluded due to mutual impact, e.g. caused by assumed failure of high-energy components, as turbine or vessels with high energy content, or due to fragments resulting from collapsing buildings during an earthquake. The planned arrangement of the buildings will meet the aforementioned requirement. However, the effect of possible mutual impact will have to be verified again, if the exemplary arrangement of the buildings described in the safety analysis report is modified.

3 Accident Analysis

3.1 Spectrum of Incidents

According to § 7 sec. 2 of the German Nuclear Energy Act the necessary precautions against damage from construction and operation of nuclear facilities have to be taken with respect to the state of science and technology. Especially the activity discharge to the environment must not cause radiation doses exceeding the dose limits given in § 28 sec. 3 of the Radiological Protection Ordinance.

To demonstrate that this licensing prerequisite is met the applicants have elaborated a compilation of the HTR modular reactor specific design basis accidents by applying analogically the basic principles of the so-called "Accident Guidelines" /5/ developed for pressurized water reactors. According to the procedures defined in the accident guidelines they have analysed those accidents relevant for the plant, developed design requirements based on the results of the analyses concerning buildings, components and systems and defined radiologically relevant accidents to be analysed with reference to the dose limits given in § 28 sec. 3 of the Radiological Protection Ordinance.

They further have evaluated accidents beyond design basis to illustrate the inherent safety margins of the plant for this class of incidents and to demonstrate that risk-reducing measures have been taken to the required extent.

Due to our safety assessment the applicants have modified and completed their original accident compilation, taking into account these modifications in their revised safety analysis report. In our view the present compilation is complete and defines the design basis of the modular HTR-2 NPP with respect to the German Nuclear Energy Act and the presently common licensing procedures accurately and to the necessary extent.

3.2 Analysis of Accident Progress

Apart from verifying the completeness of the accident-initiating incidents analysed by the applicants we assessed accident progress for the different incidents and evaluated the consequences resulting from the analyses.

In our assessment we applied the deterministic requirements defined in the applicable rules and guidelines or adapted to the features of the modular HTR-2 NPPP, respectively. Therefore the originally submitted accident analysis had to be revised.

In their review the applicants have based their accident analysis on the usual unfavourable assumptions typical for nuclear licensing. These assumptions are e.g.

- postulated failure of the first initiating criterion for the reactor protection system,
- unfavourable initial conditions in the system under consideration,
- single failure and - when applicable - non-availability due to maintenance of the systems required for accident management,
- neglecting operational systems for immediate accident management.

This modified procedure caused the applicants to provide for additional safety installations, e.g. a limitation of rod insertion, and additional initiation criteria for the reactor protection system, e.g. the maximum steam temperature, as well as to modify safety-relevant limits, e.g. fuel design temperature. Consequently, the applicants had to revise existing analyses or to submit new ones.

In the following sections we will summmarize some important results of the accident analyses.

3.2.1 Reactivity Accidents

The applicants have investigated reactivity accidents initiated during normal operation of the equilibrium core and performed parameter variations to analyse the influence of unfavourable operation modes. They have taken into account accident-aggravating single failures, as e.g. erroneous start-up of the primary circuit blower and maloperation of the small-sphere shut-down system.

Our investigations concerning first reactor core, zero power operation and the start-up incident as well as malfunction of single reflector rods confirm that the accident "withdrawal of all reflector rods at maximum rate during full power operation of the equilibrium core" is to be considered as a covering design basis accident.

In this accident reactor scram required to limit the maximum temperatures of the pressure vessel unit is initiated by two physically different criteria. The reactor will be shut down safely despite the postulated failure of the first initiating criterion, even under the aggravating assumption that one reflector rod will not drop. Temperature design limits will not be exceeded; unacceptable power excursions will not occur.

Apart from the effects of absorber withdrawal we have analysed the reactivity-influencing incidents

- erroneous start-up of the primary circuit blower,
- incident-induced reduction of the cold-gas temperature,
- water leakages to the primary circuit,
- densification of the pebble bed due to earthquakes.

Also in these cases our results confirm that the possible effects are covered by the design basis accident investigated by the applicants. However, in our safety assessment report we have made some hints and given several recommendations concerning different limiting conditions to be established, as e.g. maximum power limitation to 105 % of rated power and avoidance of an unplanned start-up ot the primary circuit blower at high coolant gas temperatures, which will result in the reactor state "hot, below criticality" and simultaneous removal of the decay heat via the surface cooler. We further have pointed out in our report that additional investigations are required concerning the earthquake-induced densification of the pebble bed, where transfer of experiments carried out for the THTR reactor to the modular HTR-2 NPP is of special interest. The HTR-module core differs from the experimental set-up of the THTR model with respect to core geometry and construction of the core components; this can amplify the induced oscillations and prolong their duration, the resulting effect has to be investigated both analytically and experimentally.

3.2.2 Disturbed Heat Removal Without Coolant Loss

Disturbed heat removal without coolant loss will be caused by the incidents

- interruption of primary coolant flow,
- disturbed steam removal,
- disturbed feed water supply,
- failure of auxiliary power supply.

These incidents will cause deviations of thermodynamic parameters from their normal operation values. The deviations will be indicated by the limitation eqipment and reactor protection system, which will initiate the necessary protective actions.

We want to point out that the applicants have based the HTR modul design on the assumption that electrical power supply will fail totally in these incidents, i.e. external grid supply and auxiliary power supply will fail and emergency power supply will not be available. These assumptions deviate markedly from presently common procedures in nuclear licensing.

The applicants distinguish the following scenarios:

- Short-term failure of the auxiliary power supply system (less than or up to 2 hours),
- medium-term failure of the auxiliary power system (less than or up to 15 hours),

- long-term failure of the auxiliary power supply system (more than 15 hours).

In the case of short-term failure the plant is shut down as in normal operation by means of the main heat sink after power supply from the grid has been reestablished. Design limits will not be violated.

During a medium-term failure the plant is in a safe state, i.e. the reflector rods are inserted, the primary circuit blower is turned off and the steam generator is isolated. Two hours after the initiating incident the control room is not available as its power supply is interrupted. Due to the initiation-directed failure behaviour of the reactor protection system the primary circuit is locked and the steam generator is relieved. During core heat-up the primary circuit safety valve can open. After auxiliary power supply has been reestablished within 15 hours after the initiating incident the surface cooler can be operated again and the plant is shut down.

This incident was analysed extensively. The results indicate that locally the temperature of the primary cavity concrete in the reactor building is at its design limit of 150 °C after 15 hours. Within this time interval the maximum temperature of the reactor pressure vessel amounts to 310 °C; that of the surface cooler reaches 220 °C, i.e. it can be operated after reestablishing power supply. The maximum fuel temperature does not depend on operation of the surface cooler, it will be below 1200 °C.

Our results indicate that even a complete failure of the surface cooler with the reactor kept in its pressurized state for 15 hours will not cause unacceptable loads of the reactor pressure vessel including its components, of the reactor building and of the surface cooler. The radiation exposure in the environment due to discharge of primary coolant will be far below the dose limits according to § 28 sec. 3 of the Radiological Protection Ordinance.

The applicants do not take into account the complete failure of power supply for more than 15 hours in their design basis analyses. In our view this can only be accepted with respect to the two-train design of the emergency power supply, if a short-term repair of at least one emergency power generator can be performed reliably. We have defined the respective requirements in our safety assessment report, including a demand for adequate quality assurance measures.

3.2.3 Loss-of-Coolant Accidents

In the following we distinguish between ruptures and leakages in the primary and secondary circuits as well as ruptures of steam generator heating tubes with special emphasis on their effects concerning

- maximum fuel and component temperatures,
- loads, e.g. differential pressure loads on the reactor building,
- radiation exposure.

As ruptures are excluded for the pressure vessel unit itself primary ruptures are to be assumed only for pipes connected to the pressure vessel unit. These pipes have either a maximum diameter of 65 mm or the potential leakage areas are reduced to a corresponding cross-section by constructional measures as in the case of the fuel element discharge pipe.

The rupture exclusion for the pressure vessel unit can be ensured largely by observing the principles defined in the frame specification "Base Safety" /6/, which have been applied successfully in light water reactor technology. In our view additional measures are required with respect to the high-temperature materials and their weldings close to the steam duct socket and to material embrittlement due to neutron irradiation at comparatively low operation temperatures. In our safety assessment report we have accounted for the need of an operation-preceeding investigation program and of operation-accompanying irradiation samples. We confirm on the basis of our fracture-mechanical analyses that ruptures of the pressure vessel unit can be excluded. However, a constructional modification of the steam pipe socket turned out to be necessary, as a planned welding between two different materials was identified as critical and therefore was shifted from the high-temperature area. Thus we confirm the applicants' rupture assumptions for the pressure vessel unit (see sec. 2.2.1).

Primary ruptures and leakages must be analysed in different groups:

- Ruptures of large connecting pipes,
- ruptures of small pipes or small leakages, respectively,
- ruptures or leakages of primary coolant containing pipes outside the reactor building.

The double-sided rupture of the largest connecting pipe with a diameter of 65 mm containing primary coolant, which cannot be closed due to its position close to the reactor pressure vessel, proved to cover the effects of all other possible primary ruptures.

We have verified by independently performed calculations taking into account unfavourable initial conditions and both systematic and statistical uncertainties of the relevant parameters that the acceptable maximum fuel temperature will not be exceeded, i.e. the fuel temperature will be below design limit.

Further our calculations indicated that the isolation in the lower part of the reactor pressure vessel must be modified to avoid excess temperatures above design limit taking into account uncertainties of the temperature-influencing parameters. The design temperature of the reflector rods amounting to 650 °C will be exceeded markedly, although this will not affect their function and integrity. We consider this result not to be relevant for the overall design.

The radiation exposure in the environment caused by the pressure relief accident - even taking into account the subsequent core heat-up and a postulated failure of the filtering system - is far below the dose limits according to § 28 sec. 3 of the Radiological Protection Ordinance.

In our safety assessment report we have defined requirements concerning the automatic filtering of the activity discharges due to small leakages and we have justified in detail the need for further investigations concerning medium-size leakages, which will increase the pressure inside the building to values below the initiation level of the pressure relief valves. Again, these results are not relevant for the overall design.

As a result of our investigations concerning secondary ruptures and leakages, e.g. ruptures of the feed water or steam ducts, the concept to interrupt steam removal had to be modified to keep the pressure inside the reactor building for a two-module plant within the planned design limits. We have calculated independently from the applicants pressure increases for each of two rupture positions covering all possible primary and secondary ruptures. Our results confirm the design excess pressure of the reactor building amounting to 0.3 bar. However, this value is valid for the reactor hall and the largest part of the reactor building outer walls only. Parts of the building close to the rupture location will be charged with distinctly higher differential pressures, which have to be taken into account as special loads and combined with other loads in static design.

Our analyses of water leakages into the primary circuit due to steam generator heating tube leakages indicated that design and operational measures will ensure that the leakage size will be limited to the cross section of a single heating pipe. Thus we confirm the water inventory of 600 kg to be conservative, which the applicants have determined to penetrate into the primary circuit. The initiation of protective actions by only a single initiation criterion developed from the moisture measurement can be accepted, if that part of the reactor protection system relevant for this classs of incidents will be designed sophisticatedly according to the KTA-rules. Further we have investigated the influence of steam ingressed into the primary circuit on reactivity behaviour of the core and corrosion of the fuel elements accompanied by formation of water gas. Based on our results we confirm that safety-relevant limiting values are not exceeded.

3.2.4 External Events

We have investigated the influence of external events on safety of the modular HTR-2 NPP according to the usual methodology in nuclear licensing. Based on our results we have demanded that the extent of plant components and buildings designed for earthquake had to be enlarged. The applicants are planning to generate the basic seismic load assumptions by means of a new, empiric-statistical method, which aims at deducing the seismic-structural design data from a probabilistic analysis of seismic hazard. We have verified that the planned procedure is a consistent method providing realistic seismic load data. However, we have recommended that for a concrete site of the modular HTR-2 NPP the seismic structural design data should be generated additionally by means of the proven deterministic method.

Our analysis of further natural external events did not result in any requirements to modify plant design.

3.3 Risk-Reducing Measures

The events "aircraft impact" and "external shock wave" are characterized by an extremely low probability to occur. For this reason their possible effects are not classified as design basis accidents in the accident guideline for pressurized water reactors. Thus the design characteristics to be met for these events aim at reducing the risk due to operation of the plant.

According to present plant design the reactor building and safety-relevant components and systems within this building shall be designed for loads from aircraft impact and an external shock wave. In compliance with the RSK guideline /6/ the loads due to an aircraft impact are assumed to be independent from the site.

The design of the reactor building and the safety-relevant components and systems for an external shock wave is based on the standard pressure-time graph given in the BMI guideline for design of nuclear power plants against shock waves from chemical reactions /7/. If due to site-specific features, e.g. industrial plants in close neighbourhood, shock waves inducing higher loads appear to be possible, design will be based on a site-specific pressure-time graph of the shock wave.

An aircraft impact as well as a shock wave will destruct partially or completely the switch gear and emergency supply building. This can affect the function of the reactor protection system and emergency power supply system to such an extent that both systems will fail.

The applicants are planning to design the reactor protection system such that the protective actions

- insertion of the reflector rods,
- turn-off of the primary circuit blower,
- isolation of the secondary circuit,
- isolation of the primary circuit,
- steam generator relief

will be initiated when necessary due to plant behaviour or as a result of damage to the reactor protection system itself.

Surveillance of the plant and long-term control of subcriticality will be performed in the emergency control room, which is located in the reactor building designed for the aforementioned events. The emergency control room is unrestrictedly accessible and has a separate external entrance.

The emergency control room is equipped with a battery-supported one-train emergency supply system to ensure its function and to supply peripheric systems with power (ventilation, lighting and communication systems). The battery is designed for operation up to 15 hours. After this time interval power supply will be maintained by a mobile energy supply engine using a special cable connection until grid power supply will be reestablished; the energy supply engine will be provided by the fire brigade or another institution. When necessary, two sections of the surface cooler can be operated by connecting them to fire hoses joined externally to the intermediate cooling system.

The possible interruption of decay heat removal for at least 15 hours in conjunction with the independent emergency power supply of the emergency control room grant a sufficient time interval to perform measures to reestablish heat removal by means of the surface cooler.

Thus we confirm on the basis of our design review that the extent of risk-reducing measures for the modular HTR-2 NPP by protecting it against civilization-induced external events and by providing steps to establish an external feedwater supply of the surface coolers as well as power supply of the emergency control room meets the applicable requirements and takes into account the plant-specific properties sufficiently.

4 Radioactive Substances and Radiation Protection Measures

4.1 Radiation Protection and Radiation Exposure of the Personnel

The assessment of plant design with respect to radiation protection of the personnel shall ensure that planning measures have been taken

- to avoid any unnecessary radiation exposure and contamination and
- to keep any unavoidable radiation exposure or contamination not only below the limiting values given in the Radiological Protection Ordinance, but as low as possible

according to the requirements of the Radiological Protection Ordinance.

There will be different radiation sources in the modular HTR-2 NPP: The prompt nuclear radiation and the predomimnant part of fission product radiation stem from the nuclear processes in the reactor core. The γ-radiation due to electron capture and the activated nuclides will also be generated by neutron radiation outside the reactor core. Finally, activated nuclides and a small part of the fission products can be transported by the coolant to systems and components adjoined to the primary circle.

We have verified, if the applicants have taken into account the relevant radiation sources in plant design and if the source terms can be applied to generate realistic radiation fields and levels to be expected in the plant. Based on our results we confirm that the applicants' methodology is suitable to generate the required data and the information given in the safety analysis report and further design documents is consistent and complete.

The radiation field of the reactor core will be shielded in axial and radial direction by different components, which as a whole form a sandwiched structure. The inner layer will mainly consist of the graphite of the fuel elements, the borated carbon material parts and the reactor pressure vessel. The outer layers will be made from concrete and form the biological shield. Apart from the reactor core especially the following components and systems have to be shielded due to their activity inventory:

- Fuel handling equipment,
- helium purification plant,
- liquid waste treatment equipment,
- storage facility for radioactive waste.

The basic material of all shielding walls is concrete. Further shielding materials are steel and lead, e.g. for shielded doors or local shieldings, as well as special concrete at all places, where due to lack of space standard concrete would be insufficient as shielding material.

The modular HTR-2 NPP does not diiffer from other nuclear power plants with respect to the basic design principles of shielding. The planned shielding concept and the radiological protection measures described in the following are suitable to protect the personnel against the hazards of radioactive radiation according to the state of science and technology. If local measurements indicate the need for additional shielding measures, the space to install further shielding equipment appears to be sufficient. The applicants have to plan the radiation shielding measures in detail within constructional design of buildings and components.

Apart from the shielding concept further measures are planned to protect the personnel of the modular HTR-2 NPP against the hazards of radioactive radiation:

- Separation of nuclear and conventional components and systems,
- separation of high- and low-active components in nuclear systems,
- separation of individual high-active components with respect to maintenance and repair,
- local separation of components, valves and operating stations on the one hand and of internal passages for the personnel on the other hand.

The planned arrangement of rooms and components will contribute extensively to reduce the radiation exposure of the personnel. Examples are:

- Both modular reactors will be erected separately in primary cavities,
- the pumps of the secured intermediate cooling system will be arranged in different rooms of the reactor building annex and will be shielded,
- the containers for concentrated liquid waste will be erected in separate, shielded rooms of the reactor auxiliary building.

The concept to mutually shield activity-containing components will be realized to a large extent in the controlled area. This does not apply to both waste water vessels, which will be arranged in a common room. We recommend to arrange these components in separate rooms.

According to the classification system of radiological protection areas described in the safety analysis report the reactor building and the predominant part of the reactor auxiliary building are part of the controlled area.

The rooms containing the steam generator and the reactor pressure vessel are inaccessible.

The applicants are intending to classify the rooms within the controlled area with respect to local dose rates. The upper limits of the individual classes roughly differ by a factor of ten. This classification system shall facilitate accessibility even to areas with high local dose rates by keeping the dose rates along the access passages at low values. Thus all rooms within the reactor building apart from the primary cavities will be accessible during operation.

As a result of the radiological protection measures planned by the applicants the dose rates outside the controlled area will be below dose limits applicable to operational surveillance areas and outside the power plant area below those for non-operational surveillance areas.

We agree to the planned concept to install distinguished radiation protection areas and to classify the rooms within the controlled area according to local dose rates; during future datail planning the different rooms can be assigned to radiological room classes. When necessary, further constructional measures to improve shielding will be possible. Our independently carried out calculations confirm the assignment of the operational and non-operational surveillance areas performed by the applicants. In our view there will be no need to install a non-operational surveillance area, if the plant area is separated from the neighbouring areas in an appropriate manner, i.e. the distance to the plant fence is sufficiently large.

In additionally submitted documents the applicants have described in detail several important maintenance activities and regular periodic inspections to demonstrate how the requirements of the guideline on precautionary measures in radiation protection /8/ will be met: The applicants are planning to apply special tools or measures to reduce the required time for maintenance or repair of primary circuit components or others in adjoining systems, e.g.

- camera assistance in preparing reactor pressure vessel inspections and preceding automized ultrasonic testing,
- application of easy-to-detach isolations,
- stud tensioning device for reactor pressure vessel closure head,
- mobile shielded working platform above the upper thermal shield.

The basic requirements concerning precautionary protective measures for regular periodic inspections as well as for maintenance and repair especially of the reactor pressure vessel are met by the intended measures. However, the detailed planning of the working procedures accompanying future design and construction steps must be based on a break-down of the collective doses with respect to separate contributions to individual doses and a detailed description of the working sequence taking into account local dose rates.

4.2 Discharge of Radioactive Substances During Normal Operation and Radiation Exposure in the Environment

The applicants have described the processes, the applied calculation models and the extent of activity release from the fuel elements to the primary coolant in the safety analysis report. Several phenomena contribute to activity release to the coolant:

- Activity release from intact particles,
- activity release from particles damaged during manufacture,
- activity release due to radiation-induced damage,
- activity release due to contamination of the graphite matrix.

The intact coatings of the particles will form an efficient barrier against activity release, i.e. intact particles will not contribute relevantly to release rates.

However, during manufacture sporadic damaging of single particles cannot be excluded completely, i.e. the relevant barriers for activity retention will not be effective. In these cases the activation and fission products will migrate from their origin in the fuel grain to the grain boundaries by diffusion, followed by grain boundary diffusion to the graphite matrix and finally to the fuel element surface.

Release of gaseous fission products will only be retarded by the slow diffusion process in the fuel grain. Grain boundary diffusion and diffusion in the graphite matrix are relatively fast processes. Diffusion of non-volatile fission and activation products both in the fuel kernels and in the graphite matrix is a relatively slow process with the effect to retard release of these nuclides and thus to reduce the release rates especially of the short-lived isotopes. Design calculations are based on an assumed part of defect particles (expected value), which doubles that verified in specific investigations.

In principle, additional particle defects, induced e.g by burn-up, fast neutron fluence or temperature loads, can occur during operation of the fuel elements in the reactor core. The applicants have deduced the expected part of irradiation-induced particle defects from irradiation experiments. Design is based on a defect rate assumed conservatively to be higher than the experimental results by a factor of ten.

The natural graphite contained in matrix graphite contains inter alia traces of uranium stemming from natural contamination. Thus, apart from activation of further contaminants, fission of uranium will occur to a minor extent in the graphite matrix outside the kernels. The resulting fission and activation products can migrate through the graphite matrix to the fuel element surface by means of the aforementioned transportation mechanism. The applicants are combining the inventories of uranium due to matrix contamination and to fabrication-induced particle defects to a so-called "free uranium inventory" to specify the acceptable maximum uranium content in the matrix and to calculate the release rates. The design value for the fabrication-induced particle defects includes the uranium contamination of the graphite matrix.

In the following we summarize the results of our evaluation concerning the release rates calculated by the applicants:

- The break-down of the source terms into fabrication-induced and irradiation-induced particle defects as well as into matrix contamination and the different release models deduced from these mechanisms appear to be reasonable,
- the physical models applied in the calculations describe the relevant transportation phenomena of fission and activation products and take into account the state of science and technology,
- the input data, e.g. particle defect rates, diffusion coefficients and other material data, appear to be sufficiently conservative for the calculations,
- the nuclide vector has been determined with respect to the radiologically relevant isotopes.

Our calculations of the activity inventory based on conservatively estimated release rates indicate for some nuclides distinctly lower release rates as compared to the applicants' results, whereas we confirm the release rates calculated for the remaining nuclides. Thus we expect that the release rates calculated by the applicants will cover those to be expected during normal operation.

The radioactive nuclides released to the coolant or generated there by activation, respectively, are transported by the coolant from the core to the primary circuit. In normal operation a quasi-steady-state coolant activity will result, which can be calculated performing a balance of all source and loss terms.

We have already dealt with the most relevant source term due to activity release from the fuel elements. To a minor extent aerosole nuclides stemming from radioactive decay of the short-lived noble gases contribute to the coolant activity, too.

The relevant loss terms are governed by radioactive decay, the filtering effect of the helium purification system and losses via primary circuit leakages. Additionally plate-out of radionuclides on the inner surface of the primary circuit will reduce the aerosole activity in the coolant.

In our safety assessment we have verified the design data of the primary coolant activity given by the applicants and performed some independent calculations. The nuclide-specific results are listed in our safety assessment report and confirm the design data.

Further we have determined and evaluated the activity inventories in the auxiliary systems, e.g. in the helium purification system, starting from the primary coolant activity. The results have been presented in our safety assessment report. In the following we are giving some comments concerning the possibility to operate the secondary circuit in an "unclosed" mode, i.e. to use part of the steam for process purposes.

The applicants have applied for a tritium concentration in the process steam, which meets the requirements of § 4 sec. 2 of the Radiological Protection Ordinance. In conjunction with attachment III sec. 2 of the Radiological Protection Ordinance radioactive substances can be applied provided that their specific activity is less than 100 Bq/g.

As the applicants are planning to use process steam outside the power plant and thus outside the surveillance area, the condensate of the process steam must be reusable without any restriction. If in future licensing steps no annual limit for the yearly discharge of tritium with process steam is established, the requirements of § 46 sec. 4 will have to be met, according to which process steam condensate can only be drained off to public sewers or rivers, lakes, and canals, if the waste water activity is not greater than 1.25 times the value given in attachment IV, tab. IV 1, column 6 of the Radiological Protection Ordinance.

If the use of condensate as drinking water cannot be excluded, the acceptable maximum tritium uptake per year according to § 46 will be exceeded slightly on the basis of the planned tritium concentration. This aspect has to be clarified before the corresponding licence is issued.

The applicants are planning to verify by sampling that the discharges to the process steam are kept below the limits given in the licence. If sampling is sufficient or a continuous surveillance is required, depends on the planned use of the process steam and has to be decided during future licensing steps.

For verification of the room activity concentrations in the reactor cavities we have investigated the mechanisms

- activation of the air close to the reactor pressure vessel,
- activation of the metallic surfaces of the reactor pressure vessel and the surface coolers,
- primary coolant leakages.

We confirm the applicants' data on these mechanisms to be conservative.

Further we have verified the activity discharges during normal operation of the modular HTR-2 NPP, based on the acitivity concentrations in systems and rooms, and listed the results in our safety assessment report. Our nuclide-specific results refer to discharges with

- exhaust air from the ventilation stack and the turbine building,
- waste water from the liquid waste treatment system and from secondary circuit leakages as well as
- solid radioactive waste, as e.g. spent fuel or normal operational waste.

These discharges of radioactive substances will cause a radiation exposure in the environment, which must not exceed the dose limits according to § 45 of the Radiological Protection Ordinance. In licensing this radiation exposure has to be calculated with respect to the most unfavourable receiving points taking into account all relevant exposure pathways and including the food chains. The administrative guideline on the applicaton of § 45 of the Radiological Protection Ordinance /9/ contains calculational models and parameters to determine the radiation exposure.

According to this guideline the human radiation exposure is defined as the radiation exposure of a member of the critical population group and is caused by external irradiation, i.e. the contribution due to externally effective radiation sources, and internal irradiation, i.e. the contribution due to incorporated radionuclides. A member of the critical population group is a person, who is exposed to the maximum radiation exposure due to one or more exposure pathways at the most unfavourable receiving point. The radiation exposure shall be determined on the basis of realistic habits to be assumed for part of the population. Extreme habits, e.g. in food consumption, shall not be considered. The most unfavourable receiving point is that location in the environment of the modular HTR-2 NPP, which due to the distribution of radionuclides discharged to the environment will experience the highest radiation exposure. It is assumed that the consumed food is produced at the location characterized by the highest food contamination in the respective area.

In the safety analysis report and in additional documents the applicants have described their calculations and listed the results of the radiation exposure of adults and infants due to radioactive effluents from the modular HTR-2 NPP with the exhaust air and waste water. Their calculations are based on the discharge data submitted in their application and on the procedures defined in the aforementioned guideline, taking into account all exposure pathways and conservative transfer data.

We have determined in an analogous procedure the radiation exposure caused by radioactive effluents from the modular HTR-2 NPP with the exhaust air and waste water and applied the calculational models and data given in the guideline on the applicaton of § 45 of the Radiological Protection Ordinance. Whenever necessary, i.e. in cases lacking site-specific data, e.g. meteorological data, we have used the same data as the applicants, which are listed in the safety analysis report. Before applying these data, we have verified, if they are realistic and can be taken as representative for a possible site.

Our calculation of the radiation exposure is based on the discharge data determined in our safety assessment instead of the activity discharge data the applicants have applied for. Thus our results of the radiation exposure are below those of the applicants. However, we confirm that the radiation exposure due to radioactive effluents both with the exhaust air and waste water will be distinctly below the dose limits according to § 45 of the Radiological Protection Ordinance.

We have compiled the radiation exposure determined by us in relation to the dose limits according to § 45 of the Radiological Protection Ordinance in tables 4-1 and 4-2. Deviations from the results of our safety assessment report (October 1989) are caused by a recent review of the Radiological Protection Ordinance and the calculational procedures. However, these deviations are of minor importance only.

The results are distinctly below the dose limits according to § 45 of the Radiological Protection Ordinance. However, for a concrete site the additional radiation exposure due to other nuclear facilities or handling of radioactive substances has to be taken into account. As a concrete site for a modular HTR-2 NPP has not been proposed these contributions to radiation exposure are not considered in the aforementioned tables.

After a concrete site has been chosen, it might become necessary to modify the obtained data, as site-specific features can influence the calculated radiation exposure, as e.g. the flow conditions of the run-off water, meteorological, orographic or settlement conditions in the environment of the plant as well as specific food consumption habits of the population. Thus the calculation of the radiation exposure to be expected under normal operation conditions has to be repeated once more after a site has been determined.

As by now the calculated dose rates are distinctly below the dose limits according to § 45 of the Radiological Protection Ordinance, we expect the dose rates of a concrete site to be within these limits.

The radiation exposure due to direct radiation from the plant is negligible as compared to that due to radioactive effluents with the exhaust air or waste water. Thus it is distinctly below the dose limit according to § 44 of the Radiological Protection Ordinance.

Tissue	Annual dose in µSv		
	Adults	Infants	Dose limit § 45 RPO
Bladder	27	47	900
Breast	28	48	900
Upper large intestine	27	47	900
Lower large intestine	27	47	900
Small intestine	27	47	900
Brain	28	48	900
Skin	29	48	1800
Testes	28	47	300
Bone surfaces	28	48	1800
Liver	28	47	900
Lung	28	47	900
Stomach	28	47	900
Spleen	28	47	900
Adrenals	27	47	900
Kidneys	28	47	900
Ovaries	27	47	300
Pancreas	27	47	900
Red bone marrow	28	47	300
Thyroid	28	49	900
Thymus	28	47	900
Uterus	27	47	300
Effective dose equivalent	28	47	300

Table 4-1: Potential radiation exposure due to activity discharge with the exhaust air

Tissue	Annual dose in μSv		
	Adults	Infants	Dose limit § 45 RPO
Bladder	0.9	0.3	900
Breast	0.8	0.3	900
Upper large intestine	1.0	0.3	900
Lower large intestine	1.0	0.3	900
Small intestine	1.0	0.3	900
Brain	0.8	0.3	900
Skin	0.7	0.4	1800
Testes	0.9	0.3	300
Bone surfaces	0.9	0.3	1800
Liver	1.0	0.3	900
Lung	0.9	0.3	900
Stomach	0.9	0.3	900
Spleen	0.9	0.3	900
Adrenals	1.0	0.3	900
Kidneys	0.9	0.3	900
Ovaries	0.9	0.3	300
Pancreas	0.9	0.3	900
Red bone marrow	0.9	0.3	300
Thyroid	1.0	0.5	900
Thymus	0.8	0.3	900
Uterus	0.9	0.3	300
Effective dose equivalent	1.0	0.3	300

Table 4-2: Potential radiation exposure due to activity discharge with waste water

Ordinance. Thus the required protective measures against the effects of accidents have been taken by the applicants according to the state of the art.

Accident	Accident doses in µSv							
	Bone	Liver	Total body	Thyroid	Kidney	Lung	Gastro-intestinal tract	Skin
Rupture of pipe (Ø 65 mm) Pressure relief phase	57	12	16	58	11	11	15	7.3
Core heat-up (version 1) 0-34 h not considered	0.7	0.7	0.7	120	0.7	0.7	3.4	0.5
Core heat-up (version 2) time intervals 0-34 h, 34-42 h etc.	4.1	4.0	4.0	730	4.0	4.4	24	2.8
Rupture of measuring pipe	1.0	1.0	1.0	21	1.0	1.1	2.4	1.6
Rupture steam generator heating tube, close to preheater part	1.4	0.9	0.9	90	0.8	0.9	2.8	0.6
Rupture steam generator heating tube, close to superheater part	1.7	1.2	1.1	75	1.0	1.0	2.5	0.8
Rupture of pipe in helium purification plant	2.1	7.1	7.1	9.3	7.1	7.1	7.3	2.0
Leakage of waste water evaporator	57	50	46	45	45	42	46	41
Earthquake reactor auxiliary building	240	510	490	550	490	490	500	170
Core heat-up (version 2) - Distance: 2 km -	2.4	2.4	2.4	1200	2.4	2.6	25	0.4
Dose limits § 28 sec. 3	300000	150000	50000	150000	150000	150000	150000	300000

<u>Table 4-3:</u> Radiation exposure in the environment due to activity discharges in radiologically relevant accidents, receiving point distance: 100 m - Adults

Accident	Accident doses in µSv							
	Bone	Liver	Total body	Thyroid	Kidney	Lung	Gastro-intesti-nal tract	Skin
Rupture of pipe (∅ 65 mm) Pressure relief phase	56	9.5	13	280	8.0	7.6	8.1	7.3
Core heat-up (version 1) 0-34 h not considered	2.2	2.6	1.6	760	1.0	1.6	0.6	0.5
Core heat-up (version 2) time intervals 0-34 h, 34-42 h etc.	14	17	9.9	4700	6.1	9.9	3.8	2.8
Rupture of measuring pipe	1.3	1.5	1.2	120	1.2	1.2	1.1	1.6
Rupture steam generator heating tube, close to preheater part	2.5	2.3	1.5	560	1.1	1.5	0.8	0.6
Rupture steam generator heating tube, close to superheater part	2.6	2.4	1.6	450	1.2	1.5	1.0	0.8
Rupture of pipe in helium purification plant	1.5	3.6	3.5	16	3.5	3.5	3.5	2.0
Leakage of waste water evaporator	57	51	45	45	44	42	48	41
Earthquake reactor auxiliary building	230	300	280	600	280	280	280	170
Core heat-up (version 2) - Distance: 2 km -	19	2.2	13	8100	5.8	13	1.5	0.4
Dose limits § 28 sec. 3	300000	150000	50000	150000	150000	150000	150000	300000

Table 4-4: Radiation exposure in the environment due to activity discharges in radiologically relevant accidents, receiving point distance: 100 m - Infants

5 Summary

TÜV Hannover e.V. has performed an independent safety assessment concerning the design of a high temperature reactor plant with two modular reactors to generate simultaneously electrical power and process steam or heat for district heating, respectively.

The assessment was based on the evaluation criteria to be applied in licensing of nuclear facilities in Germany, as e.g. the KTA-rules, and on publications representing the present state of research in high temperature reactor technology. Exceptional design features due to the new concept of the modular high temperature reactor have been dealt with in detail and their safety relevance has been assessed.

The applicants have demonstrated successfully that the radiation exposure in the environment caused by the discharge of radioactive substances from the modular HTR-2 NPP during normal operation is far below the dose limits according to § 45 of the Radiological Protection Ordinance and that the measures for radiation protection of the personnel have been planned adequately.

An important prerequisite for our assessment was to elaborate a complete and representative catalogue of design basis accidents analogously to the accident guideline for pressurized water reactors. This task inter alia facilitated to distinguish between design basis accidents and accidents beyond design basis.

Based on the design documents prepared in 1989 by the Siemens AG/Interatom GmbH project team an extensive accident analysis was performed and design requirements were developed for all components and systems. We further verified if the planned design of the buildings, systems and components as well as the operation modes of the plant will meet these requirements. These investigations were performed considering the conservative assumptions typical for licensing procedures of nuclear facilities.

A further result of the accident analysis was the identification of radiologically representative accidents and a successful verification that after these accidents the dose limits given in § 28 sec. 3 of the Radiological Protection Ordinance are not exceeded and that the necessary protective measures against the hazards of nuclear technology have been planned according to the state of the art.

Based on our safety assessment we confirm that the design of the modular HTR-2 NPP meets the safety requirements to be imposed on nuclear facilities in Germany. Our safety assessment report summarizes the design requirements as "conditions" to be met in future detailed design.

Our investigations on risk-reducing measures indicate that the modular HTR-2 NPP has pronounced inherent safety properties, which govern the plant behaviour in incidents beyond design basis.

6 Documents and Literature

/1/ Siemens/Interatom
Modular High Temperature Power Plant
Safety Analysis Report
November 1988

/2/ TÜV Hannover e.V.
Safety Assessment Report of the Modular HTR-2 Power Plant
October 1989

/3/ Nuclear Energy Act in its Version of 23 December 1959 and
Amendments of 15 June 1985 and 18 February 1986

/4/ Radiological Protection Ordinance
Amendment of 30 June 1989

/5/ Federal Ministry of the Interior
Guideline for the Assessment of Pressurized Water Reactor Design against Accidents According to § 28, Sec. 3, of the Radiological Protection Ordinance
18 October 1983

/6/ Reactor Safety Commission (RSK)
RSK-Guidelines for Pressurized Water Reactors
3rd Edition, 14 October 1981, and Annex 2 to Chapter 4.2 - Frame Specification "Basis Safety of Pressurized Components"

/7/ Federal Ministry of the Interior
Guideline for the Protection of Nuclear Power Plants against Shock Waves from Chemical Reactions by Design with Respect to Stability and Induced Oscillations and by Safety Zones
22 September 1976

/8/ Federal Ministry of the Interior
Guideline for the Radiation Protection of the Personnel during Maintenance in Light Water Reactors: The Protective Measures to be Planned in Plant Design
August 1978

/9/ Federal Ministry of the Environment
General Administrative Regulation Concerning § 45 of the Radiological Protection Ordinance: Determination of the Radiation Exposure due to Discharge of Radioactive Substances from Nuclear Plants or Facilities
21 Feruary 1990

/10/ Regulation for the Calculation of Accident Consequences According to § 28, Sec. 3, of the Radiological Protection Ordinance to Assess the Design of Nuclear Power Plants with Pressurized Water Reactors
31 December 1983

Review of the safety concept of the HTR 2 module reactor plant

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Received 13 June 1990, revised version 19 October 1990

After discontinuation of the preliminary decision procedure in the Federal State of Lower Saxony (Federal Republic of Germany) the review of the safety concept of the HTR 2 module reactor plant was carried out by TÜV Hannover e.V. (Technical Inspection Agency of Hanover) within the framework of a research and development order awarded by the Federal Ministry of Research and Technology (Bundesminister für Forschung und Technologie). In the course of the evaluation, modifications to the plant concept have proved to be necessary which have been included in the revised planning documents. So additional safety facilities, such as control rod insertion limitation, additional release criteria for reactor protection, such as maximum mainsteam temperature, had to be included in the concept and limiting safety values, such as the maximum permissible fuel element temperature, had to be changed. As a summary it can be stated that the concept of the HTR 2 module plant meets all the severe requirements applicable in the Federal Republic of Germany and additionally the inherent safety characteristics provide enhanced safety margins. Another aim of our evaluation was to check where safety-related further developments might be necessary. According to this aspect our report of October 1989 contains a multitude of proposals and relevant information.

1. Definition and course of the evaluation

In April 1987 Siemens AG and Interatom GmbH filed an application with the Ministry of Environmental Protection (Ministerium für Umweltschutz) of Lower Saxony in compliance with section 7a of the Atomic Energy Act [1] for the submittal of a preliminary concept decision for an HTR 2 module reactor plant for combined generation of electrical energy and process steam, resp. district heating.

Shortly after the Technische Überwachungs-Verein Hannover e.V. was entrusted with the review of the safety concept within the framework of the licensing procedure through the Ministry of Environmental Protection of Lower Saxony. TÜV Rheinland e.V. was participated by means of a subcontracting order.

In a series of technical discussions which took place during the second half of 1987 we requested further documents apart from the Safety Analysis Report submitted as an application document, and a number of modifications to the safety requirements were discussed and substantiated.

Due to these proposals and the discussions which took place during the first half of 1988, the system planner revised the concept of the plant, and submitted supplementary technical documents. The Safety Analysis Report [2] was revised as well. The most important documents were available at the beginning of 1989.

In April 1989 the application for a preliminary decision of the concept was withdrawn, the licensing procedure was discontinued by the Ministry of Environmental Protection, and the order to TÜV Hannover was shelved.

The work to be carried out for the review of the safety concept had already progressed a great deal at this point in time. Therefore, TÜV Hanover offered the Federal Ministry for Research and Technology (BMFT) to prepare an evaluation of the safety concept without reference to a licensing procedure, and to draw up whether the concept of the HTR module is in compliance with the requirements for nuclear facilities applicable in the Federal Republic of Germany, and to

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establish where further developments might be necessary. This project is to safeguard that future research and development work on the HTR module is carried out on the basis of the requirements for nuclear safety to be adhered to in the Federal Republic of Germany.

Within the framework of the order awarded by the Federal Ministry for Research and Technology the report of approx. 900 pages on the review of the safety concept of the HTR 2 module reactor plant was rendered and presented under the date of October 1989 [3].

The following chapters will repeat the most essential statements of this report. Some examples will be used to demonstrate which modifications of the plant concept, and which proof to be furnished, result from that report.

2. Basis of evaluation and design requirements

The Atomic Energy Act and the Radiation Protection Ordinance [4] in the Federal Republic of Germany are the fundamental legal basis for planning, construction, operation, and shutdown as well as supply and disposal of nuclear facilities, and thus lay down also the framework for the safety review. They do not relate to plant concepts and detailed technical solutions. The requirements demanded have to be met at any rate.

Below statutory determinations there are a number of guide-lines and regulations which already have a strong reference to technical concepts, and which have been worked out parallel to the development of nuclear technology in order to support a standardization of the evaluation criteria, and to document the state of the technology. These criteria mainly refer to light-water reactors. Thus, the standards on nuclear facilities are largely characterised by the physical properties and plant features of the light-water reactors, and here especially by pressurized water reactors with a core design for high power density, which may not be transferred directly to other reactor concepts [e.g. 5].

Gas-cooled high-temperature reactors stand out especially for the distinctive characteristics of inherent safety against reactivity accidents and failures of the heat removal system. This fact is all the more true for small power output per reactor unit, and for low power density in the reactor core. Therefore, it would be inappropriate to transfer the requirements stipulated in the technical standards for light-water reactors to the HTR module without reflection, all the more as the safety characteristics of a gas-cooled high-temperature reactor with small power density were consistently used to develop a reactor concept with the aim of limiting the fuel element temperatures even in case of failure of all active cooling systems and the loss of coolant to such an extent that no considerable release of radioactive fission products from the fuel elements takes place. Such a requirement exceeds the standards set up to now. With respect to the applicability of the standards on nuclear facilities this fact means that even existing provisions for high-temperature reactors [e.g. 6] have to be adapted and transferred to the conditions of the HTR module reactor.

Apart from the acts and statutory instruments to be adhered to without respect to the concept, we have used the existing regulations and guide-lines for our concept review only to the extent we considered them applicable for the HTR module concept. On the basis of the specific safety features of the HTR module we have substantiated every deviation in our report. However, it was not the purpose of our project to check and graduate the applicability of all standards on nuclear facilities to the HTR module concept within the framework of the review of the safety concept.

The precautionary measures against damage according to the state of the art demanded by the Atomic Energy Act can only be safeguarded by additional recourse to the present state of research in case of a pilot plant with a novel design concept. Therefore, we have included corresponding publications in our considerations to a large extent.

3. Assumptions on the location

The concept planning of the HTR 2 module reactor plant was carried out independent of the site. Thus, the plant planner has set the design assumptions on the site features in such a way that they are applicable for a large group of potential sites, provided assumptions independent of the site must not be taken into consideration to comply with the rules and guide-lines.

For this reason, we have evaluated the data on the site features mentioned in the Safety Analysis Report only with respect to completeness and consistency in our analysis, and have checked whether the chosen procedures to determine the site data are in compliance with the standards on nuclear facilities in the Federal Republic of Germany. In doing so, we have only taken those features into consideration which according to our experience in assessing other nuclear plants could have an influence on the concept of the HTR module.

4.3 Radiation Exposure after Accidents

The spectrum of light water reactor accidents to be analyzed radiologically is defined in the accident guideline. According to this guideline the accident analyses have to be based on sufficiently conservative assumptions, calculational models and input parameters to describe the cours of the accident, radioactivity discharge to the environment and propagation of radioactive substances.

Due to the different properties of a pressurized water reactor and the modular HTR-2 NPP the design basis accidents ot these plants differ substantially (see sec. 3.1). We have verified by applying the accident guideline for light water reactors analogously, if in their analysis the applicants have selected representative accidents covering the radiological consequences of similarly occurring events. The applicants have examined the following scenarios:

- Leakage of a pipe between reactor pressure vessel and primary circuit isolation valve,
- leakage of a measurement pipe containing primary coolant,
- failure of a steam generator heating tube followed by long-term failure of water separation and primary circuit pressure control.

These analyses mainly aim at demonstrating that after release of radioactive substances to the reactor building the radiation exposure in the environment will be limited sufficiently.

Further the applicants have investigated as accidents with activity discharge outside the reactor building the

- failure of the largest pipe containing primary coolant outside the reactor building accompanied by isolation of the primary circuit,
- leakage of a vessel containing contaminated water.

The reactor auxiliary building is not designed for loads from an earthquake. For this reason

- component leakages of the helium purification system and of the evaporator concentrate vessel

have been investigated by the applicants, too.

In our safety assessment we have restricted ourselves to the accidents examined by the applicants, as these events are radiologically representative for the events to be considered and cover their radiological effects. In all other events less activity will be discharged to the environment, whereas the nuclide composition and the activity discharge mechanisms essentially remain unchanged.

We have verified the correctness of the applicants' results, the relevant input parameters, the applied calculational models and the assumed release and discharge conditions by performing independent calculations. In our safety assessment report we have described in detail deviations in our assumptions and calculations and justified them, and we have listed the nuclide-specific quantities of activity discharge. These data served as a basis to calculate the radiation exposure due to accidents.

In § 28 sec. 3 of the Radiological Protection Ordinance the dose limits in the environment of a nuclear facility are defined, which have to be observed in planning structural or technical protective measures against the radiological consequences of design basis accidents ("design basis planning limits"). In sec. 4 of the BMI guideline for assessment of the design of nuclear power plants with a pressurized water reactor according to § 28 sec. 3 of the Radiological Protection Ordinance /10/ calculational models and input data are recommended to determine the radiation exposure. These models take into account the following exposure models:

- External radiation exposure due to β-irradiation from the exhaust air (β-submersion),
- external radiation exposure due to γ-irradiation from the exhaust air (γ-submersion),
- external radiation exposure due to γ-irradiation from radionuclides deposited on the ground (γ-ground surface radiation),
- internal radiation exposure due to radionuclides inhaled by respiration (inhalation),
- internal radiation exposure due to radionuclides ingested with contaminated food (ingestion).

The calculational models and input data are not PWR-specific. Thus they can be applied unrestrictedly to activity discharges during an accident from the modular HTR-2 NPP. For this reason we have calculated the radiation exposure due to accidents based on the calculational models and radioecological parameters given in the aforementioned guideline; as a site for this plant has not yet been defined, we did not apply site-specific data. After a concrete site has been selected it has to be verified, if the applied input data still are applicable or if additional, site-specific parameters and exposure pathways have to be taken into account.

In tables 4-3 and 4-4 the results of our calculations of the radiation exposure after accidents are summarized and compared to the dose limits applicable in planning. We did not recalculate our results with respect to recent modifications of the Radiological Protection Ordinance, as these modifications are of minor influence. Our calculations confirm the magnitude of the data given by the applicants in the safety analysis report. They are distinctly below the dose limits according to § 28 sec. 3 of the Radiological Protection

According to our opinion the data on these fields contained in the Safety Analysis Report take all load assumptions and design requirements into consideration, which are applicable in the Federal Republic of Germany independent of the site [e.g. 7], resp. which could arise from the conditions at many potential sites. As, however, the assumptions do not cover the entire potential range of site features, it will definitely be necessary, after selection of a definite site, to check whether the real site features are taken into consideration sufficiently by the design of the plant. Therefore, our statements on the technical design of the reactor system are given with the provison that a definite site does not produce any deviating marginal conditions. Thus, in assessing the concept of the emergency power supply, for example, we have assumed a reliability of the external current supply as given in the Federal Republic of Germany.

4. Accident analysis

4.1. Range of accidents

In compliance with section 7 subsection 2 of the Atomic Energy Act precautionary measures against damage from erection and operation of the plant have to be taken for nuclear facilities according to the state of the art and knowledge of science. It has to be safeguarded especially that in case of activity discharge after an accident the planning guide-lines in compliance with section 28 subsection 3 of the Radiation Protection Ordinance are not exceeded.

To furnish the evidence for these licensing requirements the plant planner has drawn up a list of all design accidents, which have been taken as a basis for the design of the HTR module, in analogous application of the principles of the so-called "accident guide-lines" for nuclear power plants with pressurized water reactors [8]. In doing so, the plant planner followed the line of action of the accident guide-lines and investigated incidents which are decisive for the design of plant parts, and has derived design requirements for buildings, components, and systems from the analysis on the one hand, and on the other hand he defined radiologically representative incidents which he analysed to prove the adherence to the emergency reference levels of section 28 subsection 3 of the Radiation Protection Ordinance.

Furthermore, the plant planner has commented on hypothetical incidents to demonstrate the available safety reserves of the plant and to prove that risk-reducing measures have been taken to a sufficient extent beyond the scope of construction as well.

Due to our review, revisions and extensions to the originally presented accident catalogue were necessary which have been included in the revised version of the Safety Analysis Report. According to our opinion the list on hand now is complete and defines the design scope of the HTR 2 module reactor plant adequately and in the necessary scope in compliance with the requirements of the Atomic Energy Act and the applicable proceedings in nuclear licensing procedures.

Risk-reducing measures have been taken to a sufficient extent by protecting the plant against civilisation-related external events, and by providing the possibility of external supply of the surface coolers and the external power supply to the emergency control unit under consideration of the special design features of the HTR module.

4.2. Analysis of the course of accidents

Apart from checking the accident-related incidents analysed by the plant planner for completeness, the task of our review was also to check and evaluate the analysis of the course of accidents and the statements on the effects resulting thereof.

In doing so we have taken the requirements to be adhered to in compliance with the applicable and transferable rules and guide-lines as a basis, resp. we adapted them to the conditions of the HTR module. This procedure revealed the necessity to revise the originally presented accident analysis.

Unfavourable marginal conditions, for example, had to be taken as a basis for the analyses, as is usual for establishing proof in nuclear licensing procedures, such as

- the assumed failure of the first criterion for the release of the reactor protection system,
- the consideration of an unfavourable initial state of the system under scrutiny,
- the consideration of a single failure and of a potential repair fault in the system crucial for the control of the accident, and
- the non-consideration of the available systems which are not concidered to be safety-related for immediate accident control.

The consequence of this procedure was that additional safety facilities, such as control rod insertion limitation, additional release criteria for reactor protection, such as maximum mainsteam temperature, had to be included in the concept on the one hand, on the other hand limiting safety-values, such as the maximum permissible fuel element temperature, had to be changed, and thus additional proof had to be furnished on their permissibility.

The following paragraphs will briefly deal with some essential results of the revised accident analyses.

4.2.1. Shut-down margin

The temperature-dependent reactivity behaviour of high-temperature reactors causes automatic shut-down at high temperatures through the negative temperature coefficient, whereas it is difficult to provide the necessary shut-down reactivity by means of absorbers located in the side reflectors in case of low temperatures. Therefore, we have placed special attention to this aspect in our investigations of the shut-down system and reactivity accidents.

To safeguard shut-down in all operating modes to below a core temperature of 50°C we requested a limitation of the control rod insertion depth and of the filling height for the pebble shutdown system (KLAK), and we have pointed out to the necessity of an initial start-up measuring programme and a thus combined modification of the loading strategy.

4.2.2. Reactivity accidents

The plant planner has investigated reactivity accidents for the equilibrium core in normal operation of the plant, and has determined the influence of unfavourable operating conditions by means of parameter studies.

Accident-aggravating individual faults on active system parts, such as a faulty acceleration of the blower, and mis-operation of the pebble shut-down system, have been assumed in this case.

Our investigations into the first reactor core, part-load operation, zero-load operation, and start-up accidents as well as investigations into the mistravelling of single absorber rods have confirmed that the course of accident "withdrawal of all reflector rods with maximum speed at full load and equilibrium core" described in the Safety Analysis Report can be considered a covering design basis accident.

Two physically different release criteria are used to start off the reactor scram system necessary to limit the maximum temperatures of the pressure vessel during the covered accident investigated. Even if the first release fails the reactor will be shut down safely, also under the assumption that one reflector rod does not drop in. Temperature limiting values are not exceeded in the process. Impermissible reactor excursions do not occur.

Apart from the effects caused by the withdrawal of the absorber rods we have examined the reactivity-effective incidents of

- faulty acceleration of the blower,
- accident-related lowering of the cold-gas temperature,
- ingress of water into the primary circuit, and
- compaction of the pebble bed in case of earthquakes.

In these cases as well we can confirm that the effects are covered by the design basis accident. In our report we have, however, listed in detail and substantiated some notes and proposals with respect to marginal conditions to be covered, such as power limitation to a maximum of 105% of the rated reactor power and prevention of an unintentional blower start at high helium temperatures as occur with shut-down hot reactor and decay heat removal by the surface cooler. At this point statement is also made that we consider further investigations necessary in the field of compaction of the pebble bed caused by earthquakes under consideration of the transfer conditions of the THTR-tests to the situation of the HTR module core. With respect to the core geometry and the construction of the internal core parts the HTR module core differs from the THTR test model which can cause increased vibrations and extended vibration periods, which have to be analytically, resp. experimentally safeguarded further.

4.2.3. Disturbances in heat removal without loss of coolant

Disturbances in heat removal without loss of coolant are caused be the following incidents

- interruption of the primary coolant circuit,
- disturbances in the main steam removal system,
- disturbances in the feedwater supply system, and
- loss of auxiliary power.

Caused by these failures deviations of the thermodynamic states from the normal operating conditions occur. By the limitation and reactor protection facilities becoming active these deviations are recognized and the necessary protecting actions are initiated.

At this point it is essential to emphasize that contrary to the usual practice at present the total loss of electrical energy, i.e. the failure of the external mains feed as well as the loss of auxiliary power in the absence of emergency energy generation in the HTR module has been taken into consideration in the concept plan as a design basis case.

The plant planner differentiates between the following cases:

- short-term loss of auxiliary power (less or equal 2 hours),
- longer-term loss of auxiliary power (less or equal 15 hours), and
- long-term loss of auxiliary power (more than 15 hours).

In case of short-term loss the plant is shut down via the main heat sink after return of the mains. Design limits are not reached.

In case of longer-term loss the plant is in a safe condition, i.e. the reflector rods have dropped in; the blower has been switched off, and the steam generator is isolated. After two hours the control room is no longer supplied with energy, and is thus inoperative. Due to the initiation orientated failure response of the reactor protection system the primary circuit is locked off and the steam generation relief is released. In the course of the core heat-up phase the safety valve of the primary circuit can react. After return of the auxiliary power within 15 hours, the surface cooler is started up again and the plant is shut down.

Extensive analyses have been carried out for this case. These analyses have revealed that in limited areas of the reactor building the design temperature of the primary cell concrete of 150°C is reached after 15 hours. During this period of time the maximum temperature of the reactor pressure vessel rises to 310°C; the surface cooler reaches 220°C, and thus remains operational. The maximum temperature of the fuel elements is independent of the function of the surface cooler and stays below 1,200°C.

In summarizing it can be stated that even in case of a complete failure of the surface cooler and a reactor under pressure during a period of 15 hours no impermissible loads occur on the reactor pressure vessel inclusive of its internal parts, the containment, and the surface cooler itself. The exposition of radiation into the surrounding environment caused by removal of primary coolant, when the safety valve reacts, remains far below the emergency reference levels stipulated in section 28 subsection 3 of the Radiation Protection Ordinance.

The long-term, total loss of any kind of energy supply for more than 15 hours has not been assumed by the plant planner for the design range. According to our opinion, the planned two-tier design of the emergency energy generation system is only permissible, if a short-term repair of at least one emergency energy generation facility is possible. Our report determines the necessary requirements and demands quality-assuring measures.

4.2.4. Loss-of-coolant accidents

Below, the loss-of-coolant accidents are subdivided into primary and secondary ruptures and leakages as well as damage on the steam generator. The effects are investigated with respect to

- maximum temperatures of fuel elements and components,
- load effects, such as differential pressure loads of the reactor building, and
- radiological loads.

Due to the fact that a break of the pressure vessel is excluded, primary ruptures are only assumed to occur in the joining pipes to the pressure vessel. These pipes have a maximum diameter of 65 mm, resp. in case of the rupture of the withdrawal pipe of the fuel elements the construction is limited to this diameter.

The fact that a break of the pressure vessel can be excluded is safeguarded basically according to the principles determined in the specification of basic safety, and proven in LWR technology. In our opinion additional safety measures are necessary due to neutron-induced embrittlement at low operating temperatures, and for the heat-resistant materials and their weld joints in the area of the main steam nozzle. For the determination of the neutron-induced embrittlement we have demanded a preliminary investigation programme and inset probes.

The substantiation of a break exclusion has been safeguarded in addition by fracturemechanical analyses within the framework of our review. The analyses revealed that in the area of the main steam nozzle a constructive modification was necessary whereby a mixed-seam connection recognized as being critical has been taken out of the high-temperature area. Thus, the break assumptions for the pressure vessel made by the plant planner can be confirmed by us.

Analytically the primary ruptures and leakages have to be differentiated as follows:

- rupture of a large connection pipe,
- rupture of a small pipe, resp. small leaks, and
- ruptures or leaks on pipes carrying primary coolants outside the reactor containment.

The non-lockable double-ended rupture of the largest connection pipe carrying primary coolant with a rated diameter of 65 mm directly at the reactor pressure vessel can be considered typical for all other primary ruptures according to these analyses.

For this accident the adherence to the maximum permissible temperatures of the fuel elements has been checked by independent comparative calculations under consideration of unfavourable initial conditions and uncertainties in the evaluation of influencing parameters to be considered partly systematically, partly statistically. Hence, the temperatures of the fuel elements remain within permissible limits.

Our calculations have also revealed that the insulation of the outer section of the reactor pressure vessel has to be improved so that an adherence to the design temperature can be shown there as well even under consideration of uncertainties in the establishment of initial parameters. The design temperature of the reflector rods stated at 650°C is exceeded considerably. As this fact, however, does not jeopardize the function, and the integrity is not endangered, this matter is not relevant to the concept according to our opinion.

The exposition of the environment to radiation caused by the depressurization accident remains far below the emergency reference levels stipulated in section 28 subsection 3 of the Radiation Protection Ordinance, even when taking the consequential core heat-up phase and the assumed failure of the available filtering system into consideration.

For smaller leakages we have set forth and substantiated in our report the requirements for automatic change-over to a filtering system, and the examinations on the effects of medium-sized leaks which lead to an increase in pressure below the set release pressure of the relief valves. However, these aspects are not relevant to the concept.

Our investigations into secondary ruptures and leakages as well as into the rupture of feedwater or main steam pipes revealed the necessity of changing the concept for the main steam lock-off system in order to keep the increase in pressure inside the reactor building within the limits of the concept for a two-module plant as well. The increase in pressure has been detected by us independently of the design calculations for two typical primary and secondary ruptures each. Therefore, we can confirm the value of 0.3 bar quoted for the excessive design pressure of the reactor building. However, we have to modify the statement as this value refers to the pressure in the reactor hall and to most of the surface of the outside walls of the reactor building. Areas near the rupture are in parts subjected to considerably higher differential pressures which have to be taken into consideration as special loads together with other types of load in the design of the statics.

With respect to the ingress of water into the primary circuit as a consequence of leakages in the steam generator heating tube our analyses revealed that due to the design and operational measures a leakage remains restricted to the diameter of a single heating tube. The water amount of 600 kg flowing into the primary circuit conservatively assessed by the plant planner can thus be confirmed. The initiation of safety actions by only one release criterion to detect moisture can be accepted with a correspondingly high-quality execution within the meaning of KTA-Standard 3501 [9], as is applicable, by the way, for the recognition of a main-steam pipe rupture. The effects of the water steam on the reactivity behaviour in the reactor core and the corrosion of the fuel elements with the appertaining formation of water gas have been examined, and we can confirm that limiting values unacceptable for safety are not reached.

4.2.5. External events

The approaches and methods applicable for licensing procedures have also been used to examine external events with the aim of detecting a potential influence on the plant safety. As a result we have demanded the design of some additional plant parts to withstand the effects of earthquakes. The plant planner has the intention of working out the seismic load assumption to be taken as a basis according to a new procedure by means of an empirical-statistical method of determining seismic ratings on the basis of a probalistic analysis of earthquake threats [10,11]. We have checked this method, and have come to the opinion that this procedure is a consistent method to realistically determine earthquake loads. However, for a real case we have recommended the use of the traditional method of determining the seismic data relevant for the specific site, in addition.

The analysis of other natural events has not caused any concept-relevant demands. Due to their low frequency of occurrence the incidents "aircraft crash" and "blast wave" do not belong to the design basis accidents. Within the framework of our concept review we have convinced ourselves that against these incidents the planned design scope of buildings and other plant parts as well as the potential external supply of the surface cooler and emergency power supply in addition provide sufficient precautions in the sense of risk-reducing measures.

5. Incidents beyond design

It was not the subject matter of our order to comment on the hypothetical incidents examined by the plant planner.

Within the framework of our review of the risk-reducing measures we have, however, convinced ourselves that the inherent safety measures of the HTR module become apparent to a high degree in the examination of such incidents.

References

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[5] Reaktorsicherheitskommission, Leitlinien für Druckwasserreaktoren (3. Ausgabe vom 14.10.1981), 2. Anhang zu Kap. 4.2: Rahmenspezifikation "Basissicherheit von druckführenden Komponenten".

[6] BMI, Sicherheitskriterien für Anlagen zur Energieerzeugung mit gasgekühlten Hochtemperaturreaktoren (Entwurf vom September 1980).

[7] BMI, Richtlinie für den Schutz von Kernkraftwerken gegen Druckwellen aus chemischen Reaktionen durch Auslegung der Kernkraftwerke hinsichtlich ihrer Festigkeit und induzierter Schwingungen sowie durch Sicherheits-abstände (Bundesanzeiger Nr. 179 vom 22.09.1976).

[8] BMI, Leitlinie zur Beurteilung der Auslegung von Kernkraftwerken mit Druckwasserreaktoren gegen Störfälle im Sinne des § 28, Abs, 3 der StrlSchV (Störfall-Leitlinien) vom 18.10.1983 (Bundesanzeiger Nr. 245a vom 31.12.1983).

[9] Sicherheitstechnische Regel des KTA, Reaktorschutzsystem und Überwachungseinrichtungen des Sicherheitssystems KTA-Regel 3501 (6/85).

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THTR 300 MWe Prototype Reactor – Safety Assessment

1. Main design features

- primary circuit

– reactor core	pebble bed consisting of 675000 spherical fuel elements with a diameter of 6 cm (0,96 g high-enriched uranium 235 and 10,2 g thorium 232)
– coolant	helium at a pressure of 39 bar is heated from 250 °C to 750 °C and transported by means of six circulators
– control and shut down systems	for power control and reactor scram 36 absorber rods are inserted or dropped in by effect of gravity into borings in the side reflector (reflector rods), for long term shut down 42 absorber rods with pneumatic drives are inserted directly into the pebble bed (incore rods)
– reactor pressure vessel	pre-stressed concrete reactor vessel with a wall-thickness of 5 m, a diameter of 25 m and a height of 29 m using a steel liner.
• secondary circuit	largely conventional type with steam–feedwater–circuit at a maximum of 535 °C
• beginning of construction	1971
• commissioning	16. Nov. 1985 (first electricity generation)
• beginning of decommissioning	1. Sept. 1989

2. Experience during construction and licensing

A main problem when beginning construction in 1971 was the missing of reliable technical rules and guidelines for the THTR-specific components and for the THTR-specific reactor concept. Therefore the necessary rules and guidelines had to be developed by project accompanying programmes.

The BMI[1] Safety Criteria for nuclear power plants did not come into force until 1977. They were valid for all reactor types, specially for the light water reactor, but they did not take into consideration the specific characteristics of an HTR. For the THTR 300 therefore in 1978 the so called "THTR-planning basis" (THTR-Planungsgrundlagen) were established, which got the agreement of the responsible licensing authority MWMT[2] in 1978. These planning bases were a reactor specific interpretation of the German BMI-Safety criteria from 1977. The safety criteria for HTR, which were developed under contract of BMI by RWTÜV, made the technical requirements on the HTR more precise in 1980.

In consequence some new or more detailed requirements came into force during the construction phase of the THTR:

- external impact (e. g. aircraft crash, pressure wave, earthquake)
- internal impact (e. g. pipe whipping, pressure vessel damage)
- new radiation protection requirements (e. g. reduction of the radiation exposure of the personnel).

[1] BMI – German Federal Ministry of the Interior (the responsibility for nuclear safety was later changed to the Federal Ministry for Environmental Protection)
[2] MWMT – Ministry for Economy, Trade and Technology of the State of North Rhine Westphalia

3. Operational history

The electric power output during the operation time of the THTR 300 reached a total of 2.891.068 MWh. The plant was in operation over 16.410 h and had a time utilization factor of 61 %.

The time history shows a lot of power changes and several prolonged plant shut down times (see fig.):

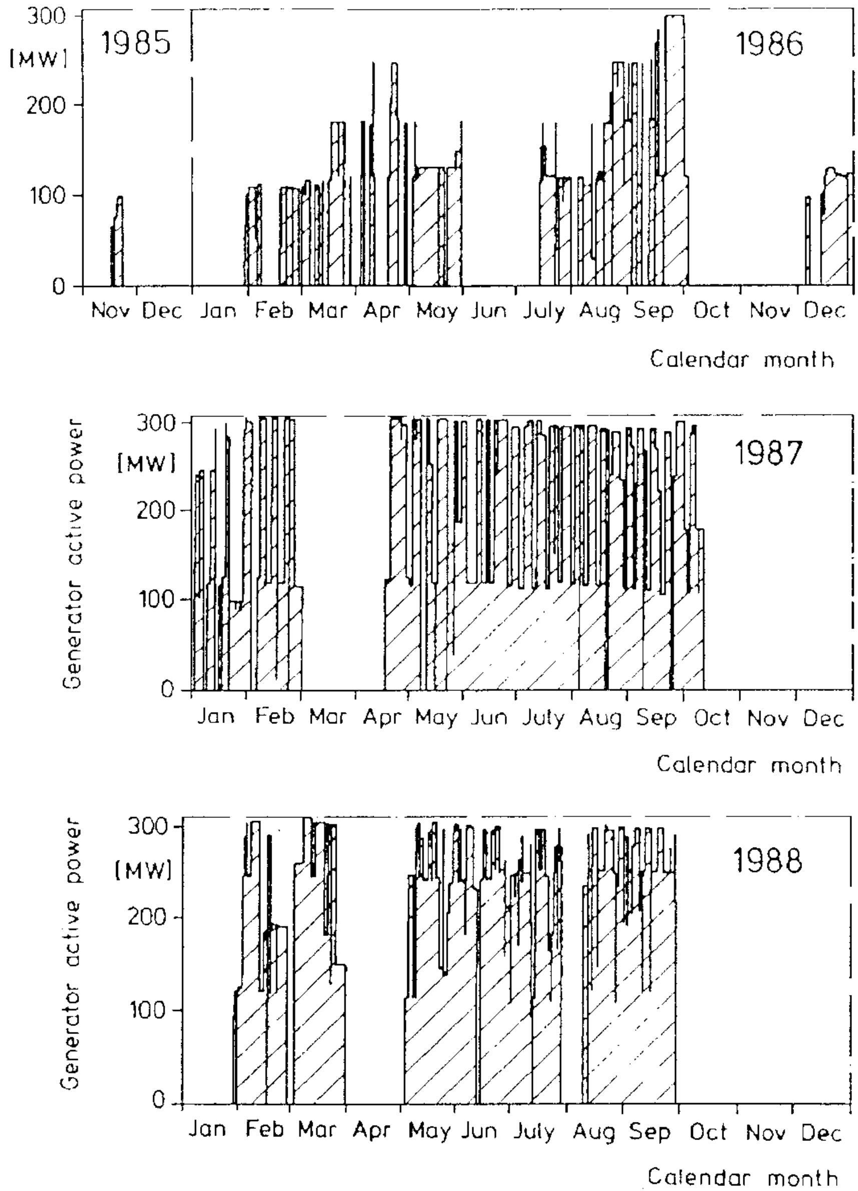

Fig. THTR 300, electric power output during operating phase between 16 Nov 1985 and 31 Dec 1988.

These plant shut times had different causes, which are explained in section 5. Experiences during construction and licensing have been detailed in /1/ to /3/.

4. Main positive experiences

There were a lot of positive experiences from the early phase of the commissioning tests to the actual operation phase:

- The first criticality was reached with a load of 198.180 spherical elements. That means a deviation of only 4.500 elements from the design value and showed a good correspondence between theoretical calculations and real loading.
- All reactivity measurements during the commissioning phases with the different core cooling media air, nitrogen and helium have confirmed the precalculations.
- There were no problems with the core power control in any level between 40 and 100 % power.
- The two independent shutdown systems (reflector rods and incore rods) ensured sufficient subcriticality in all cases, in the short-term shut down as well as in the long-term shut down.
- The two shut down procedures, which were planned for the THTR 300, were repeatedly triggered by the plant protection system. Experience showed that the systems had sufficient availability and functional capability.
  (For the reasons of the multitude of unplanned triggering see 5.)
- The design data of the primary and secondary system could be confirmed during operation e.g.:
  - The special THTR-components such as the fuel, reflector and incore rods, helium circulators, steam generators and the concrete reactor pressure vessel and even the dry cooling tower were tested in a nuclear plant with full success.
  - The in service inspection of primary components could be performed at low radiation exposure of the plant personnel.
  - The radiation of the plant personnel was generally at low values.
  - The spherical fuel elements showed the planned good retaining of the fission products, although some of the elements were broken into pieces caused by the insert of the incore control rods. But this was never a problem of increased radiation.

5. Main incidents and problems

The operation of the THTR 300 showed some incidents and problems:

- Difficulties with the refuelling system, because the withdrawal of the spherical fuel elements was only possible with reduced helium-mass-flow; this problem was solved in 1987 by a complex repair operation (Further openings were cut by means of spark erosion in the region of the flow cross section near the singularizer disc. That was only possible under the depressurised primary circuit.)

- Damage of spherical fuel elements caused by frequent and deep insertion of the incore control rods during the commissioning phase; the share of damaged elements in the total amount of the withdrawn elements decreased from 1,5 % in the beginning to 0,6 % towards the later periods of operation.
  Due to the higher share of damaged elements the casks for broken elements had to be changed earlier than planned.

- Damage of some bolts (35 of 2.600 bolts) of the thermal insulation in the hot-gas-ducts[1],which were discovered during the routine inspection in 1988. The analysis of this event by RWTÜV showed, that the insulation was still sufficiently safe fixed; and furthermore there were enough possibilities and means to detect a loosening of the metallic insulation by the operational monitoring.

- The measurements of the primary system data showed that the core outlet temperature was locally higher and lead locally to higher fuel element temperatures, which however remained below the design values of the fuels and the other materials. It may have been caused by a higher bypass of the helium mass flow than expected.

- Graphite dust mass in the primary circuit was higher than expected. This was found during in service inspection. The reasons for this could not been cleared during the operation time.

[1] This construction is specific to the THTR design inside the prestressed pressure vessel

6. Overall performance and safety features

In the above sections of this report, we explained, that the THTR has fully reached its operational target and confirmed the feasibility and safe operability of a high temperature reactor (HTR) based on the pebble bed principle.

In detail:

- the performance for full power operation was demonstrated;
- the principle of continuous reactor refuelling with new elements without operation interruption was demonstrated;
- the inherent safety characteristics of the reactor were proven;
- it was shown that the maintenance and the in service inspection of this type of reactor was possible under low radiation conditions for the personnel and the surroundings.

The relatively long time from the beginning of construction in 1971 until the first electric power generation in 1985 was caused by:

- the prototype character of the THTR 300,
- the requirements of the German law ("Atomgesetz") to bring the technical concept up to the status of science and technology and
- the missing HTR-specific rules, which still had to be created.

Some of the technical requirements, which had great consequences on the plant concept and therefore on the time schedule of the plant construction are as follows:

- the redundance of decay heat removal system
- constructional requirements due to earthquake load (particularly the additional consideration of a vertical component of the earthquake)
- external impact particularly aircraft crash
- internal impact due to a conventional pressure vessel damage
- assumption of pipe fractures up to 2 F-breaks and their consideration in the course of plant construction
- experimental proof of leak before break concept in order to minimize the number of safety mechanisms against pipe whipping
- optimise the accessibility for plant maintenance (e. g. in service inspections)

The main difficulties due to the prototype character of the reactor are discussed above. They hade a great contribution to the delays during construction, commissioning and operation.

After the discovery of the damages of some insulation bolts of the hot-gas-ducts in 1988 extensive investigations were done by the constructor and by the independent expert - RWTÜV. The result was – also confirmed by the authority -, that there were no technical objections against further operation.

There were no technical and safety reasons for finishing the operation in 1989. The reasons were financial and economical considerations.

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Reprint from

"CURRENT
NUCLEAR POWER PLANT
SAFETY ISSUES"

VOL. II

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 1981

PROPOSED SAFETY CRITERIA FOR HIGH-TEMPERATURE GAS-COOLED REACTORS

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Abstract

PROPOSED SAFETY CRITERIA FOR HIGH-TEMPERATURE GAS-COOLED REACTORS.

Several countries have carried out programmes for the development of the High-Temperature Gas-Cooled Reactor (HTGR). However, until now little work has been done in developing criteria and guides for HTGRs. In the Federal Republic of Germany (FRG), nuclear power plants have to meet the "Safety Criteria for Nuclear Power Plants". They were mainly established for Light-Water Reactors (LWRs). They also have to be applied to other reactor types, indirectly however when plant specific systems are considered. For developing safety criteria for HTGRs in the FRG the German safety criteria have been taken as a basis while considering proposed foreign regulations for HTGRs. The safety criteria have been divided into three different groups, each of which has been treated in a different way: the safety criteria which refer to inspections and testability, shutdown systems, reactor coolant boundary, residual-heat-removal systems and containment design have been essentially revised because of the properties and inherent safety characteristics of an HTGR power system; another group would have been applicable to LWRs and HTGRs without modifications but was improved and completed following experience with nuclear power plants and work in establishing standards; the third group was found to be independent of the reactor system and it is proposed without modifications for HTGRs. This group is formed of criteria referring to basic plant safety principles, radiation exposure of the environment, external influences, fire and explosions, plant security, escape routes and communications, decommissioning and ventilation systems. At present a draft of safety criteria for HTGRs is being discussed with the different groups participating in the licensing process. Because of its general character the IAEA standard "Design for Safety of Nuclear Power Plants, A Code of Practice" is applicable to HTGRs without the need for much interpretation; in the case of "Emergency Core Cooling" analogous requirements in the HTGR design are to be met.

1. INTRODUCTION

Several countries have carried out programmes for the development of High-Temperature Gas-Cooled Reactors (HTGRs). In the USA and the Federal

Republic of Germany two experimental power stations with HTGRs have been operated successfully: Peach Bottom No. 1 with an electric output of 40 MW and the reactor of the Arbeitsgemeinschaft Versuchsreaktor (AVR-Reactor) with 15 MW. In the USA, the Fort St. Vrain (FSV) power plant with prismatic fuel elements and an electric output of 330 MW is already in operation, and in the FRG the Thorium-High-Temperature-Reactor THTR-300 with 300 MW and a pebble bed reactor core is still under construction. More units are in the design phase.

Until now, little work has been done in developing safety criteria and guides for HTGRs. In the FRG, nuclear power plants have to meet the "Safety Criteria for Nuclear Power Plants" [1]. Although they were established for Light-Water Reactors (LWRs) in the first place, they also apply to other reactor types. Such guides may be applied indirectly when considering plant-specific systems of the other reactor types. In order to avoid interpretations of Ref. [1] in further HTGR projects which will not always lead to solutions tailored to the HTGR, the Federal Ministry of the Interior suggested that separate safety criteria defining design principles for HTGRs be developed [2]. These will facilitate the safety assessment of these plants during the licensing procedure and serve as a planning objective for the vendor.

2. STARTING POINT FOR DEVELOPING HTGR SAFETY CRITERIA

A search for safety criteria and guides specific to HTGR power plants shows the following results: most activities in developing criteria and guides have been undertaken in the USA. The following are the most important ones:

> A draft of "Nuclear Safety Criteria for the Design of Stationary Gas-Cooled Reactor Plants" with supplements prepared by the American Nuclear Society [3]
>
> An analysis showing whether the Regulatory Guides are applicable to HTGRs
>
> Revision of the "General Design Criteria for Nuclear Power Plants" presented in Appendix A of Part 50, Title 10, Code of Federal Regulations, for the application to HTGRs (draft) [4].

In the FRG, within the standards of the Kerntechnischer Ausschuß (KTA) the KTA 3102 [5] "Core Design of HTGRs" is being developed.

The Safety Criteria for Nuclear Power Plants [1] issued by the Federal Ministry of the Interior of the FRG have been taken as a basis for establishing Safety Criteria for High-Temperature Gas-Cooled Reactors [2].

The intentions were:

- to retain as close as possible the basic concept of the safety criteria [1],

- to revise the safety criteria [1] with consideration of inherent HTGR safety characteristics, experience gained during the THTR-300 licensing process, experience from HTGR plant operation and safety criteria already proposed,

- to revise plant non-specific safety criteria with respect to the need of modifications due to experience with LWR-plant operation and standards,

- to keep the safety criteria for HTGRs so general that they can be applied to different HTGR concepts such as pebble bed or prismatic core or process heat application.

According to these principles, the safety criteria [1] have been divided into three different groups:

- criteria which have to be revised for an application to the HTGR because of being too LWR-specific,

- plant non-specific criteria which could be improved or completed because of licensing or operating experience,

- plant non-specific criteria not needing any modification because of their general applicability.

These groups will be treated subsequently.

3. CRITERIA WITH HTGR-SPECIFIC MODIFICATIONS

The first group comprises the following proposals for HTGR safety criteria:

No. 2.2[1]	Testability
No. 3.1	Reactor core design
No. 3.2	Coupling characteristics of the reactor core
No. 3.3	Internals of the pressure-bearing vessel
No. 3.4	Systems for control and shutdown of the reactor

[1] The numbers refer to Ref. [2].

No. 4.1	Reactor coolant boundary
No. 4.2	Design basis of the reactor coolant boundary
No. 4.3	Pressure-bearing vessel
No. 5.1	Residual heat removal after operation
No. 5.2	Residual heat removal after accidents
No. 8.1	Nuclear reactor containment
No. 8.2	Containment design basis

As these include HTGR characteristics they will be discussed in some detail.

3.1. Testability

Fundamentally, the safety criteria require that all parts of a nuclear power plant shall be so constructed and arranged that they can be tested and inspected to an extent corresponding with their significance for safety. However, in the HTGR design, there are components of high importance to safety with limited accessibility, e.g. the liner as a part of the reactor coolant boundary and some graphite structures in the pressure bearing vessel. Therefore special measures shall be taken for these components to compensate for the disadvantages of limited accessibility, e.g.:

Additional safety margins in the design
Special material properties, e.g. purity
Fabrication quality
Design of systems and components, e.g. redundant structures
Limiting and controlling of operational parameters
Periodic replacement of components.

As a result of these measures, a fault-free condition or function of components must be maintained or the consequences of failures must be limited, in order to assure safe shutdown of the reactor, residual heat removal and limitation of any radioactive release below acceptable limits under all operational and accident conditions.

3.2. Core design and systems for control and shutdown of the reactor

Similarly to LWRs, two independent and diverse reactivity control systems are required. One of these shall be capable of shutting down the reactor from all operational and accident conditions for a sufficient period; the second shall be capable of maintaining cold shutdown for unlimited time. A single failure which may result in a failure of control elements shall not impair the system from fulfilling its safety function.

The following inherent safety characteristics of the HTGR:

graphite structure with a high heat capacity, thermal conductivity and phase stability,

phase stability of the coolant helium,

fuel element not sensitive to overheating,

negative temperature coefficient of reactivity,

should be considered when specifying requirements for shutting down the reactor.

Therefore, a design is acceptable in which hot shutdown conditions are provided by increasing the average core temperature. This has been verified experimentally with the AVR-high-temperature reactor power plant by turning off the helium circulators without moving the control rods. This results in a reduction of the coolant flow and an increase of the core temperature. These means are now the usual shutdown procedure for this reactor [6].

In HTGR safety criteria it is proposed that inherent safety characteristics of the nuclear reactor may be taken into account to reduce shutdown system hardware.

HTGR-specific operation and accident conditions must be considered in core and shutdown system layouts, e.g. water ingress from a damaged heat exchanger into the reactor core which may result in an increased neutron multiplication factor.

The influence of the HTGR characteristics on the remaining core design criteria (3.1–3.3) of Ref. [1] is of minor importance.

3.3. Reactor coolant boundary and pressure-bearing vessel

In the case of LWRs, the reactor coolant pressure boundary including the pressure vessel consists entirely of metallic components; therefore one criterion in Ref. [1] proved sufficient for this system. According to HTGR design it is preferable to specify requirements for the metallic components which represent the enclosure of the reactor coolant and for the pressure-bearing vessel separately.

In detail, the reactor coolant boundary consists of:

the leaktight liner of the pressure bearing vessel,

penetrations through the vessel including their closures,

reactor coolant piping including the first isolation valves,

isolation lines including the first isolation valves, pipelines penetrating the vessel and interfacing with the reactor coolant at their outer surface (e.g. heat exchangers in the primary circuit).

In addition to general requirements for design, testing, materials and leakage monitoring instrumentation, HTGR-specific features are to be considered: The penetrations of the pressure-bearing vessel must be secured against outward forces, and consequences of closure failure must be mitigated by limiting the blowdown flow area, e.g. by provision of flow restrictors.

In current HTGR design, a prestressed concrete pressure vessel bears the pressure of the primary circuit in the liner region and together with the liner provides for the safe enclosure of the radioactive substances. Apart from general design requirements, the following special safety requirements have to be considered:

To provide a thermal protection of the vessel if necessary, e.g. an isolation or a heat removal system

To consider additional pressure and temperature loads by possible liner leakage of the reactor coolant

To withstand loads induced from pressure waves, airplane crashes and earthquakes

To achieve a sufficient safety margin for the stress limit of the pressure-bearing vessel in all relevant accident conditions.

3.4. Residual-heat-removal systems

Reliable residual-heat-removal systems are required for operational and accident conditions. However, the accident residual-heat-removal system can be used for residual-heat removal in normal operation, if it is adequately designed, e.g. with respect to reliability. This may be essential for process heat application of the HTGR because the heat sink for normal operation might not be suitable for all residual-heat-removal conditions.

The following features have to be considered when establishing requirements for the accident residual heat removal system:

No total loss of coolant occurs in an HTGR system so that a minimum helium pressure remains in the primary circuit.

Ingress of foreign media, e.g. water, air into the primary circuit or chemical reactions may occur.

Inherent safety characteristics should be considered, e.g. the properties of graphite and the fuel element.

The accident residual-heat-removal system is required to be reliable and redundant. It has to fulfil its safety-related functions even in maintenance during operation with simultaneous occurrence of an additional single failure. An emergency residual-heat-removal system with special requirements, e.g. core flooding, is not necessary.

The proposals for the design requirements include the following aspects:

Ingress of foreign media, e.g. water, into the primary circuit or chemical reactions has to be considered in some accidents

The residual-heat-removal systems for normal operation and accidents may possess common components, if the reliability and the requirements for maintenance of the accident system are not negatively influenced and the quality of these components is adequate

If inherent characteristics can assure residual-heat removal or storage after accidents so that design limits are not exceeded, hardware requirements can be softened, e.g. the requirement for meeting the single-failure criterion with respect to the residual-heat-removal system during its maintenance can be suspended for an adequate time period.

A three-hours interruption of residual-heat removal has been investigated theoretically for the THTR-300 power plant [7]. According to this, the structure of the core is preserved in order that the residual-heat removal can be resumed by the active systems after three hours and that the reactor can be brought into a safe state without exceeding radiological limits at all times.

3.5. Nuclear reactor containment

For HTGRs which are at present in the design phase, a concrete containment with an inner liner is planned. The following containment criteria are proposed:

No. 8.1	Nuclear reactor containment
No. 8.2	Containment design basis
No. 8.3	Leakage tests of the containment
No. 8.4	Containment penetrations

The requirements of the last two are not HTGR-specific.

In the design requirements of the containment the option to have a high-pressure containment rather than a controlled vented containment is left open. It has been explicitly stated that during external events

– the containment shall remain both leaktight and structurally intact, if it cannot be shown that the requirements of the Radiation Protection

Ordinance [8] with respect to radioactive releases are adhered to as a result of accident or operational leakage from a non-leaktight structure, or

– the containment need remain only structurally intact, if it can be shown that even without leaktightness the requirements of the Radiation Protection Ordinance are met.

To prevent damage of the containment or the safety systems from possible inner explosions, it is required that the formation of potentially combustible gas mixtures or the consequences of the reaction of these mixtures to the containment are to be limited during potential accidents in order that the fulfilment of the containment function be maintained.

This requirement has to be considered especially for process heat application of the HTGR.

It is required that the containment and the safety systems in the containment be designed to withstand ambient accident conditions, e.g. the temperatures arising. Therefore, a system for heat removal from the containment does not seem to be necessary if the interior of the containment is designed adequately and is not explicitly required in HTGR criteria.

4. CRITERIA APPLICABLE TO LWRs AND HTGRs

This group of safety criteria in principle could be applied to LWRs and HTGRs without modifications but could be improved in some aspects following experience gained with nuclear power plants and during establishing standards, e.g. the KTA-Standards. The group comprises the following criteria:

No. 2.1[2]	Quality assurance
No. 2.4	Radiation exposure in the plant
No. 2.5	Working conditions
No. 6.1	Reactor protection system
No. 6.2	Accident instrumentation
No. 6.3	Operational instrumentation
No. 6.4	Control room and emergency control station
No. 7.1	Electrical power supply
No. 8.3	Leakage tests of the containment
No. 8.4	Containment penetrations
No. 8.5	Liquid contaminant barrier

[2] The numbers refer to Ref. [2].

No. 10.1 Radiation protection monitoring
No. 10.2 Activity monitoring in exhaust air and waste water
No. 10.3 Environmental monitoring

No. 11.1 Handling and storage of nuclear fuel and other radioactive substances.

Examples of modifications are as follows.

Within criterion 2.4 it is explicitly required that provision be made in advance that maintenance operations of inspection, periodic tests, repair or replacement of components can be performed in accordance with the requirements of the Radiation Protection Ordinance [8]. These provisions may cover transport equipment, storage facilities, shielding and the corresponding space.

A separation into two parts and separate specifications of the accident instrumentation, namely the accident event and accident consequence instrumentation is proposed within criterion 6.2, "Accident instrumentation".

A criterion not previously included in the Safety Criteria of Nuclear Power Plants [1] was added to provide a barrier against release of radioactive liquids in the plant buildings for building and ground-water protection (Criterion 8.5 "Liquid contaminant barrier"). Such a barrier is to be provided inside the building and should allow for easy decontamination.

5. COMMON CRITERIA

Because of their general nature, the following safety criteria of Ref. [1] were found to need no modifications:

No. 1.1 Basic principles of the safety precautions

No. 2.3 Radiation exposure of the environment
No. 2.6 Effects from external events
No. 2.7 Protection against fire and explosions
No. 2.8 Access control, off-limit areas
No. 2.9 Escape routes and means of communication
No. 2.10 Decommissioning of nuclear power plants

No. 9.1 Ventilation and air filtration systems.

6. APPLICABILITY OF THE IAEA STANDARD "DESIGN FOR SAFETY OF NUCLEAR POWER PLANTS, A CODE OF PRACTICE" TO HTGRs

On the international level, the Code of Practice "Design for Safety of Nuclear Power Plants" [9] published within the IAEA Safety Standards corresponds to the national "Safety Criteria for Nuclear Power Plants" [1]. The discussion whether this is applicable to HTGRs without detailed interpretation can be restricted to the five parts of section 3, because the other parts of the standard are citing fundamental protection objectives or, as already discussed above, contain requirements that are not plant-specific.

6.1. Provision for in-service testing, maintenance, repair, inspection and monitoring (Section 2.9 of Ref. [9])

In principle, here measures are required for in-service testing, maintenance, repair, inspection and monitoring of the functional capability of components. As described above, for some of the HTGR components there is restricted accessibility. In section 2.9 of [9] this restricted accessibility is taken into account by requiring "adequate safety precautions" to compensate for potential undiscovered failures. These adequate measures should especially be considered in the case of HTGRs as described and particularly emphasized in the corresponding German criterion 2.2 "Testability".

6.2. Reactor core (Section 4 of Ref. [9])

The criteria for core and fuel design and reactor control and shutdown system layout are kept sufficiently general so that they can be applied also to HTGR power plants. In the German proposal for a shutdown criterion we emphasized inherent safety characteristics by which hardware measures for the shutdown systems may be simplified.

6.3. Reactor coolant system (Section 6 of Ref. [9])

Section 6 of Ref. [9] comprises design requirements for the residual-heat-removal systems and for the reactor-coolant boundary including the pressure vessel. The requirements for the reactor-coolant boundary are general, so that they can be applied to HTGRs as well. However, for criteria more specific to HTGRs, it is better to differentiate between the enclosure of the coolant and the pressure-bearing vessel as discussed above.

Section 6.6 of Ref. [9] "Emergency core cooling" is LWR-specific in essential aspects so that interpretations for the application to HTGRs are necessary. Thus, section 6.6 requires an emergency-core-cooling system for the case of the loss-of-coolant accident in order to comply with the design value of the cladding temperature of the fuel elements. Since a total loss of coolant does not occur in HTGRs, we think it is useful to establish the criterion considering all accident conditions including the depressurization accident. This system can also be used for the residual-heat removal in normal operation, provided it is adequately designed. The requirements for this system have been discussed in section 3.4.

6.4. Containment system (Section 8 of Ref. [9])

Within the requirements of section 8.1 of Ref. [9] "Purpose of containment system" it is possible to have a vented or a hermetically sealed containment depending on other means of limiting the release of radioactive substances. These features are consistent with the requirements for an HTGR-containment within section 3 of this paper. Effects of potential HTGR-specific energy sources on the containment structure, e.g. from reactions of air with graphite or the formation of combustible gases, are included within section 8.2 of Ref. [9] "Containment structure strength". Our opinion is that the requirements of the Code of Practice cover the HTGR containment requirements as well.

7. STATE OF HTGR CRITERIA DEVELOPMENT

A draft of Safety Criteria for High-Temperature Reactors [2] has been established in the FRG. At the moment this draft version is being discussed with the different groups participating in the licensing process.

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DISCUSSION

O.J.A. TIAINEN: In developing your safety criteria for high-temperature gas-cooled reactors did you take the closed-cycle gas-turbine system into account?

K. HOFMANN: Yes, we wanted to establish criteria of a basic character applicable to different HTGR systems, e.g. pebble bed core, block fuel core, or HTGRs for process heat application.

J. DECKERS: The Ministry of the Interior has issued special safety guidelines requiring high toughness, low stresses, and meticulous fabrication and testing for the pipework, pressure vessels, etc. of the auxiliary systems of light-water reactors in the Federal Republic of Germany. This constitutes the so-called "basic safety" framework. Is it intended to develop similar requirements which take into account the high temperature in the reactor and the use of special materials for high temperatures?

K. HOFMANN: In my opinion it is too early to establish specifications for "basic safety" with respect to HTGRs because of the high temperatures involved and the special operating conditions. For instance, before we can think about the process-heat application of HTGRs we must first develop special materials.

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Hochtemperaturreaktor-Technologie,
Genehmigungsentscheidende Sicherheitsaspekte beim THTR;

Besuch einer US NRC-Delegation vom 23. bis 26.7.2001
im FZ Jülich und bei der GRS Köln

1. THTR, verfahrens-/sicherheitstechnisch wesentliche Konstruktionsmerkmale

- Zweikreisanlage
  (Primärkreis: Helium; Sekundärkreis: Wasser/Dampf)

- Reaktorkern (Kugelhaufen) mit hochangereichertem Brennstoff (U/Th-Oxide, 93% U 235) in Partikeln mit keramischer Hülle in Grafit-Kugeln, die abhängig vom Abbrand mehrfach umgewälzt, im Kern umgelagert und schließlich ausgeschleust werden

  Brennstofffreie Grafit- und Bor-Kugeln im Einlaufbetrieb (Leistungsprofilabflachung, Kompensation von geringer Überschussreaktivität)

  Kühlgas strömt von oben nach unten durch den Reaktorkern

- Brennstoffwechsel unter Last

- Reaktorabschaltsysteme (Stabsysteme)

  Reflektorstäbe (B_4C), 1. Abschaltsystem, in freie Bohrungen des Grafitreflektors einfahrend

  Kernstäbe (B_4C), 2. Abschaltsystem, direkt in den Kern eintauchend, mit NH_3-Einspeisung zur Minderung der Reibkräfte

  ursprünglich zusätzlich ein BF_3-Abschaltssystem (Notabschaltsystem)

- Spannbetonreaktordruckbehälter mit innenliegenden Spannkabeln, axial und in Umfangsrichtung vorgespannt

- Wasser/Dampf-Kreislauf mit einfacher Zwischenüberhitzung (Heißdampf/Turbo-Generatorsatz), Zwischenüberhitzer im Gleitdruckbetrieb

  6 Helium/Wasser-Wärmetauscher (Dampferzeuger/Gebläse-Einheiten) in Kaverne des Reaktordruckbehälters eingesetzt (so genannte integrierte Bauweise)

  elektrisch angetriebene Helium-Gebläse

  je 2 Dampferzeuger/Gebläse-Einheiten für die Notkühlung vorgewählt, 2 separate und räumlich getrennte Notkühlstränge

- Trockene Rückkühlung (Seilnetz-Kühlturm) im Normalbetrieb (bestimmungsgemäßen Betrieb)

  Nasse Rückkühlung über Zellen-Kühler nach Anforderung der Reaktor-Schnellabschaltung (Reaktornachkühlung)

2. Genehmigungsverfahren - Chronologie

12.01.1970	Antrag auf Genehmigung zur Errichtung und zum Betrieb
26.11.1970	Erörterungstermin in Unna, 497 Einwender
03.05.1971	1. Teilgenehmigung zur Errichtung (1. TEG)
13.01.1979	1. Klage vor Gericht (gegen Genehmigung des Gebäudes für Speisewasserbehälter und Anfahrentspanner)
19.07.1983	1. Inbetriebnahmegenehmigung (Nullenergieversuche)
13.09.1983	1. Kritikalität (198 180 BE)
09.04.1985	1. Betriebsgenehmigung (Leistungsversuchsbetrieb)
16.11.1985	Erste Stromlieferung ins öffentliche Netz (Synchronisation mit Verbundnetz)
23.09.1986	Erstmals Volleistung (750 MJ/s, 300 MW)
29.09.1988	Planmäßige Außerbetriebnahme zwecks Revision

27.10.1988	Feststellung von Schäden an Befestigungselementen der Heißgaskanalisolierung; von ca. 2600 Bolzen sind 33 Zentralbolzenköpfe abgesprengt
26.04.1989	Nach umfassender sicherheitstechnischer Auswertung Anlage anfahrbereit EVU + GOV. (POLITICS)
15.08.1989	Beschluss der Landesregierung zur geordneten Abwicklung bei sofortiger endgültiger Stilllegung
26.09.1989	Antrag auf Stilllegung Licensing Procedure
seit 10.1997	Anlage im so genannten Sicheren Einschluss für ca. 30 Jahre Safe Enclosure

3. Sicherheitstechnisch wesentliche Auslegungsfragen im Genehmigungsverfahren

3.1 Stabiler Leistungsbetrieb, sichere Reaktorabschaltung

3.1.1 Warum noch ein BF_3-Notabschaltsystem?

Ein solches System war für den Fall des Versagens eines großen Teils der Kernstäbe, denen 1971 noch die Funktion des 1. Abschaltsystems zugewiesen war, als 2. Abschaltsystem zur langfristigen Abschaltung des drucklosen Reaktors vorgesehen; die Reflektorstäbe sollten in erster Linie der Reaktorleistungsregelung dienen (vgl. Bescheid Nr. 7/1). 1981 waren durch Analysen und Experimente die für erforderlich gehaltenen Reserven in der Abschaltsicherheit nachgewiesen, sodass von der Installation eines Notabschaltsystems Abstand genommen werden konnte.

Mit dem Reflektorstabsystem wurde die Schnellabschaltung durchgeführt. Hierzu wurden 4 Gruppen der steuerungstechnisch in 6 Gruppen zu je 6 Stäben zusammengefassten Reflektorstäbe in der oberen Endstellung für die Abschaltung bereitgehalten. Die übrigen beiden Reflektorstabgruppen wurden mit zur betrieblichen Leistungsregelung herangezogen. Bei einer Schnellabschaltung fielen diese Stäbe jedoch ebenfalls auf die volle Tiefe ein, wurden aber in ihrer Wirksamkeit bei der kernphysikalischen Auslegung nicht berücksichtigt, da ihre jeweilige Ausgangsstellung vom Betriebszustand des Reaktors abhängig war. Bei einer Schnellabschaltung im Gleichgewichtscore reichten nach den Analysen die Reflektorstäbe in der Regel allein aus, um den Reaktor für mehr als eine Stunde um mindestens 0,5 Nile unterkritisch zu halten. Nach den Vorschriften betreffend den Automatisierungsgrad des Reaktorschutzsystems (KTA 3501) musste der Reaktor für mindestens eine halbe Stunde durch automatische Schaltmaßnahmen in einen sicheren Zustand gebracht und gehalten werden.

Mit dem Corestabsystem wurde die Langzeitabschaltung durchgeführt. Daneben wurden die Corestäbe auch für die betriebliche Regelung verwendet. Je nach Aufgabenstellung wurden die Stäbe durch den Kurzhub- oder den Langhubkolbenantrieb bewegt. Der Kurzhubantrieb wurde zur betrieblichen Leistungsregelung und zur Kompensation langfristiger Reaktivitätsänderungen des Reaktorkerns benutzt. War nach einer Schnellabschaltung eine Langzeitabschaltung erforderlich, wurden die Corestäbe mit dem Langhubkolbenantrieb über Handauslösung in den Kern nachgefahren. Ging der Schnellabschaltung ein Störfall voraus, der zum Einsatz der Notkühlung führte, wurde je nach Störfallart der Langhubantrieb auch automatisch über das Reaktorschutzsystem ausgelöst.

Die Corestäbe allein waren selbst bei Ausfall des reaktivitätswirksamsten Stabes in der Lage, den Reaktor für beliebig lange Zeiten und die niedrigste im System mögliche Temperatur aus allen Betriebszuständen heraus bei konservativer Bilanz um mindestens 1,2 Nile unterkritisch zu halten.

3.2 Ausreichende Nachkühlung des abgeschalteten Reaktors

3.2.1 Beherrschung der anlageninternen Störfälle (Rohrreißer im Wasser Dampf/Kreislauf, Sprödbruchversagen von Behältern, Folgeschäden durch Dampfstrahlkräfte)?

Etwa ab Mitte der 70er Jahre wurde allgemein viel über die Vermeidung spröden Versagens von druckbeaufschlagten Komponenten diskutiert. Die Ergebnisse führten zu Forderungen nach weiterer Redundanz, räumlicher Trennung redundanter Sicherheitssysteme, Rohrausschlagsicherungen oder lokaler Verbunkerung. Insbesondere der Wasser/Dampf-Kreislauf war hiervon betroffen. So wurden z.B. an Stelle eines großen Speisewasserbehälters zwei kleinere Behälter für erforderlich gehalten und die beiden Anfahrentspanner verbunkert. Die vier Behälter wurden abseits von den übrigen Nachwärmeabfuhrkomponenten in einem separaten Gebäude angrenzend an das Maschinenhaus errichtet. Ausgerechnet gegen diese eindeutig sicherheitstechnische Verbesserung richtete sich nach 9 Jahren die erste Klage im Genehmigungsverfahren vor Gericht. Die Klage blieb in 2. Instanz ohne Erfolg.

Für den THTR kam hinzu, dass für den Wasser/Dampf-Kreislauf nur beschränkt auf Werkstoffe aus der Leichtwasserreaktor-Technologie zurückgegriffen werden konnte; gleichwohl lagen hinreichend Erfahrungen mit geeigneten Werkstoffen aus dem Betrieb fortschrittlicher Kohle-Kraftwerke vor, die jedoch nach aus der Verfahrenstechnik für LWR abgeleiteten Regeln qualifiziert werden mussten.

3.3 Einschluss radioaktiver Stoffe

3.3.1 Das der ersten Teilerrichtungsgenehmigung zu Grunde liegende Sicherheitskonzept?

In Anlehnung an die bekannte LWR-Sicherheitsphilosophie wurde 1971 dem THTR als "Größter anzunehmender Unfall (GaU)" zu Grunde gelegt:

Totaler Verlust des Primärkühlgases durch Bruch einer primärgasführenden Rohrleitung von 65 mm Durchmesser und das gleichzeitige Versagen der Sicherheitsabsperrarmatur. Die Dosis an der ungünstigsten Stelle in der Umgebung betrug unter Anwendung damaliger Rechenvorschriften < 1,50 mSv (< 150 mrem) (vgl. Bescheid Nr. 7/1).

Von dieser Auslegungsphilosophie wurde in den folgenden Jahren entsprechend dem Fortschritt in den Sicherheitsbetrachtungen mehr und mehr Abstand genommen und die Systemauslegung durch Ergebnisse aus Zuverlässigkeitsuntersuchungen ergänzt und verfeinert. Solche Ergebnisse haben sich insbesondere auf die Auslegung der Nachwärmeabfuhr- und Notstromversorgungssysteme ausgewirkt.

3.3.2 Kein Containment?

Nach den Prüfergebnissen zu den Auswirkungen anlageninterner Störfälle war ein Containment zur Einhaltung von gesetzlichen Strahlenschutzgrenzwerten nicht erforderlich; zunehmend verschärfte Anforderungen zum Schutze gegen Einwirkungen von außen hätten sich in einigen Fällen leichter bei Vorhandensein eines entsprechend ausgelegten Containments erfüllen lassen. Der geforderte Schutz wurde durch Maßnahmen zur räumlichen Trennung redundanter Sicherheitssysteme, zusätzliche Gebäude (z.B. Gebäude für Speisewasserbehälter und Anfahrentspanner - GESA) sowie durch lokale Verbunkerung von Sicherheitssystemen erreicht.

3.3.3 Spannbetonreaktordruckbehälter ohne Sicherheitsventil?

Die Überprüfung aller anzunehmenden Störfälle im Hinblick auf den maximal~~en~~ Behälterinnendruck im Rahmen einer Zuverlässigkeitsanalyse für die Dampferzeugerabsperrarmaturen hat ergeben, dass der sich einstellende Störfalldruck in jedem der untersuchten Fälle unter dem Prüfdruck des Behälters (46 bar) lag (vgl. Bescheid Nr. 7/2).

Es wurde auch der hypothetische Fall des Versagens aller Frischdampf-Systemrohrleitungen eines der insgesamt 6 Dampferzeuger mit gleichzeitigem Versagen der automatischen Lokalisierung des undichten Dampferzeugers über die Feuchtemessung und unbegrenztem Einströmen von Dampf bzw. Wasser in den Spannbetondruckbehälter untersucht. Hierbei ergab sich, dass die Grenztragfähigkeit des Behälters noch bei weitem nicht erreicht wird und der maximale Druck im Behälter infolge Kondensation des einströmenden Dampfes um ca. 40% der Differenz zwischen dem Betriebsdruck und dem Druck im nachgewiesenen Grenzzustand der Linerintegrität unter dem Druck im Grenzzustand der Linerintegrität bleibt. Auf ein Sicherheitsventil konnte somit verzichtet werden (vgl. Bescheid Nr. 7/8c).

3.4 Beherrschung der Einwirkungen von außen (insbesondere Explosionsdruckwelle und Flugzeugabsturz u.ä.)?

Von den verschärften Anforderungen zum Schutze gegen Einwirkungen von außen waren ab Mitte der 70er Jahre alle neuen Kernkraftwerke betroffen. Für das Hersteller-Konsortium des THTR wirkte sich erschwerend aus, dass diese Anforderungen noch nachträglich in einem vor Jahren geplanten Projekt realisiert werden mussten. Ohne Realisierung der verschärften Anforderungen wäre wohl die Betriebsgenehmigung in Frage gestellt gewesen; im Falle einer Klage hätte die Genehmigung vor Gericht wohl kaum Bestand erlangt. Maßnahmen mussten insbesondere zur Stabilisierung der Reaktorhalle und zur räumlichen Trennung der in der Reaktorhalle bzw. im Maschinenhaus befindlichen Reaktorabschalt- bzw. Nachwärmeabfuhrsysteme getroffen werden. Die Notstromversorgungssysteme wurden entmascht (kein "Hosenbein") und erweitert (von 3 auf 4 x 100% der erforderlichen Leistung) sowie durchgängig räumlich getrennt. Die Bühnen der Reaktorhalle wurden im Verbund mit dem Spannbetonbehälter zur Sicherung der Halle gegen Torsion infolge Flugzeugabsturz auf eine Rahmenstütze ausgesteift und an den Lisenen des Spannbetonbehälters arretiert.

Anforderungen zum Schutze gegen Auswirkungen infolge Erdbeben waren seit der 1.TEG bekannt; auf Grund verfeinerter Berechnungsverfahren mussten im Einzelfall zusätzliche Maßnahmen ergriffen werden.

3.5 Beherrschung des Einzelfehler-Kriteriums?

Die Erfüllung dieser Anforderung hat zu einer Erweiterung der Redundanz in Teilen der Sicherheitssysteme und zu ihrer Entmaschung geführt (z.B. bei den Notkühlsystemen und der Notstromversorgung). Der Aufwand für Regelung und Steuerung ist trotzdem beachtlich geblieben, da die Nachwärme stets über Dampferzeuger/Gebläse-Einheiten abgeführt werden musste (1 Einheit bei Reaktor unter Betriebsdruck, 2 Einheiten bei Reaktor drucklos).

4. Abschließende Bemerkungen/Feststellungen

Über den HTR ist viel theoretisiert und geschrieben worden, dennoch kam er nur schwer zum Zug und nur wenige haben einen gebaut.

Vom AVR zum THTR war ein großer Schritt. Der Aufwand für Steuerung und Regelung beim THTR war beachtlich.

Die Forderung nach sinngemäßer Anwendung von aus der LWR-Verfahrenstechnik abgeleiteten Sicherheitskriterien war ein weites Feld. Es sind HTR-verfahrensspezifische Technische Regeln erforderlich, die längere Betriebserfahrungen voraussetzen.

// Bundesanzeiger

Herausgegeben vom Bundesminister der Justiz

Jahrgang 42 | Ausgegeben am Sonnabend, dem 28. April 1990 | Nummer 81

Empfehlung der Reaktor-Sicherheitskommission auf ihrer 250. Sitzung am 24. Januar 1990

Empfehlung zum Sicherheitskonzept einer Hochtemperatur-Modul-Kraftwerksanlage

Inhaltsübersicht

1 Einleitung

Das HTR-Modul-Kraftwerkskonzept der Projektgemeinschaft Siemens AG — Interatom GmbH ist dadurch gekennzeichnet, daß standardisierte nukleare Wärmeerzeugungseinheiten von 200 MJ/s thermischer Leistung zu Kraftwerken zusammengeschaltet werden können. Die HTR-Modul-Kraftwerksanlage ist zur kombinierten Erzeugung von elektrischer Energie und Prozeßdampf oder Fernwärme geeignet.

Die RSK hat aus Anlaß einer im Land Niedersachsen beantragten Genehmigung nach § 7a AtG zur Erteilung eines standortunabhängigen Konzeptvorbescheides für eine HTR-Modul-Kraftwerksanlage die für einen Konzeptvorbescheid relevanten sicherheitstechnischen Fragen beraten. Der Beratung sollten bisher übliche Standorteigenschaften zugrundegelegt werden. Die Prüfung davon abweichender Standorte, z. B. an Industriestandorten oder in Ballungsgebieten, bleibt einer Beratung in einem Verfahren nach Vorliegen eines Antrags für einen konkreten Standort vorbehalten. Das Genehmigungsverfahren ist im April 1989 eingestellt worden. Die Konzeptbegutachtung wurde weitergeführt. Die RSK hat bei ihren Beratungen das Gutachten des TÜV Hannover (Entwurf, Stand 8/89) berücksichtigt. Spätere Ergänzungen sind bei den weiteren Beratungen der RSK eingeflossen. Der BMU hat die RSK gebeten, eine Empfehlung zum Sicherheitskonzept abzugeben. Die RSK stellt fest, daß diese Empfehlung eine spätere Beratung und Empfehlung im Rahmen eines Genehmigungsverfahrens für ein konkretes Projekt nicht ersetzt.

2 Nukleares Wärmeerzeugungssystem

2.1 Primärsystem

Die HTR-Modul-Kraftwerksanlage besteht aus 2 nuklearen 2-Kreis-Dampferzeugungssystemen von je 200 MJ/s thermischer Leistung. Die beiden Moduleinheiten befinden sich in einem gemeinsamen Reaktorgebäude.

Die Systemelemente des nuklearen Wärmeerzeugungssystems zur Erzeugung von Hochtemperatur-Frischdampf bzw. Prozeßdampf sind:

- Der Reaktor in einem Stahldruckbehälter, Kühlung des Cores durch Helium von oben nach unten strömend, sphärische Brennelemente mit 15 Durchläufen durch das Core, Abschalteinrichtungen im Reflektor mit 6 Abschaltstäben zur Heißabschaltung und Regelung und 18 Kleinkugel-Absorbersysteme zur Langzeitabschaltung.
- der Dampferzeuger mit Aufwärtsverdampfung in einem separaten Stahldruckbehälter,
- das Gebläse zur Umwälzung des Heliums, am Dampferzeugerbehälter angeflanscht,
- der Verbindungsdruckbehälter mit koaxialer Heißgas/Kaltgasführung.

Der Reaktordruckbehälter und der Dampferzeuger je einer Einheit befinden sich in einer unterteilten Stahlbetonzelle.

Die Hauptdaten des nuklearen Wärmeerzeugungssystems des HTR-Modul mit Dampferzeuger sind:

Thermische Leistung	MJ/s	200
Mittlere Leistungsdichte des Reaktorkerns	MJ/s · m³	3,0
Heliumdurchsatz, primär	kg/s	85
Primärkreisdruck	bar	60
Eintrittstemperatur Reaktor	°C	ca. 250
Austrittstemperatur Reaktor	°C	700
Frischdampfdruck	bar	190
Frischdampftemperatur	°C	530
Dampfmenge	kg/s	77
Speisewassertemperatur	°C	ca. 170

Reaktordruckbehälter, Verbindungsdruckbehälter und Dampferzeugerdruckbehälter sind hinsichtlich ihrer Abmessungen, Wandstärken, Abstützungen sowie Betriebs- und Auslegungsdaten (p, T) mit den Komponenten moderner Leichtwasserreaktoren vergleichbar. Eine Ausnahme bildet der Frischdampfstutzen am Dampferzeuger, da der Frischdampfzustand dem von Hochtemperaturreaktoren bzw. konventionellen Dampfkraftwerken entspricht.

2.2 Brennelement

Das der Auslegung des HTR-Moduls zugrunde gelegte kugelförmige Brennelement entspricht dem im Projekt „Hochtemperaturreaktor-Brennstoff-Kreislauf (HBK)“ entwickelten Element. Es hat einen Durchmesser von 6 cm. Die innere Kugelzone von 5 cm Durchmesser enthält den Brennstoff (7 g niedrig angereichertes Uran (LEU) pro Brennelement, Anreicherung 8,0 +/— 0,5%) in Form beschichteter Brennstoffteilchen. Die 5 mm dicke äußere Kugelzone ist brennstofffrei. Der UO_2-Brennstoff in Form sphärischer Teilchen von 500 µm Durchmesser ist von einer sog. TRISO-Beschichtung (Pyrokohlenstoff und Siliziumcarbid) umschlossen. Diese beschichteten Brennstoffteilchen mit einem äußeren Durchmesser von 920 µm sind in dem auf Basis einer Graphitmatrix gepreßten Brennelement eingebettet.

Die Spaltproduktfreisetzung aus diesen Brennelementen ist beim vorgelegten Konzept sehr gering. Dies wurde durch zahlreiche Bestrahlungsversuche und ergänzende Ausheiztests an bestrahlten Brennelementen nachgewiesen. Aus diesen Experimenten lassen sich folgende Freisetzungsmechanismen temperaturgestuft ableiten:

— bis ca. 1200° C Freisetzung allein aus

	Auslegungswerte (Anteile)
· herstellungsbedingtem Partikelbruch	$6 \cdot 10^{-5}$
· bestrahlungsbedingtem Partikelbruch	$1 \cdot 10^{-4}$ (bei mittl. Abbrand des Modulcores) $2 \cdot 10^{-4}$ (bei Zielabbrand eines Brennelementes)

Der Auslegungswert für den herstellungsbedingten Partikelbruch berücksichtigt auch den Urangehalt der Graphitmatrix (Natururan, herstellungsbedingte Kontamination). Dieser Urangehalt wird auf einen Anteil von $7 \cdot 10^{-7}$ am Gesamtinventar an Uran-235 des Brennelementes begrenzt.

– Oberhalb ca. 1200° C

Einsetzender, noch sehr geringer Transport der metallischen Spaltprodukte aus intakten Partikeln.

– Oberhalb des Bereiches von 1600° C – 1650° C

Einsetzende Spaltprodukt-Korrosion der SiC-Schicht

Nach Ansicht der RSK ist die sicherheitstechnische Auslegung des HTR-Modul geprägt durch die geringe Aktivitätsfreisetzung aus den Brennelementen während des bestimmungsgemäßen Betriebes und bei Störfällen.

Der Hersteller hat die Auslegung des Reaktors durch die Wahl der niedrigen Leistung und Leistungsdichte sowie einer günstigen Coregeometrie konservativ auf eine Begrenzung der maximalen Brennstofftemperatur auf 1620° C abgestimmt. Während des Normalbetriebs erreichen die Brennelemente nur eine Maximaltemperatur von ca. 850° C.

Die RSK geht auf Grund von Analysen des Instituts für Nukleare Sicherheitsforschung (ISF) der KFA Jülich davon aus, daß bei Störfällen und sogar bei Unfällen nur bei wenigen Prozent der Brennelemente eine Temperatur von 1500° C für eine Zeit von wenigen Tagen überschritten wird, wobei der Auslegungswert von 1620° C nicht erreicht wird. Quelltermanalysen des ISF zeigen, daß nur eine geringe Jodmenge in den Primärkreis freigesetzt wird, die fast ausschließlich aus dem bereits zu Störfallbeginn außerhalb intakter Partikeln vorhandenen Inventar stammt; da ein effektiver Transportmechanismus in die Umgebung fehlt, verbleibt das Jod weitgehend im Primärkreis. Aus Partikeln diffundiertes Cäsium und Strontium wird nahezu vollständig am Coregraphit zurückgehalten.

2.3 Abschalteinrichtungen und Reaktorregelung

Zur Reaktorregelung und zur Heißabschaltung des Kerns dienen 6 Reflektorstäbe, deren Ausführung im wesentlichen der THTR-Konstruktion entspricht. Bei Heißabschaltung wird die elektrische Versorgung des Reflektorstabantriebsmotors unterbrochen, wodurch der Stab unter Schwerkraft in seine tiefste Stellung (1 m unter Kernmitte) einfällt.

Das Kleinkugel-Absorbersystem (KLAK-System) stellt die Kaltabschaltung des Kerns sicher. Die 18 KLAK-Behälter sind oberhalb des thermischen Deckenschildes angeordnet. Zur Auslösung wird die Versorgung eines Haltemagneten unterbrochen, wodurch sich der Behälterverschluß unter Schwerkraft öffnet und die Kleinkugel-Absorber frei in Reflektorbohrungen einfallen. Als Behälterverschluß dient ein sogenannter Stauverschluß, der, ohne Kugelbruch zu erzeugen, auch die Eingabe von Teilmengen ermöglicht.

Mittels einer pneumatischen Fördereinrichtung können die Absorber dosiert wieder in die Vorratsbehälter zurückgeführt werden, in denen Meßeinrichtungen den Füllstand kontrollieren. Schaltelemente und Fördereinrichtungen sind außerhalb des Reaktordruckbehälters angeordnet und auch bei Reaktorbetrieb zugänglich.

Der Reaktorkern ist für einen uneingeschränkten Lastwechselbetrieb zwischen 100% und 50% Nennleistung ausgelegt. Der dafür erforderliche Reaktivitätsbedarf beträgt 1,2% Δk. Der Gleichgewichtskern hat eine maximale Überschußreaktivität von 7,8% Δk in kaltem unvergiftetem Zustand. Die Abschalteinrichtungen (Reflektorstäbe und KLAK-System) gewährleisten unter Berücksichtigung eines Einzelfehlers eine Abschaltreaktivität von mindestens 10,6% Δk. Der Gesamttemperaturkoeffizient der Reaktivität ist immer negativ.

Bei Schnellabschaltung fallen alle Reflektorstäbe in ihre tiefste Position im Seitenreflektor, und der primär- und sekundärseitige Massenstrom wird durch Abschaltung des Gebläses und Schließen aller sekundärseitigen Absperrarmaturen unterbrochen.

Die RSK ist der Ansicht, daß die für den HTR-Modul vorgesehenen Abschalteinrichtungen, die Reaktorregelung und das Reaktorschutzsystem (s. 2.4) grundsätzlich in der Lage sind, die sicherheitstechnischen Aufgaben zu erfüllen.

Die zur Heißabschaltung dienenden Reflektorstäbe haben bei Ausfall eines Stabes eine minimale Wirksamkeit von 2,6% Δk. Dieser Reaktivitätsbetrag reicht aus, den Reaktor selbst im ungünstigsten Reaktivitätsstörfall (Stabausfahren, Wassereinbruch) sicher unterkritisch zu machen.

Durch die Schnellabschaltung wird der Reaktor in einen heiß unterkritischen Zustand gebracht, der größere thermische Beanspruchungen an den Komponenten verhindert und im übrigen ein schnelles Wiederanfahren ermöglicht.

Zur Langzeitabschaltung ist die Betätigung des Kleinkugel-Absorbersystems (KLAK) vorgesehen. Eine Langzeitabschaltung aus Sicherheitsgründen ist frühestens nach einigen Tagen erforderlich. Die Auslösung des KLAK-Systems nach dem Ruhestromprinzip stellt eine sehr zuverlässige Einrichtung dar.

Die RSK stellt fest, daß die Wirksamkeit der Reflektorstäbe und des KLAK-Systems ausreicht, den Reaktor sicher in einen unterkritischen Zustand zu überführen und darin zu halten. Ein zusätzliches inhärentes Merkmal des HTR-Moduls ist es, daß der Reaktor bei unterstelltem Ausfall der Abschalteinrichtungen allein durch die Gebläseabschaltung von selbst zunächst unterkritisch wird. Längerfristig stellt sich ein niedriges Leistungsniveau von 0,5% der Anfangsleistung ein, bei dem sich der Reaktor bei einem gegenüber dem bestimmungsgemäßen Betrieb höheren Temperaturniveau der Brennelemente, aber weit unter 1600° C stabilisiert.

2.4 Reaktorschutzsystem

Jede Moduleinheit besitzt ihr eigenes, nur ihr zugeordnetes Reaktorschutzsystem. Eine Rückkopplung bzw. Beeinflussung anderer Moduleinheiten ist damit ausgeschlossen. Die Autonomie des Reaktorschutzsystems beginnt auf der Meßwerterfassungsebene und bleibt bis zur Ansteuerung der aktiven Sicherheitseinrichtungen erhalten. Bei Störfällen wird nur die betroffene Moduleinheit abgeschaltet und nur an dieser werden evtl. weitere erforderliche Schutzaktionen eingeleitet. Störfälle, die sich auf beide Moduleinheiten auswirken, werden von jedem Reaktorschutzsystem separat erkannt.

Die zur Überwachung erforderlichen Prozeßvariablen lassen sich auf 8 Meßgrößen zurückführen:

- Neutronenfluß
- Reaktoraustrittstemperatur
- Reaktoreintrittstemperatur
- Feuchte im Primärsystem
- Druck im Primärsystem
- Druck im Sekundärsystem
- Massendurchsatz im Primärsystem
- Speisewasserdurchsatz

Die Meßwerterfassung und der übrige Analogteil des Reaktorschutzsystems ist grundsätzlich dreifach redundant aufgebaut. Im anschließenden Logikteil erfolgt die logische Wertung der aus den Meßgrößen abgeleiteten Anregekriterien und die Bildung der Auslösesignale zur Ansteuerung der Schutzaktionen:

- Einfall der Reaktorstäbe
- Abschalten des Primärkreislaufgebläses
- Absperren des Dampferzeugers
- Dampferzeugerentlastung (bei Feuchtdetektion)
- Primärkreisabschluß (bei Druckentlastung)

Die Ansteuerung erfolgt nach dem Ruhestromprinzip. Das gesamte Reaktorschutzsystem ist „fail safe" ausgelegt. Die erforderlichen Stellenergien sind an den steuernden Komponenten gespeichert. Nach der Auslösung hat der Reaktorschutz keine Funktion mehr zu erfüllen.

Die RSK erwartet auch für die Realisierung des Reaktorschutzsystems für den HTR-Modul keine prinzipiellen Schwierigkeiten, da es einerseits ganz nach dem „fail safe"-Prinzip aufgebaut ist und sich andererseits die gesamte Meßwertaufbereitung und logische Verknüpfung der Sicherheitsvariablen auf die bei LWR-Anlagen bewährte Technik des nach den RSK-Leitlinien geforderten Reaktorschutzsystems stützt.

2.5 Elektrische Energieversorgung

Die gemeinsame Eigenbedarfsanlage der HTR-Modul-Kraftwerksanlage mit 2 Moduleinheiten ist zweisträngig aufgebaut. Sie kann von einem Blockgenerator oder aus dem Netz versorgt werden. Die Anbindung an das Netz erfolgt über einen Hauptnetzanschluß und einen Reservenetzanschluß. Durch die Zuordnung der Moduleinheiten zu den Eigenbedarfsschienen wird sichergestellt, daß der Ausfall einer Schiene nur zum Ausfall einer Moduleinheit führen kann.

Sicherheitstechnisch wichtige Verbraucher sind an das für beide Moduleinheiten gemeinsame zweisträngige Notstromsystem angeschlossen. Jeder Notstromstrang verfügt über eine eigene Dieselnotstromerzeugungsanlage und kann die Mindestversorgung für beide Module gewährleisten.

Die Gleichspannungsversorgung erfolgt pro Strang aus einem Gleichrichter mit parallelgeschalteter 220-V-Batterie und nachgeschalteten Gleichspannungswandlern zur Versorgung der 24-V-Verbraucher.

Die 220-V-Batterien sind für eine zweistündige Entladung bemessen. Bei einer Unterschreitung der zulässigen Spannung werden alle 24-V-Verbraucher (betriebliche und sicherheitsrelevante Leittechnik) automatisch abgeschaltet und damit auch die Schutzaktionen ausgelöst.

Für die Notsteuerstelle ist eine einsträngige Energieversorgung mit einer eigenen 24-V-Batterie vorhanden. Diese Batterie ist für eine Entladungszeit von 15 Stunden ausgelegt.

Das Notstromsystem wird nach dem derzeitigen Planungsstand nicht durchgehend als Sicherheitssystem qualifiziert. Dies entspricht dem Sicherheitskonzept der Gesamtanlage, das einen Totalausfall der elektrischen Energieversorgung — und damit auch der Kühlung — für eine Dauer von bis zu 15 Stunden ohne Überschreitung von Auslegungswerten zuläßt.

Die RSK hat keine Bedenken gegen das Konzept der elektrischen Energieversorgung. Sie geht dabei davon aus, daß durch entsprechende Auslegung und qualitätssichernde Maßnahmen Ausfalldauern über 15 Stunden vermieden werden. Bei einem Ausfall der betrieblichen und sicherheitsrelevanten Leittechnik kann die Anlage von der Notsteuerstelle ausreichend überwacht werden.

2.6 Nachwärmeabfuhrsystem

Im bestimmungsgemäßen Betrieb und bei Störfällen wird der Wasser/Dampf-Kreislauf auch zur Nachwärmeabfuhr eingesetzt. Steht dieser nicht zur Verfügung, erfolgt beim HTR-Modul die Nachwärmeabfuhr über Wärmeleitung, Wärmestrahlung sowie über Naturkonvektion passiv an die außerhalb des Reaktordruckbehälters angeordneten Flächenkühler. Eine maximale Brennelementtemperatur von 1620° C wird sowohl bei allen Störfallereignissen mit auslegungsgemäßer Nachwärmeabfuhr als auch bei zusätzlichem Ausfall der Nachwärmeabfuhr über die Flächenkühler nicht überschritten. Die Einhaltung dieser maximalen Brennelementtemperatur ist ein inhärentes Sicherheitsmerkmal dieses Reaktorkonzepts.

Der Flächenkühler umgibt den Reaktordruckbehälter in der Reaktorzelle im Bereich des Kerns in einem Abstand von ca. 1,5 m Er ist vor der Betonwand der Zelle installiert. Der Flächenkühler dient dem Schutz der Betonstrukturen vor unzulässig hohen Temperaturen. Daneben stellt er im Normalbetrieb die Wärmesenke für die aus dem Reaktor gelangende Verlustwärme dar. Bei Nichtverfügbarkeit des Hauptkühlsystems begrenzt der Flächenkühler die Temperaturen der Reaktorstrukturen, insbesondere des Reaktordruckbehälters, auf die Auslegungstemperaturen.

Der Flächenkühler jeder Moduleinheit ist dreisträngig aufgebaut, wobei die Redundanz dadurch erreicht wird, daß jeweils drei nebeneinander liegende Kühlrohre je einem separaten Kühlkreislauf zugeordnet sind. Zwei Stränge des Flächenkühlers werden über das zweisträngige notstromgesicherte Zwischenkühlwassersystem gespeist, der dritte Strang vom betrieblichen Zwischenkühlsystem. Zur Versorgung der Flächenkühler mit Kühlwasser nach Flugzeugabsturz und Explosionsdruckwelle sind im Reaktorgebäude Schlauchanschlüsse für eine externe Einspeisung in das gesicherte Zwischenkühlsystem vorgesehen.

Die RSK ist der Ansicht, daß die wärmetechnische Auslegung des Flächenkühlsystems ausreichend ist. Die RSK geht davon aus, daß das Flächenkühlsystem mit seinem dreisträngigen Aufbau und der schaltungsmäßig einfachen Ausführung ausreichend zuverlässig ausgebildet werden kann, um die Wärmeabfuhr bei Störfällen sicherzustellen.

2.7 Aktivitätseinschluß

Das HTR-Modulkonzept verzichtet auf einen gasdichten Sicherheitseinschluß. Es beruht darauf, daß der zuverlässige Einschluß der radioaktiven Spaltprodukte in den Brennelementen so gewährleistet ist, daß die Umgebungsbelastungen bei allen Störfällen unterhalb zulässiger Grenzwerte bleiben.

Der Auslegungsdruck des Primärkreises und der Ansprechdruck des Druckentlastungssystems wurden so gewählt, daß der durch Ausfall der Hauptwärmesenke resultierende Druckanstieg nicht zum Ansprechen des Druckentlastungssystems führt. Erst ein großer Wassereinbruch führt langfristig zum Ansprechen des Druckentlastungssystems, wenn sowohl die Druckregelung als auch der Wasserabscheider in der Gasreinigungsanlage versagen. Das Druckentlastungssystem besteht aus zwei parallel angeordneten Strängen mit Abblaseventilen, die auf gestaffelte Ansprechdrücke eingestellt werden. Zur Begrenzung der an die Umgebung abgegebenen Spaltprodukte sowie zur Minimierung der Heliumverluste wird bei Absinken des Drucks auf einen Wert unter 60 bar das Druckentlastungssystem wieder geschlossen.

Das Reaktorgebäude stellt kein klassisches Volldruck-Containment dar, sondern dient bei Störfällen der gezielten Aktivitätsführung. Die Modul-Kraftwerksanlage besitzt daher einen Druckentlastungskanal, der aus der für beide Moduleinheiten gemeinsamen Reaktorhalle in den Kamin einmündet. Die Primärräume einer Moduleinheit sind außerdem untereinander durch Öffnungen verbunden, um einen möglichst raschen Druckausgleich zu erreichen. Nach Erreichen des Ausgleichsdrucks von 1 bar schließen die Druckentlastungsklappen selbsttätig und eine gerichtete Luftführung im Gebäude wird wieder hergestellt. Die Entlastungskanäle sind außerdem mit je einer fernbedienbar schließenden Klappe versehen.

Die Räume im Reaktorgebäude sind lüftungstechnisch zur gezielten Abgabe radioaktiver Spaltprodukte ausgelegt. Primärkreisleckagen bis zu einer Leckgröße von 2 cm^2 (beim Abriß einer Meßleitung DN 10 oder dem Ansprechen des kleinen Sicherheitsventils des Druckentlastungssystems) können vom Lüftungssystem aufgefangen und gefiltert werden.

Die RSK hat keine sicherheitstechnischen Bedenken gegen das Konzept des Aktivitätseinschlusses. Dieses ist geeignet, die Einhaltung der Vorschriften der Strahlenschutzverordnung für den bestimmungsgemäßen Betrieb und für Auslegungsstörfälle zu gewährleisten.

2.8 Notsteuerstelle

Für die 2 Moduleinheiten ist eine Notsteuerstelle vorgesehen. Sie wird für folgende Fälle benötigt:

— Ausfall der Warte
— langfristiger Eigenbedarfsausfall und Ausfall der Notstromdiesel
— Explosionsdruckwelle
— Flugzeugabsturz

Die Notsteuerstelle ist mit folgenden Einrichtungen ausgestattet:

— Störfallinstrumentierung
— Auslösung des Abschaltsystems KLAK
— Kommunikationseinrichtungen

Die Notsteuerstelle hat eine Notstrombatterie für 15 Stunden und einen Anschluß zur Noteinspeisung von elektrischer Energie nach 15 Stunden. Die Notsteuerstelle ist einsträngig aufgebaut. Dies ist nach Ansicht des Antragstellers ausreichend in Verbindung mit der „fail safe"-Ausführung der Reaktorschutzaktionen und des Abschaltsystems KLAK. In den oben genannten Fällen ist die Nachwärmeabfuhr durch eine externe Bespeisung über Schlauchanschlüsse gewährleistet.

Die RSK hat die Ausführung der Notsteuerstelle beraten. Sie ist der Ansicht, daß für die vorgesehene Aufgabenstellung der Notsteuerstelle, die beim HTR-Modul lediglich Überwachungsfunktionen und die von Hand auszulösende Langzeitabschaltung (KLAK-System) in „fail-safe"-Ausführung vorsieht, eine einsträngige Ausführung ausreichend ist. Auf der Grundlage des derzeitigen Planungsstandes ist nicht erkennbar, daß weitere Steuerungsfunktionen erforderlich werden könnten.

3 Auslegung und Qualitätssicherung der druckführenden metallischen Komponenten des Primärsystems und des Sekundärkreislaufs

3.1 Auslegung und Qualitätssicherung des Primärkreislaufs

Die Druckbehältereinheit besteht aus dem Reaktor- und dem Dampferzeuger-Druckbehälter sowie dem Verbindungsdruckbehälter. Die zylindrischen Teile der Behälter sind aus geschmiedeten Ringen hergestellt, die durch Schweißnähte miteinander verbunden sind. Der Reaktordruckbehälter ist durch einen Behälterdeckel verschlossen, der nach Demontage den vollen Behälterquerschnitt freigibt.

Für die Schmiedeteile der Druckbehälter ist der Werkstoff 20 MnMoNi 55 vorgesehen. Dieser auch für die Herstellung von Reaktordruckbehältern, Dampferzeugern und Rohrleitungen von Leichtwasserreaktoren verwendete Stahl zeichnet sich insbesondere durch seine hohe Zähigkeit und durch die Schweißsicherheit bei der Fertigung aus. Die Erfahrungen im Leichtwasserreaktorbau zeigen, daß die Schweißnähte mit Hilfe von Ultraschall sicher prüfbar sind.

Durch die Heißgasführung im Reaktordruckbehälter (Einschluß des Reaktorkerns in einem Kernbehälter), Verbindungsdruckbehälter und Dampferzeugerdruckbehälter und die Maßnahmen zur Wärmedämmung zwischen der Heißgas- und der Kaltgasseite wird nach Ansicht des Antragstellers sichergestellt, daß die Druckbehälterwände nicht mit Heißgas beaufschlagt werden können. Unter dieser Voraussetzung können die Auslegungsmerkmale von LWR-Druckbehältern auf die des HTR-Moduls übertragen werden. Zwischen Kernbehälter und Reaktordruckbehälter besteht eine lose aufgelagerte Schiebeverbindung. Leckagen zwischen der Kaltgasseite und der Heißgasseite sind durch die Druckverhältnisse vom Kaltgas zum Heißgas gerichtet.

Die Spezifikation und Auslegung der Druckbehältereinheit basieren auf dem anerkannten Sicherheitskonzept für Leichtwasserreaktoren. Aufgrund der vergleichbaren technischen Basis wird für die Druckbehältereinheit des HTR-Moduls das Konzept der Basissicherheit erfüllt und durch wiederkehrende Prüfungen abgesichert. Der Frischdampfstutzen gehört bis zum Anschluß der Frischdampfleitung zur Druckbehältereinheit. Er wird ebenfalls so ausgelegt, daß ein Bruchausschluß angesetzt werden kann.

Für die Bauteile und Komponenten der Druckbehältereinheit werden die Anforderungen der KTA 3201, Teile 1 bis 4 im wesentlichen erfüllt. Spezielle Spannungsanalysen wurden für Flanschverbindungen am Reaktordruckbehälter und am Dampferzeugerdruckbehälter und den Boden-Zylinderanschluß am Reaktordruckbehälter durchgeführt. Dabei wurden die Methoden Stufenkörpermethode und Schalentheorie angewandt. Verstärkungen bei Stutzenkonstruktionen wurden vorzugsweise an den Druckbehältern vorgenommen.

Zum Zweck wiederkehrender Prüfung müssen Isolierungen entfernt werden. Für die Prüfung selbst werden die entsprechenden Vorkehrungen (z. B. Manipulatorsysteme für Ultraschallprüfungen) von der Leichtwasserreaktortechnik übernommen.

Bezüglich wiederkehrender Prüfungen ist für den Reaktordruckbehälter und den Dampferzeuger-Druckbehälter ein 4jähriger Prüfungszyklus vorgesehen, wobei die Schweißnähte der Druckbehältereinheit vollvolumetrisch geprüft werden. Wiederholungsdruckprüfungen sind systembedingt nur mit Gas möglich, wobei der Prüfdruck durch die gültige Regel für Gasdruckprüfungen auf das 1,1-fache des Auslegungsdrucks beschränkt ist. Die Erstdruckprüfung wird mit Wasser beim 1,3-fachen Auslegungsdruck durchgeführt. Vor Wiederholungsdruckprüfungen wird eine Ultraschall-Wiederholungsprüfung durchgeführt. Stellen, an denen Befunde vorliegen, werden nach der Wiederholungsdruckprüfung erneut einer Ultraschall-Prüfung unterzogen. Am Sekundärkreis werden Wiederholungsdruckprüfungen mit Wasser durchgeführt.

Zur Betriebsüberwachung hinsichtlich der Werkstoffe hat der Antragsteller ausgeführt, daß voreilende Bestrahlungsproben nicht möglich sind. Statt dessen ist vorgesehen, die Werkstoffveränderungen während der Lebensdauer der Anlage durch Experimente nachzuvollziehen und begleitend die jeweils neueste Literatur in Betracht zu ziehen. Zusätzlich werden dann mitlaufende Bestrahlungsproben eingesetzt und von Zeit zu Zeit untersucht.

Bruchmechanische Untersuchungen wurden durchgeführt, die zeigen, daß ein Bruchausschluß der Druckbehältereinheit angesetzt werden kann. Es wurde außerdem gezeigt, daß das Leck-vor-Bruch-Kriterium eingehalten wird.

Die RSK sieht die Übertragung der in den RSK-Leitlinien für Druckwasserreaktoren und in der KTA-Regel 3201 festgelegten Grundsätze für die Sicherheit gegen Versagen druckführender Komponenten und die Beschränkung der Annahmen von Leckgrößen auf den Querschnitt von Anschlußleitungen (DN 65) als gerechtfertigt an. Die Anforderungen des Abschnitts 4.1 der RSK-Leitlinien für Druckwasserreaktoren unter Berücksichtigung der Rahmenspezifikation Basissicherheit, der KTA-Regel 3201 und des im Vorbericht zur KTA-Regel 3221 festgehaltenen Grundlagenmaterials für Einsatztemperaturen oberhalb 400° C können hinsichtlich der Werkstoffe, ihrer Verarbeitung, der konstruktiven Gestaltung, der Spannungsbegrenzung, der Ermüdungssicherheit und der wiederkehrenden Prüfungen im wesentlichen erfüllt werden.

Die Ausbildung des unteren Bodens im Reaktordruckbehälter mit Bodenverstärkungsring und Kalotte entspricht dem unteren Boden im Reaktordruckbehälter von Siedewasserreaktoren mit Pumpenstutzen. Daher sind entsprechende Nachweise über die Spannungsbegrenzung zu führen.

Wichtig ist ein rechtzeitiges Erkennen von wesentlichen Überschreitungen der Auslegungstemperatur in einzelnen Bereichen der druckführenden Umschließung. Zum Teil können hierzu die vom Anlagenkonzept her gegebenen Möglichkeiten der Feststellung von Leckagen zwischen den Kreisläufen genutzt werden. Darüber hinaus kann durch betriebliche Messungen eine Kontrolle des durch die Kühlmittelführung bewirkten Wärmeschutzes erfolgen.

3.2 Lagerung des Reaktordruckbehälters und des Dampferzeugers

Die Lagerung des Reaktordruckbehälters und des versetzt angeordneten Dampferzeugerdruckbehälters erfolgt auf 3 Abstützebenen. Auf der unteren Abstützebene ist der Dampferzeugerdruckbehälter, auf der mittleren Ebene beide Behälter und auf der oberen Ebene der Reaktordruckbehälter gelagert, wobei unterschiedliche Konstruktionen in Verbindung mit Gleitlagern und Stoßbremsen zur Anwendung kommen. Kräfte aus Einwirkungen von außen wurden berücksichtigt.

Erfahrungen aus den USA, wo es bei Leichtwasserreaktoren zur Versprödung von Lagern kam, sind dem Antragsteller bekannt. Er schließt eine Versprödung der Stützkonstruktion beim HTR-Modul aus, da die Neutronenfluenz für einen solchen Effekt zu gering sei.

Die RSK hat keine Bedenken gegen die Lagerung des Reaktordruckbehälters und des Dampferzeugers. Sie stellt jedoch fest, daß der zuverlässigen Erhaltung der Funktionsfähigkeit der Behälterauflagerung wegen der starren Verbindung zwischen Reaktordruckbehälter und Dampferzeugerdruckbehälter eine große Bedeutung zukommt. Betriebliche Wartung, Reparaturmöglichkeit und wiederkehrende Prüfung sind daher erforderlich.

3.3 Sekundärkreislauf

Für den Sekundärkreislauf innerhalb des Reaktorgebäudes hat der Antragsteller eine Auslegung gemäß konventionellen Anforderungen mit kerntechnischen Zusatzanforderungen vorgesehen. Die Zusatzanforderungen beziehen sich auf die Sekundärkreisarmaturen, da diese zum Absperren des Dampferzeugers benötigt und vom Reaktorschutz angesteuert werden.

Außerhalb des Reaktorgebäudes wird der Sekundärkreislauf nur nach konventionellen Regeln ausgelegt. Dies wird damit begründet, daß der Sekundärkreislauf keine sicherheitstechnische Bedeutung hat, da die Nachwärme immer über den dreisträngigen Flächenkühler abgeführt werden kann.

Zum Schutz gegen Einwirkungen von Bruchstücken, die beim Versagen von Sekundärkreiskomponenten (z. B. Turbine, Behälter) auftreten können, wird der nukleare Teil des Kraftwerkes so angeordnet bzw. ausgelegt, daß keine unzulässigen Einwirkungen auftreten können.

Die wiederkehrenden Prüfungen des Sekundärkreislaufs außerhalb des Reaktorgebäudes werden entsprechend den üblichen konventionellen Anforderungen durchgeführt. Im Reaktorgebäude werden zusätzlich wiederkehrende Prüfungen entsprechend kerntechnischen Anforderungen durchgeführt.

Die RSK ist der Meinung, daß die vorgesehene konventionelle Auslegung des Sekundärkreislaufs im Grundsatz akzeptiert werden kann, daß aber für Komponenten mit hohem Energieinhalt höhere Anforderungen zu stellen sind. Bezüglich der wiederkehrenden Prüfungen hält sie eine genaue Darlegung der Durchführbarkeit unter dem Gesichtspunkt der Zugänglichkeit für erforderlich. Unter der Voraussetzung entsprechender Herstellerqualität und betrieblicher Überwachungsmaßnahmen ist es nach Ansicht der RSK gerechtfertigt, unterstellte Dampferzeugerlekkagen auf die Größe eines Heizrohrquerschnitts zu begrenzen.

4 Auslegung der Gebäude

Die Auslegung der Gebäude der HTR-Modul-Kraftwerksanlage entspricht dem Sicherheitskonzept der Gesamtanlage. Die Einwirkungen

— Erdbeben
— chemische Explosion
— Flugzeugabsturz

sind für das Konzept des Schutzes gegen Einwirkungen von außen bestimmend. Außer diesen werden die Einwirkungen

— Blitz
— Wind, Sturm
— Schnee, Regen, Hagel
— Hochwasser, Niedrigwasser
— Gefährliche Gase

berücksichtigt.

Entsprechend dem Sicherheitskonzept der Anlage ist nur für das Reaktorgebäude eine Auslegung gegen Druckwellen aus chemischen Explosionen und Flugzeugabsturz zusätzlich zur Auslegung gegen die übrigen der vorstehend genannten Einwirkungen von außen vorgesehen.

Die RSK hat keine Bedenken bezüglich der Baubarkeit der Anlage und die Erfüllbarkeit der Anforderungen bezüglich der Auslegung gegen Einwirkungen von außen. Die Anbindung des Reaktorhilfsanlagengebäudes an das Reaktorgebäude sollte nach Ansicht der RSK optimiert werden, insbesondere im Hinblick auf die Grundwasserabdichtung und die Auslegung gegen Erdbeben.

Sollten industrienahe Standorte gewählt werden, müßte geprüft werden, ob Lastfälle zu unterstellen sind, die über die bisher unterstellten hinausgehen. Ebenso können die Bemessungserdbeben erst nach Auswahl eines Standortes festgelegt werden. Dies gilt auch für eine Stellungnahme zum Baugrund.

5 Beherrschung von Auslegungsstörfällen und auslegungsüberschreitenden Ereignisabläufen

5.1 Auslegungsstörfälle

Die für die HTR-Modul-Kraftwerksanlage mit Dampferzeuger betrachteten Auslegungsstörfälle sind so festgelegt, daß sie bezüglich ihres Schadensausmaßes und ihrer Umgebungsbelastung abdeckenden Charakter haben.

Die Auslegungsstörfälle sind unter den folgenden Kategorien zusammengefaßt:

- Reaktivitätsstörfälle
- Störungen am Hauptwärmeübertragungssystem
- Primärseitige Brüche
- Sekundärseitige Brüche
- Ausfall der Stromversorgung
- Störungen an Hilfs- und Nebenanlagen
- naturbedingte Einwirkungen von außen

Zusätzlich wurden ATWS-Störfälle untersucht.

Das Vorgehen des Antragstellers bezüglich der Störfallanalysen lehnt sich an die Praxis bei Leichtwasserreaktoren an. Er geht von 4 Sicherheitsebenen aus, wobei die beiden ersten Ebenen dem Normalbetrieb und den Betriebsstörungen zugeordnet sind. Die 3. Ebene ist den Störfällen und ihrer Beherrschung und die 4. Ebene der Restrisikominderung bei seltenen Einwirkungen von außen und hypothetischen Ereignisabläufen in der Anlage zugeordnet.

Die Auslegungsstörfälle werden in Anlehnung an die Störfalleitlinien des BMI für Druckwasserreaktoren eingeteilt.

Zum Reaktorgebäude führt der Antragsteller aus, daß er ein druckfestes Reaktorgebäude aus HTR-spezifischen Gründen nicht für erforderlich hält. Beim Druckentlastungsstörfall ist die Aktivitätsableitung gering, weil die Kühlgasaktivität im bestimmungsgemäßen Betrieb gering ist und bei diesem Störfall die Integrität der Spaltproduktbarriere, der Brennelemente, erhalten bleibt. Eine relevante Aktivitätsfreisetzung in das Reaktorgebäude könnte erst nach einer physikalisch nicht möglichen unzulässigen Aufheizung des Reaktorkerns erfolgen. Daher genügt nach Ansicht des Antragstellers ein Reaktorgebäude mit gezielter Luftführung („vented confinement"). Er betont, daß bei allen Primärkreisleckagestörfällen bis hin zum Druckentlastungsstörfall die Planungsrichtwerte des § 28 Abs. 3 StrlSchV auch ohne Filterung eingehalten werden. Zur Minimierung der Ableitung ist eine Filterung des entweichenden Primärgases in der späten Phase eines Druckentlastungsstörfalls vorgesehen. Erst zu diesem Zeitpunkt ist ein langsamer und geringfügiger Anstieg der Primärkreisaktivität infolge Kernaufheizung nach Druckentlastung zu erwarten.

Beim Ausfall des Dampferzeugers wird das Gebläse abgeschaltet. Eine Gefährdung des Dampferzeugers durch heiße Gasströmung ist ausgeschlossen, da die Komponentenanordnung und Gasführung beim HTR-Modul so gestaltet sind, daß eine Naturkonvektion innerhalb des primären Kreislaufsystems, die zu einer Schädigung der metallischen Bereiche führen könnte, verhindert wird.

Zur Frage der Wärmeabfuhr hat der Antragsteller folgende Möglichkeiten genannt:

- bei Funktion des Primärkreislaufgebläses und des Dampferzeugers
  - über die Turbine
  - über Kühler im An- und Abfahrsystem
  - über den Kondensator
- bei Ausfall des Gebläses
  - über Flächenkühler (3 x 100%)
  - über die Gasreinigungsanlage

Die Gasreinigungsanlage ist 3-strängig ausgelegt, wobei ein Strang mit einem Wasserabscheider ausgestattet ist, der primär zur Entfernung von Feuchte aus dem Primärkreis bei einer Dampferzeugerleckage dient. Dieser Strang kann aber auch zur Nachwärmeabfuhr herangezogen werden.

Die RSK hat keine Bedenken bezüglich der Vollständigkeit und der Beherrschung der Störfälle.

Sie hat auch keine Bedenken gegen das Konzept des Reaktorgebäudes, aus dem bei einem Druckentlastungsstörfall Primärgas bis zum Druckausgleich über den Kamin in die Umgebung abgegeben wird. Sie stellt zum Druckentlastungsstörfall fest, daß eine Filterung des ausströmenden Primärgases zur Einhaltung der Planungsrichtwerte nach § 28 Abs. 3 StrlSchV nicht erforderlich ist. Für einen begrenzten Einsatzbereich im bestimmungsgemäßen Betrieb und bei kleinen Leckstörfällen ist eine gefilterte Ableitung vorgesehen.

Die RSK hat die Frage behandelt, ob ein Frischdampfleitungsbruch in Kombination mit Heizrohrversagen unterstellt werden muß. Sie ist der Meinung, daß diese Kombination nicht zu unterstellen ist, da die Heizrohre gegen die Belastungen bei einem Frischdampfleitungsbruch ausgelegt sind und bereits kleinste Leckagen detektiert werden.

Die vom Antragsteller vorgenommene sinngemäße Übertragung der in den Leitlinien für DWR spezifizierten Auslegungsstörfälle auf die Verhältnisse des HRT-Modul hat zu einer Auswahl von Auslegungsstörfällen geführt, die repräsentativ und vollständig sind; probabilistische Analysen des Instituts für Nukleare Sicherheitsforschung (ISF) der KFA Jülich lassen eine Grenze zu hypothetischen Ereignisketten bei ca. 10^{-6}/Reaktorbetriebsjahr erwarten.

5.2 Auslegungsüberschreitende Ereignisabläufe

Die hypothetischen Ereignisabläufe hat der Antragsteller in einem gesonderten Bericht außerhalb des Sicherheitsberichtes behandelt.

Das ISF hat eigene Untersuchungen zu hypothetischen (auslegungsüberschreitenden) Ereignisabläufen durchgeführt und den Bericht des Antragstellers zum „Verhalten des HTR-Moduls bei hypothetischen Ereignisabläufen" geprüft. Es faßt die Ergebnisse seiner Überprüfung wie folgt zusammen:

Die vom Antragsteller untersuchten Ereignisketten sind erklärtermaßen exemplarisch ausgewählt für die Problemkreise

- Ausfall aktiver Wärmeabfuhrsysteme, einschließlich des Flächenkühlsystems,
- Reaktivitätszufuhr,
- Versagen von Scram-Aktionen,
- Wassereinbrüche nach Lecks im Dampferzeuger,
- Lufteinbruch nach Lecks am Primärkreis mit Druckentlastung;

sie sind ausgerichtet auf die Möglichkeit von Schadensereignissen, die zu größeren Freisetzungen radioaktiver Stoffe führen könnten.

Die Analysen des ISF (im wesentlichen die Aktualisierung des Jül-Spez-Berichts 260: Zum Störfallverhalten des HTR-Moduls) bestätigen, daß die Ereignisketten entsprechend der Zielsetzung und mit Blick auf die wichtigsten Szenarien und Freisetzungspfade vom Antragsteller richtig ausgewählt sind.

Sie sollten allerdings durch Szenarien nach Dampferzeuger-Leck ergänzt werden, wie

- mögliche Wasser-/Wasserdampfeinbrüche aus dem zweiten Modul wegen der verbundenen Sekundärkreisläufe
- Freisetzung radioaktiver Stoffe über die Frischdampf- oder Entlastungsleitung in die Umgebung nach Versagen von Absperrorganen.

Zur Frage möglicher Wasser-/Wasserdampfeinbrüche aus dem Nachbarmodul bei einem Dampferzeuger-Leckstörfall weist der Antragsteller darauf hin, daß eine dampfseitige Absperrung des betroffenen Dampferzeugers durch zwei Armaturen und eine Rückschlagklappe erfolgt.

Ereignisketten mit Dampferzeuger-Leck und Freisetzung radioaktiver Stoffe über die Frischdampf- und Entlastungsleitung nach Versagen von Absperrorganen sind nach Ansicht des ISF wegen relativ großer Freisetzungswerte unter den auslegungsüberschreitenden Ereignisabläufen risikodominant. Freisetzungen mit großen Konsequenzen sind aber auch dafür nicht zu erwarten, da im wesentlichen nur die auf dem Dampferzeuger abgelagerte Aktivität freigesetzt werden könnte und keine Brennelemente- bzw. Partikelschäden induziert würden.

Die Behandlung der durch die hypothetischen Ereignisketten ausgelösten physikalischen Vorgänge durch den Antragsteller deckt sich mit der durch das ISF; die Ergebnisse sind plausibel und vergleichbar. Das gilt insbesondere für die Behandlung des Ausfalls der Core-Zwangskühlung hinsichtlich maximaler Temperaturen und Spaltproduktrückhaltung der Brennelemente, ebenso für die Behandlung möglicher Folgen eines Gebläseweiterlaufs.

Für den Totalausfall der Kühlsysteme einschließlich des Flächenkühlers ergeben die neueren Rechnungen des ISF, die die Wärmebindung durch Betonwasserverdampfung berücksichtigen, eine maximale Reaktordruckbehälter-Temperatur von ca. 500° C. Diese würde nach ca. 1 Woche erreicht und rund 100° C über der Auslegungstemperatur liegen: eine Gefährdung durch Versagen des Behälters ist dabei nicht gegeben.

Die Aufheizrate ist bei Ausfall des Flächenkühlsystems so gering, daß nach 15 h eine Betontemperatur von 150° C (Auslegungstemperatur) erreicht wird. Weiterhin ist die Möglichkeit vorgesehen, durch eine externe Einspeisung über Schlauchanschlüsse das Flächenkühlsystem mit Kühlwasser zu versorgen, so daß unabhängig vom Zustand der Energieversorgung und des Nebenkühlwassersystems eine Nachwärmeabfuhr möglich ist.

Die ISF-Modellierung

a) zum Austrag staubgebundener Aktivität,

b) zur Ablösung auf dem Dampferzeuger abgelagerter Aktivität durch Wasserdampf und

c) zur Freisetzung von Jod aus dem geringem Anteil (Auslegungswert $1{,}6 \cdot 10^{-4}$) bereits defekter Coated Particles infolge des Einwirkens von Wasserdampf (Brennstoff-Oxidation)

würde zu höheren Freisetzungswerten für Cäsium und Jod führen. Zur Verbesserung des Kenntnisstandes sind Forschungs- und Entwicklungsarbeiten erforderlich.

Zusammenfassend bestätigen die Arbeiten des ISF die wesentlichen Aussagen des Antragstellers zum Verhalten des HTR-Moduls bei auslegungsüberschreitenden Ereignisabläufen. Die genannten Unterschiede stellen das vorgelegte Anlagen- und Sicherheitskonzept nicht in Frage und sind kein Anlaß für Konzeptänderungen.

Die RSK schließt sich der Beurteilung der auslegungsüberschreitenden Ereignisse durch das ISF an und stellt zu den hypothetischen Ereignisabläufen zusammenfassend fest:

- Der Reaktor behält seine Standfestigkeit und Integrität
- Das Reaktorgebäude behält seine Standsicherheit und die Struktur der äußeren Hülle wird nicht zerstört.
- Das Aktivitätsinventar des Reaktors wird nur in einem solchen Maß freigesetzt, daß die Strahlendosis im Bereich der Planungsrichtwerte des § 28 Abs. 3 StrlSchV bleibt.

5.3 Einwirkungen von außen

Die Anlage wird gegen folgende Einwirkungen von außen ausgelegt:

- Erdbeben
- Blitz
- Wind, Sturm
- Schnee, Regen, Hagel
- Hochwasser, Niedrigwasser
- Gefährliche Gase.

Außerdem werden Schutzmaßnahmen zur Minderung des Risikos aus Flugzeugabsturz und Druckwellen aus externen chemischen Explosionen getroffen.

Anlagenteile, die notwendig sind, um bei Störungen in der Anlage durch Erdbeben die Schutzziele

- Abschaltung und langfristige Unterkritikalität,
- Nachwärmeabfuhr,
- Begrenzung der Aktivitätsfreisetzung

sicherzustellen, sind der Erdbebenklasse I zugeordnet und werden gegen Erdbeben ausgelegt.

Das Reaktorgebäude und die sicherheitstechnisch relevanten Anlagenteile innerhalb des Gebäudes werden gegen die Belastungen aus Flugzeugabsturz und Explosionsdruckwellen ausgelegt. Die äußeren Wände und das Dach des Reaktorgebäudes sind so bemessen, daß ein Vollschutz gegen Flugzeugabsturz erreicht wird. Die Innenstruktur ist entkoppelt von der Außenstruktur und nur über die Fundamentplatte mit der Außenstruktur verbunden. Dadurch wird gewährleistet, daß die am Auftreffort des Flugzeuges induzierten Erschütterungen nur abgeschwächt auf die Innenstruktur übertragen werden, was zu einer wesentlichen Reduzierung der Etagenantwortspektren, vor allem im hochfrequenten Bereich, führt.

Die RSK ist der Ansicht, daß die Maßnahmen zum Schutz der Anlage gegen Einwirkungen von außen den zu stellenden Anforderungen entsprechen und ohne Schwierigkeiten realisiert werden können.

6 Strahlenexposition des Personals

Die Betriebserfahrungen mit gasgekühlten Reaktoren haben gezeigt, daß die Vorschriften der Strahlenschutzverordnung bezüglich Strahlenexposition des Personals ohne Schwierigkeiten eingehalten werden konnen.

In Anbetracht der guten Zugänglichkeit kann nach Ansicht der RSK bei sorgfältiger Planung der Instandhaltungsarbeiten eine niedrige Strahlenexposition des Personals erwartet werden.

7 Zusammenfassung

Das HTR-Modul-Kraftwerkskonzept der Projektgemeinschaft Siemens AG-Interatom GmbH ist dadurch gekennzeichnet, daß mehrere standardisierte nukleare Wärmeerzeugungseinheiten von 200 MJ/s thermischer Leistung zu Kraftwerken zusammengeschaltet werden.

Die Begrenzung der Reaktorleistung auf 200 MJ/s und der mittleren Leistungsdichte auf 3 MJ/s · m³ in Verbindung mit der Coregeometrie wirkt sich insbesondere in folgender Hinsicht vorteilhaft aus:

- Zum einen wird die Brennstofftemperatur in allen Störfällen derart begrenzt, daß die Planungsrichtwerte des § 28 Abs. 3 der StrlSchV auch ohne Filterung der ggf. aus dem Reaktorgebäude entweichenden Gase aus Primär- oder Sekundärkreisleckagen eingehalten werden. Zum anderen können einfache und bereits erprobte Komponenten und Systeme für das HTR-Modul verwendet werden.
- Beim HTR-Modul erfolgt bei einem Ausfall der Hauptwärmesenke die Nachwärmeabfuhr über Wärmeleitung, Wärmestrahlung sowie über Naturkonvektion passiv an die außerhalb des Reaktorbehälters angeordneten Flächenkühler. Zur Nachwärmeabfuhr ist kein Zwangsumlauf innerhalb des Primärsystems erforderlich. Der Betrieb des Flächenkühlers dient dem Schutz von Anlagenteilen. Eine maximale Brennelementtemperatur von 1620° C wird sowohl bei allen Störfallereignissen mit auslegungsgemäßer Nachwärmeabfuhr als auch bei zusätzlichem Ausfall der Nachwärmeabfuhr über die Flächenkühler nicht überschritten. Die Einhaltung dieser maximalen Brennelementtemperatur ist ein inhärentes Sicherheitsmerkmal dieses Reaktorkonzepts.

Die RSK hat keine sicherheitstechnischen Bedenken gegen das Konzept des Aktivitätseinschlusses, da dieses auch ohne Gasdichtheit des Reaktorgebäudes prinzipiell geeignet ist, die Einhaltung der Vorschriften der Strahlenschutzverordnung für den bestimmungsgemäßen Betrieb und für Auslegungsstörfälle zu gewährleisten.

Die Wirksamkeit der Reflektorstäbe und des KLAK-Systems reicht aus, den Reaktor sicher in einen unterkritischen Zustand zu überführen und darin zu halten. Ein zusätzliches inhärentes Merkmal des HTR-Moduls ist es, daß der Reaktor bei unterstelltem Ausfall der Abschalteinrichtungen allein durch die Gebläseabschaltung von selbst zunächst unterkritisch wird. Längerfristig stellt sich ein niedriges Leistungsniveau von 0,5% der Anfangsleistung ein, bei dem der Reaktor sich bei einem gegenüber dem bestimmungsgemäßen Betrieb höheren Temperaturniveau der Brennelemente, aber weit unter 1600° C stabilisiert.

Beim Ausfall des Dampferzeugers wird das Gebläse abgeschaltet. Eine Gefährdung des Dampferzeugers durch heiße Gasströmung ist ausgeschlossen, da die Komponentenanordnung und Gasführung beim HTR-Modul so gestaltet sind, daß eine Naturkonvektion innerhalb des primären Kreislaufsystems, die zu einer Schädigung der metallischen Bereiche führen könnte, verhindert wird.

Die RSK stellt fest, daß die Grundsätze für die Sicherheit gegen Versagen druckführender Komponenten des Primärkreises erfüllt werden konnen.

Für den Totalausfall der Kühlsysteme einschließlich des Flächenkühlers ergeben die neueren Rechnungen des ISF eine maximale Reaktordruckbehälter-Temperatur von ca. 500° C. Diese würde nach ca. 1 Woche erreicht und rund 100° C über der Auslegungstemperatur liegen; eine Gefährdung durch Versagen des Behälters ist dabei nicht gegeben. Die Arbeiten des ISF bestätigen die wesentlichen Aussagen des Antragstellers zum Verhalten des HTR-Moduls bei auslegungsüberschreitenden Ereignisabläufen.

Die RSK schließt sich der Beurteilung der auslegungsüberschreitenden Ereignisse durch das ISF an und stellt zu den hypothetischen Ereignisabläufen zusammenfassend fest:

- Der Reaktor behält seine Standfestigkeit und Integrität.
- Das Reaktorgebäude behält seine Standsicherheit und die Struktur der äußeren Hülle wird nicht zerstört.
- Das Aktivitätsinventar des Reaktors wird nur in einem solchen Maß freigesetzt, daß die Strahlendosis im Bereich der Planungsrichtwerte des § 28 Abs. 3 StrlSchV bleibt.

Die RSK stellt zusammenfassend fest, daß das Konzept des HTR-Moduls dem Stand von Wissenschaft und Technik entspricht und auch im auslegungsüberschreitenden Bereich sicherheitstechnisch günstige Eigenschaften besitzt. Sie kommt zu der Aussage, daß das Konzept der HTR-Modul-Anlage geeignet ist, die sicherheitstechnischen Genehmigungsvoraussetzungen in der Bundesrepublik Deutschland zu erfüllen.

Rules and Standards for High Temperature Reactors

Dr. Ivar Kalinowski
Secretariat of Nuclear Safety Standards Commission (KTA) at BfS

1. KTA Safety Standards for High Temperature Reactors

Safety Standard	Title	Date of Issue	English Trans-lation	Reaffir med
KTA 1502.2	Monitoring Radioactivity in the Inner Atmosphere of Nuclear Power Plants; Part 2: Nuclear Power Plants with High Temperature Reactors Überwachung der Radioaktivität in der Raumluft von Kernkraftwerken; Teil 2: Kernkraftwerke mit Hochtemperaturreaktor	06/89	+	-
KTA 3102.1	Reactor Core Design for High Temperature Gas-Cooled Reactors; Part 1: Calculation of the Material Properties of Helium Auslegung der Reaktorkerne von gasgekühlten Hochtemperaturreaktoren; Teil 1: Berechnung der Helium-Stoffwerte	6/78	+	1988
KTA 3102.2	Reactor Core Design for High Temperature Gas-Cooled Reactors; Part 2: Heat Transfer in Spherical Fuel Elements Auslegung der Reaktorkerne von gasgekühlten Hochtemperaturreaktoren; Teil 2: Wärmeübergang im Kugelhaufen	6/83	+	1988
KTA 3102.3	Reactor Core Design for High Temperature Gas-Cooled Reactors; Part 3: Loss of Pressure through Friction in Pebble Bed Cores Auslegung der Reaktorkerne von gasgekühlten Hochtemperaturreaktoren; Teil 3: Reibungsdruckverlust in Kugelhaufen	3/81	+	1991
KTA 3102.4	Reactor Core Design for High Temperature Gas-Cooled Reactors; Part 4: Thermohydraulic Analytical Model for Stationary and Quasi-Stationary Conditions in Pebble Bed Cores Auslegung der Reaktorkerne von gasgekühlten Hochtemperaturreaktoren; Teil 4: Thermohydraulisches Berechnungsmodell für stationäre und quasistationäre Zustände im Kugelhaufen	11/84	+	1989
KTA 3102.5	Reactor Core Design for High Temperature Gas-Cooled Reactors; Part 5: Systematic and Statistical Errors in the Thermohydraulic Core Design of the Pebble-Bed Reactor Auslegung der Reaktorkerne von gasgekühlten Hochtemperaturreaktoren; Teil 5: Systematische und statistische Fehler bei der thermohydraulischen Kernauslegung des Kugelhaufenreaktors	6/86	+	1991

2. KTA Safety Standards for High Temperature Reactors, Unfinished Projects (in German)

Safety Standard	**Title**	
KTA 3106	Nukleare Auslegung der Reaktorkerne von gasgekühlten Hochtemperaturreaktoren und Ermittlung der Abschaltreaktivität Design of Reactor Cores and Determination of Shutdown Reactivity with Gas-Cooled HTR's	Abschlußbericht 12/1992 Final report 12/92
	Einführung zum Regelvorhaben KTA 3221 "Metallische HTR-Komponenten" Introduction to the Project on Standard KTA 3221	3/1993
KTA 3221.1	Metallische HTR-Komponenten Teil 1: Herstellung von Werkstoffen und Erzeugnisformen Metallic HTR-Components; Part 1: Manufacture of Materials and Product Forms	Regelentwurfs-vorschlag 12/92 Draft safety standard proposal 12/1992
KTA 3221.2	Metallische HTR-Komponenten Teil 2: Auslegung, Konstruktion und Berechnung Metallic HTR-Components; Part 2: Design, Construction and Analysis	Regelentwurfs-vorschlag 12/92 Draft safety standard proposal 12/1992
KTA 3221.3	Metallische HTR-Komponenten Teil 3: Herstellung von Komponenten Metallic HTR-Components; Part 3: Manufacture of Components	Regelentwurfs-vorschlag 12/92 Draft safety standard proposal 12/1992
KTA 3221.4	Metallische HTR-Komponenten Teil 4: Betriebliche Überwachung und wiederkehrende Prüfung Metallic HTR-Components; Part 4: Operational Monitoring and Inservice Inspections	---
KTA 3221.5	Metallische HTR-Komponenten Teil 5: Zusätzliche Anforderungen für Reaktordruckbehälter aus Stahl von HTR, die nach KTA 3201 ausgelegt und gefertigt werden Metallic HTR-Components; Part 5: Additional Requirements for Reactor Pressure Steel Vessels for HTR which were designed and manufactured according to KTA 3201	---
KTA 3231	Sicherheitstechnische Anforderungen an die Auslegung von Spannbeton-Reaktordruckbehältern für Hochtemperaturreaktoren Safety Requirements for the Design of Prestressed Concrete Reactor Vessels of HTR's	Vorbericht 4/89 First report 4/1989
KTA 3232	Keramische Einbauten in HTR-Reaktordruckbehältern HTR Ceramic Pressure Vessels Internals	Regelentwurfs-vorschlag 12/92 Draft safety standard Proposal 12/1992

3. Sicherheitskriterien für Anlagen zur Energieerzeugung mit gasgekühlten Hochtemperaturreaktoren, Entwurf 9/80, erstellt für den Bundesminister des Inneren (see presentation Hofmann)

SAFETY STANDARDS

of the
Nuclear Safety Standards Commission (KTA)

Monitoring Radioactivity in the Inner
Atmosphere of Nuclear Power Plants
Part 2: Nuclear Power Plants with
High Temperature Reactor

(Überwachung der Radioaktivität in
der Raumluft von Kernkraftwerken
Teil 2: Kernkraftwerke mit Hoch-
temperaturreaktor)

KTA 1502.2 (Juni, 1989)

Editor:
Gesellschaft für Reaktorsicherheit (GRS) mbH
Schwertnergasse 1 · D-5000 Köln 1
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TRANSLATION – ORIGINAL : GERMAN

KTA SAFETY STANDARD

ISSUED 6/78	Reactor Core Design for High-Temperature Gas-Cooled Reactor; Part 1: Calculation of the Material Properties of Helium (Auslegung der Reaktorkerne von gasgekühlten Hochtemperaturreaktoren, Teil 1: Berechnung der Helium-Stoffwerte)	KTA 3102.1

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– KERNTECHNISCHER AUSSCHUSS (KTA) –

FEDERAL REPUBLIC OF GERMANY

December 1978

SAFETY STANDARDS

of the

Nuclear Safety Standards Commission (KTA)

Reactor Core Design of High-Temperature
Gas-Cooled Reactors
Part 2: Heat Transfer in Spherical Fuel
Elements

(Auslegung der Reaktorkerne von
gasgekühlten Hochtemperaturreaktoren
Teil 2: Wärmeübergang im Kugelhaufen)

KTA 3102.2 (June, 1983)

Editor:

KTA-Geschäftsstelle c/o
Gesellschaft für Reaktorsicherheit (GRS) mbH
Schwertnergasse 1 · D–5000 Köln 1 · F.R.G.
Phone (0221) 2068-0 · Teletex 2214123 grs d

SAFETY STANDARDS

of the
Nuclear Safety Standards Commission (KTA)

Reactor Core Design of High-Temperature Gas-Cooled Reactors
Part 3: Loss of Pressure through Friction in Pebble Bed Cores

(Auslegung der Reaktorkerne von gasgekühlten Hochtemperaturreaktoren
Teil 3: Reibungsdruckverlust im Kugelhaufen)

KTA 3102.3 (March, 1981)

Editor:

KTA-Geschäftsstelle c/o
Gesellschaft für Reaktorsicherheit (GRS) mbH
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SAFETY STANDARDS

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Nuclear Safety Standards Commission (KTA)

Reactor Core Design of High-Temperature
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Part 4: Thermohydraulic Analytical Model
for Stationary and Quasi-Stationary
Conditions in Pebble Bed Cores

(Auslegung der Reaktorkerne von
gasgekühlten Hochtemperaturreaktoren
Teil 3: Thermohydraulisches
Berechnungsmodell für stationäre und
quasistationäre Zustände im Kugelhaufen)

KTA 3102.4 (November, 1984)

Editor:

KTA-Geschäftsstelle c/o
Gesellschaft für Reaktorsicherheit (GRS) mbH
Schwertnergasse 1 · D–5000 Köln 1 · F.R.G.
Phone (0221) 2068-0 · Teletex 2214123 grs d

KTA SAFETY STANDARD

Issue 6/86	Reactor Core Design for High-Temperature Gas-Cooled Reactors; Part 5: Systematic and Statistical Errors in the Thermodynamic Core Design of the Pebble-Bed Reactor	KTA 3102.5

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PLEASE NOTE: Only the original German version of this safety standard represents the joint resolution of the 50-member Nuclear Safety Standards Commission (Kerntechnischer Ausschuss, KTA). The German version was made public in Bundesanzeiger No. 162A on Sep. 3, 1986. Copies may be ordered through the Carl Heymanns Verlag KG, Luxemburger Strasse 449, D-5000 Koeln 41.

All questions regarding this English translation should please be directed to:

KTA-Geschaeftsstelle c/o GRS mbH, Postfach 101650, D-5000 Koeln 1.

Nuclear Safety Standards Commission (KTA)

Federal Republic of Germany

KTA-Dok.-Nr. 3221/93/1
(März 1993)

Einführung zum Regelvorhaben

KTA 3221 "Metallische HTR-Komponenten"

In der Technik und besonders in der Kerntechnik besteht die Notwendigkeit, die in Forschungs- und Entwicklungsprogrammen gewonnenen Ergebnisse und Erfahrungen zu bewerten und in Verfahrensvorschriften, Berechnungsmethoden und Regeln umzusetzen. Die Bearbeitung der damit verbundenen Aufgaben erfordert eine mühevolle Abstimmungs- und Formulierungsarbeit. Dennoch wurden die Ergebnisse, die in HTR-Forschungsprogrammen und im Zusammenhang mit dem Bau des THTR 300 erzielt wurden, zusammengefaßt, um die erbrachten wissenschaftlichen und ingenieurmäßigen Hochleistungen einer breiteren Anwendung nutzbar zu machen. Darüber hinaus haben die Ergebnisse für Anlagen mit hochtemperaturbeaufschlagten Komponenten richtungsweisende Bedeutung.

In dem Arbeitsgremium KTA 3221 wurden die vorliegenden Fakten und Resultate in Vorschriften umgesetzt, die es dem Ingenieur ermöglichen sollen, hochtemperaturbeaufschlagte Komponenten zu entwerfen, zu bauen und zu betreiben.

Zwei Aspekte waren bei der Erstellung der Regeltexte zu KTA 3221 "Metallische HTR-Komponenten" von besonderer Bedeutung:

- Feststellen des gesicherten Standes der Wissenschaft und der technischen Erfahrungen;

- Formulierung technisch akzeptierter Empfehlungen und Vorschriften in einer solchen Form, daß die Abwicklung bei der Errichtung von Anlagen mit hohen Arbeitstemperaturen für alle Disziplinen transparent und überschaubar ist.

Gerade der letzte Aspekt ist für die Reaktortechnik von zentraler Bedeutung (wie aus den Verzögerungen beim Bau und des Genehmigungsverfahrens des THTR 300 deutlich erkannt werden mußte), während der erste Aspekt seine Bedeutung weit über die Reaktortechnik hinausgehend hat.

Für die vielfältigen HTR-typischen Komponenten mit Arbeitstemperaturen von Raumtemperatur bis 1000°C konnte zur Formulierung des KTA-Regeltextes auf die Ergebnisse umfangreicher Programme und Vorhaben zurückgegriffen werden, wie

- Werkstoffprogramme der Projekte PNP und HHT/1/
- Baubegleitende Werkstoff- und komponentenbezogene Programme für den THTR/2/
- Sonderforschungsprogramm des BMI/3/
- Verbundforschungsprogramm des BMFT/4/
- Aktivitäten in USA und Japan.

Die Erarbeitung der Störfalltopologie des HTR und die Risikostudie /5/ diente den deutlich herauszustellenden HTR-spezifischen sicherheitstechnischen Merkmalen.

So wurden vor Erstellung der KTA-Regel die sicherheitstechnischen Vorgaben formuliert und in ein HTR-Integritätskonzept überführt /6/, das sich an den besonderen Eigenschaften des HTR orientiert. Dabei kommt der Rolle der bis zu Temperaturen bis oberhalb 1400°C stabilen kugelförmigen Brennelemente eine besondere Bedeutung zu. Das Konzept gewährleistet den sicheren Aktivitätseinschluß für die Gesamtanlage. Die jeweiligen Komponenten wurden hierbei gemäß ihrer Funktion bewertet und nach folgendem Schema eingeteilt:

1. HTR-Komponenten mit Barriere- und Rückhaltefunktion
   - Brennelemente
   - Reaktordruckbehälter
   - Reaktorschutzgebäude
2. HTR-Komponenten, bei deren Versagen die Barrierefunktionen beeinträchtigt werden
   - Reaktoreinbauten
   - Abschaltsysteme
   - Kühlsysteme für Nachwärmeabfuhr
3. HTR-Komponenten, bei deren Versagen es zum Eindringen von Fremdmedien in den Primärkreislauf kommt
   - wärmetauschende Komponenten.

Ihre sicherheitstechnische Bedeutung wurde an die radiologische Auswirkung gekoppelt, die bei einem Versagen einer Komponente für die Umgebung und das Betriebspersonal in der Anlage entsteht. Zur Ermittlung der Auswirkungen sind Störfallanalysen heranzuziehen. Die sicherheitstechnische Einordnung der Komponenten ist dann gemäß **Tabelle 1** vorzunehmen.

Sischerheits-klassen SK	Bedeutung des Versagens einer Komponente aufgrund der Sicherheitsanalyse für	
	Umgebungsschutz	Arbeitsschutz
SK 1	Dosen überschreiten Grenzwerte nach § 28.3 StrlSchV	Dosen überschreiten Grenzwerte nach § 50.2 StrlSchV
SK 2a	Dosen erreichen Grenzwerte nach § 28.3 StrlSchV	Dosen erreichen Grenzwerte nach § 50.2 StrlSchV
SK 2b	Dosen überschreiten Grenzwerte nach § 45 StrlSchV	Dosen überschreiten Grenzwerte nach § 49 StrlSchV
SK 3	Dosen erreichen Grenzwerte nach § 45 StrlSchV	Dosen erreichen Genzwerte nach § 49 StrlSchV
SK 4	keine	keine

Tabelle 1: Sicherheitsklassen für Komponenten

Die Erörterung erfolgte für typische Komponenten an damaligen Projekten, die in der Anlage zu dieser Einführung beschrieben werden. Dabei erfolgte stets ein Abwägen der Aussagen von KTA 3201 und KTA 3211 für LWR und der Vorgehensweise gemäß TRD-Vorschriften und AD-Merkblättern. Die Bearbeitung der hier besprochenen Regel KTA 3221 "Metallische HTR-Komponenten" in Teilen entspricht der Vorgehensweise wie bei KTA 3201 und KTA 3211:

KTA 3221.1 Metallische HTR-Komponenten;
Teil 1: Herstellung von Werkstoffen und Erzeugnisformen

KTA 3221.2 -; Teil 2: Auslegung, Konstruktion und Berechnung

KTA 3221.3 -; Teil 3: Herstellung von Komponenten

KTA 3221.4 -; Teil 4: Betriebliche Überwachung und wiederkehrende Prüfung

KTA 3221.5 -; Teil 5: Zusätzliche Anforderungen für Reaktordruckbehälter aus Stahl von HTR, die nach KTA 3201 ausgelegt und gefertigt werden.

Für die Regeln KTA 3221.1 und KTA 3221.2 liegen nun Regeltextentwürfe vor, die im Arbeitsgremium ausführlich erörtert wurden. KTA 3221.3 hat den Status eines noch nicht im Detail abgestimmten Vorentwurfs. Nach Abbruch der HTR-Entwicklung in der Bundesrepublik Deutschland reichen die Erfahrungen mit der Inbetriebnahme vom THTR und der Betrieb des Versuchsreaktors AVR nicht aus, realitätsbezogene Vorschriften für die Regel KTA 3221.4 zu erarbeiten. Für KTA 3221.5 fehlt nach Einstellung der Forschungsarbeiten geeignetes Grundlagenmaterial zur Bewertung HTR-spezifischer Einwirkungen auf den Behälterwerkstoff. Es handelt sich dabei um

HTR-typische Neutronenbestrahlung sowie um Störfalltemperaturen, die über den Temperaturbereich von KTA 3201 hinausgehen.

Die besondere Bedeutung von KTA 3221.1 für andere Hochtemperaturanlagen liegt in den ausführlichen Werkstoffdatenblättern, deren Informationsgehalt weit vertiefender ist als übliche VDTÜV-Blätter oder DIN-Normen.

Die wesentliche Bedeutung der Vorschläge für die Dimensionierung, die Auslegung und die rechnerischen Nachweise liegt in der grundsätzlichen differenzierten Bewertung von lastkontrollierten Spannungen (primäre Spannungen) und von dehnungskontrollierten Spannungen (sekundäre Spannungen, i. a. thermisch bedingt). Diese Regel macht einen deutlichen Schritt von "Design by rules" zum "Design by analysis" hin. Eine Bewertung inelastischer Effekte anhand von werkstoffspezifischen Temperatur-Zeit-Grenzkurven vermittelt wertvolle Hinweise für Komponenten in Energieanlagen. Gleiches gilt für die Bewertung des Einflusses der Überlagerung von Ermüdungs- und Kriechbeanspruchung für die zulässige Betriebsdauer von Komponenten ohne und mit unterstellten Fehlern. Die Beurtei lungen von Kriechbeulen und Kriechratcheting hat im konventionellen Regelwerk keine Parallele und kann diesbezüglich als richtungsweisend gelten.

Die rechnerische Beurteilung des mechanischen Verhaltens konnte nicht, wie bei KTA 3201.2, komponentweise geführt werden. Hier müßte eine mehr grundsätzliche Verfahrensweise eingeschlagen werden.

Auch wenn in der Bundesrepublik Deutschland die Entwicklungsarbeiten für HTR-Anlagen eingestellt wurden und die Kerntechnik im allgemeinen sehr stark hinterfragt wird, war es sinnvoll, die werkstoff- und komponentenbezogenen Regeln zu formulieren. So bietet sich anderen Arbeitskreisen, die sich mit technischen Regeln für Hochtemperarturkomponenten beschäftigen die Chance, ohne eigene, ausführliche Auswertung auf die Ergebnisse der mehr als 10-jährigen Forschungs- und Entwicklungsarbeiten für Metallische HTR-Komponenten zurückzugreifen.

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Anlage

zur Einführung zum Regelvorhaben
KTA 3221 "Metallische HTR-Komponenten"

1 Darstellung der Thematik (Problemstellung)

Es ist festzustellen, daß in den vorhandenen Regelwerken bisher keine geschlossene Darstellung von Regelaussagen für die Auslegung und Herstellung von metallischen Komponenten für Hochtemperaturreaktoren besteht.

Die für den THTR-300 im Hamm-Uentrop erstellten Spezifikationen bauen im wesentlichen auf den Anforderungen der TRD, der AD-Merkblätter und des ASME-Codes Sect. III und dem Code N 47 auf. Die bei Errichtung und Betrieb des gasgekühlten Hochtemperaturreaktors THTR-300 gemachten Erfahrungen wurden bei der Erarbeitung des Vorberichts sowie bei den Regelentwurfsvorschlägen berücksichtigt. Außerdem liegen bereits Ergebnisse aus den Forschungsvorhaben

SR 191: "Erarbeitung von Grundlagen zu einem Regelwerk über die Auslegung von HTR-Komponenten für Anwendungstemperaturen oberhalb 800°C" (gefördert durch BMI) und

"Auslegungskriterien für hochtemperaturbelastete metallische und keramische Komponenten sowie des Spannbeton-Reaktordruckbehälters zukünftiger HTR-Anlagen" (gefördert durch BMFT) sowie

SR 418: "Beitrag zur Absicherung gegen unzulässige Verformung warmgehender Komponenten unter überlagerten Sekundär- und Primärspannungen" (gefördert durch BMU)

vor, die in vollem Umfang für den Vorbericht und die Bearbeitung der Regelentwurfsvorschläge herangezogen worden.

2 Typische Komponenten

Zur Darstellung der Thematik der zu erarbeitenden Regel werden nachfolgend typische Komponenten von den damals geplanten Projekten beschrieben, die in den Anwendungsbereich der Regel fallen.

2.1 Kurzbeschreibung des HTR 500 (Primärkreislauf)

Der Reaktordruckbehälter des HTR 500 ist, wie beim THTR-300, als Spannbetonbehälter in Großkavernenbauweise ausgeführt. In ihm ist der gesamte Primärkreislauf integriert. Die Hauptwärmesenke bilden 6 Gegenstrom-Dampferzeuger in Helix-Bauweise. Jedem Dampferzeuger ist eine Gebläse zugeordnet.

Das Kühlmittel Helium durchströmt den aus den Brennelementkugeln gebildeten Reaktorkern von oben nach unten, steigt in den Dampferzeugern zu den Gebläsen hoch und wird von dort in den Kaltgassammelraum zurückgeleitet.

Die Nachwärmeabfuhr erfolgt über die Dampferzeuger oder über zusätzliche Nachwärmeabfuhrsysteme mit wassergekühlten Wärmetauschern und elektrisch betriebenen Gebläsen, die ebenfalls im Reaktordruckbehälter integriert sind.

Als Auslegungsdaten für Wärmetauscher sind vorgesehen:

a) Dampferzeuger

He-Eintrittstemperatur	700 °C
He-Austrittstemperatur	260 °C
Primärsystemdruck	55 bar
Speisewassertemperatur	190 °C
Frischdampftemperatur	530 °C
Frischdampfdruck	190 bar

b) Hilfswärmetauscher

He-Eintrittstemperatur	700 °C
He-Austrittstemperatur	260 °C
Primärsystemdruck	55 bar
Wassereintrittstemperatur	60 °C
Wasseraustrittstemperatur	180 °C
Sekundärsystem	35 bar

2.2 Kurzbeschreibung des HTR-Moduls (Primärkreislauf)

Eine weitere Variante ist das Konzept des HTR-Moduls, bei dem der Druckbehälter aus Stahl in einem Reaktorgebäude untergebracht ist.

Das HTR-Modulkonzept ist dadurch gekennzeichnet, daß standardisierte nukleare Wärmeerzeugungseinheiten von 200 MJ/s thermischer Leistung zu Kraftwerken zusammengeschaltet werden können, die den Bereich von 200 bis 1600 MJ/s thermischer Leistung überspannen.

Die Elemente zur Erzeugung von Hochtemperatur-Frischdampf bzw. Prozeßdampf sind:

- der Reaktor in einem Stahldruckbehälter mit einer thermischen Leistung von 200 MJ/s, Kühlung des Cores durch Helium von oben nach unten strömend, sphärischen Brennelementen, Abschalt- und Regeleinrichtungen im Reflektor zur Heißabschaltung und Regelung und Kleinkugel-Absorbersystemen zur Langzeitabschaltung,
- der Dampferzeuger mit Aufwärtsverdampfung in einem Stahldruckbehälter,
- das Gebläse zur Umwälzung des Heliums, am Dampferzeugerbehälter angeflanscht,
- Verbindungsbehälter zwischen Kern und Dampferzeuger mit koaxialer Heißgas-/Kaltgasführung.

Als Auslegungsdaten für die Gesamtanlage für die Stromerzeugung sind vorgesehen:

He-Eintrittstemperatur	700 °C
He-Austrittstemperatur	250 °C
Primärkreisdruck	60 bar

Frischdampfdruck am DE-Austritt max.	190 bar
Frischdampftemperatur an DE-Austritt max.	530 °C
Speisewassereintrittstemperatur max.	200 °C

2.3 Anwendung eines HTR für die Bereitstellung von Prozeßwärme

Mit den beiden beschriebenen HTR-Konzepten können auch die Ziele des PNP-Projektes zur Erzeugung nuklearer Prozeßwärme verfolgt werden. Wesentliche Komponenten sind hierbei der Röhrenspaltofen und der Helium/Helium-Wärmetauscher.

2.3.1 Helium/Helium-Wärmetauscher (He/He-WT)

Aufgabe des He/He-WT ist die Auskopplung der Hochtemperaturwärme aus dem Primärkreis, um über einen Zwischenkreislauf (Sekundärkreislauf) die Vergaser der Wasserdampf-Kohlevergasungs-Anlage (WKV) bzw. je nach Projekt, andere chemische Prozesse mit der erforderlichen Prozeßwärme zu versorgen.

Für die Auslegung des He/He-WT, der je nach Projekt z. B. als Helex-Wärmetauscher (Einfach- oder Tandemausführung) oder als U-Rohr-Wärmetauscher ausgeführt werden kann, sind folgende Richtwerte anzusetzen:

He-Eeintrittstemperatur He/He-WT (Primärkreis)	950 °C
He-Austrittstemperatur He/He-WT (Sekundärkreis)	300 °C
He-Eintrittstemperatur (Zwischenkreislauf)	220 - 260 °C
He-Austrittstemperatur (Zwischenkreislauf)	900 °C
Primärseitiger Druck	39 - 40 bar
Sekundärseitiger Druck	41 - 43 bar

2.3.2 Röhrenspaltofen

Aufgabe des Röhrenspaltofens (RSO) ist unter Ausnutzung der Hochtemperaturwärme, die direkt aus dem Primärkreis ausgekoppelt oder über Zwischenkreisläufe zugeführt wird, Methan aus der hydrierenden Kohlevergasung (HKV) oder Erdgas zu Synthesegas zu spalten. Dieses so gewonnene Spaltgas kann zum Beispiel zur Herstellung von Methanol oder Wasserstoff eingesetzt, zu Hydriergas für die HKV aufbereitet oder als Energietransportmedium in Fernenergiesystemen (NFE) verwendet werden.

In Abhängigkeit von den HTR-Konzepten sind für die Auslegung des Röhrenspaltofens folgende Richtwerte anzusetzen:

He-Eintrittstemperatur RSO	900 - 950 °C
He-Austrittstemperatur RSO	640 - 700 °C
Methan + H_2O-Gemisch (Eintrittstemperatur)	330 °C
Reformiertes Gas (H_2 + CO) (Austrittstemperatur)	420 - 460 °C
Primärseitiger Druck	39 - 40 bar
Sekundärseitiger Druck	41 - 43 bar

Sicherheitstechnische Regel des KTA

KTA-Dok.-Nr. 3221.1/92/3
Regelentwurfsvorschlag
(Dez. 1992)*

KTA 3221.1

Metallische HTR-Komponenten

Teil 1:Herstellung von Werkstoffen und Erzeugnisformen

V O R B E M E R K U N G

Der Kerntechnische Ausschuß (KTA) beabsichtigt, eine kerntechnische Regel des oben angegebenen Themas aufzustellen. Der Entwurf dieser Regel wird hiermit der Öffentlichkeit zur Prüfung und Stellungnahme vorgelegt, damit er erforderlichenfalls verbessert werden kann. Es wird darauf hingewiesen, daß die endgültige Fassung der Regel von dem vorliegenden Entwurf abweichen kann.

Änderungsvorschläge sind innerhalb einer Frist von drei Monaten, beginnend

am ,

bei der Geschäftsstelle des Kerntechnischen Ausschusses im Bundesamt für Strahlenschutz (BfS), Postfach 100149, 3320 Salzgitter, einzureichen.

E N T W U R F

Inhalt **Seite**

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* Diese Fassung wurde in der Dokumentationsunterlage um die Beschlüsse der 47. KTA-Sitzung am 15.06.1993 ergänzt.

Sicherheitstechnische
Regel des KTA

KTA-Dok.-Nr. 3221.2/92/6
Regelentwurfsvorschlag
(Dez. 1992) *

KTA 3221.2
"Metallische HTR-Komponenten"

Teil 2: Auslegung, Konstruktion und Berechnung

V O R B E M E R K U N G

Der Kerntechnische Ausschuß (KTA) beabsichtigt, eine kerntechnische Regel des oben angegebenen Themas aufzustellen. Der Entwurf dieser Regel wird hiermit der Öffentlichkeit zur Prüfung und Stellungnahme vorgelegt, damit er erforderlichenfalls verbessert werden kann. Es wird darauf hingewiesen, daß die endgültige Fassung der Regel von dem vorliegenden Entwurf abweichen kann.

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Sicherheitstechnische
Regel des KTA

KTA-Dok.-Nr. 3221.3/92/3
Vorentwurf zum
Regelentwurfsvorschlag
(Dez. 1992)

KTA 3221.3
"Metallische HTR-Komponenten"

Teil 3: Herstellung von Komponenten

VORENTWURF

V O R B E M E R K U N G

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Änderungsvorschläge sind innerhalb einer Frist von drei Monaten, beginnend
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bei der Geschäftsstelle des Kerntechnischen Ausschusses im Bundesamt für Strahlenschutz (BfS), Postfach 10 01 49, 3320 Salzgitter 1, einzureichen.

Entwurf

Inhalt

Beratungsunterlage zu TOP 4.1
43. KTA-Sitzung am 27.06.1989

KTA-Dok.-Nr. 3231/89/4
Stand: April 1989

Vorbericht

zum Thema KTA 3231

"Sicherheitstechnische Anforderungen an die

Auslegung von Spannbeton-Reaktordruckbehältern

für Hochtemperaturreaktoren"

Sicherheitstechnische Regel des KTA

KTA-Dok.-Nr. 3232/92/3
Regelentwurfsvorlage
(Dez. 1992)*

KTA 3232

Keramische Einbauten in HTR-Reaktordruckbehältern

Vorbemerkung

Der Kerntechnische Ausschuß (KTA) beabsichtigt, eine kerntechnische Regel des oben angegebenen Themas aufzustellen. Der Entwurf dieser Regel wird hiermit der Öffentlichkeit zur Prüfung und Stellungnahme vorgelegt, damit er erforderlichenfalls verbessert werden kann. Es wird darauf hingewiesen, daß die endgültige Fassung der Regel von dem vorliegenden Entwurf abweichen kann.

Änderungsvorschläge sind innerhalb einer Frist von drei Monaten, beginnend

am ,

bei der Geschäftsstelle des Kerntechnischen Ausschusses im Bundesamt für Strahlenschutz(BfS),Postfach 100149, 3320 Salzgitter, einzureichen.

E N T W U R F

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XI. International HTGR Conference,
Dimitrovgrad, USSR, June 1989

Lecture given in Session 4

THTR Commissioning and Operating Experience

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I. Kalinowski

Hochtemperatur-Kernkraftwerk GmbH, Hamm

1. Introduction

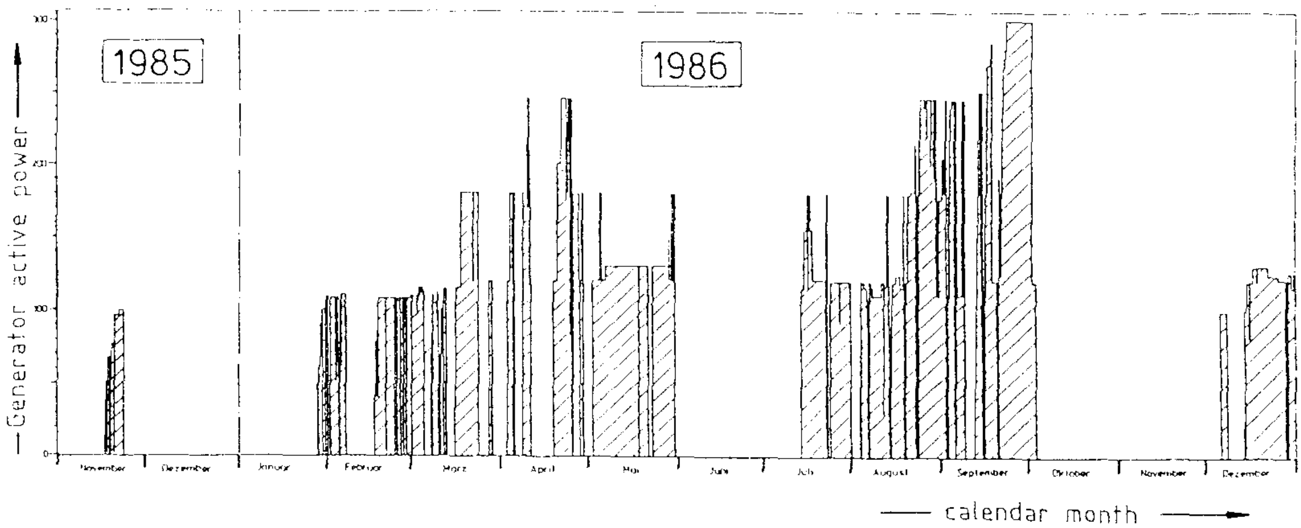
The Thorium-High-Temperature Reactor THTR 300 is the prototype power plant for a medium-sized pebble bed reactor. The commissioning period up to handover of the plant to the user was marked by the following milestones which characterize the extensive and time-consuming commissioning program:

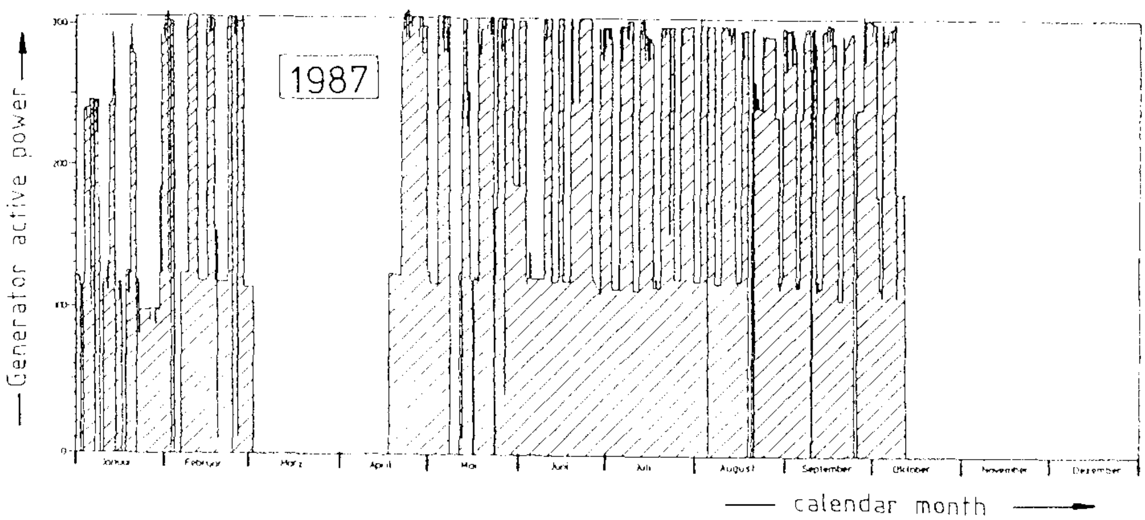
Sept 13, 1983	first criticality
Nov 16, 1985	first synchronization to power grid
Sept 23, 1986	first 100 % power operation
Juni 1, 1987	completion of nuclear trial operation and handover of the plant to the user company HKG

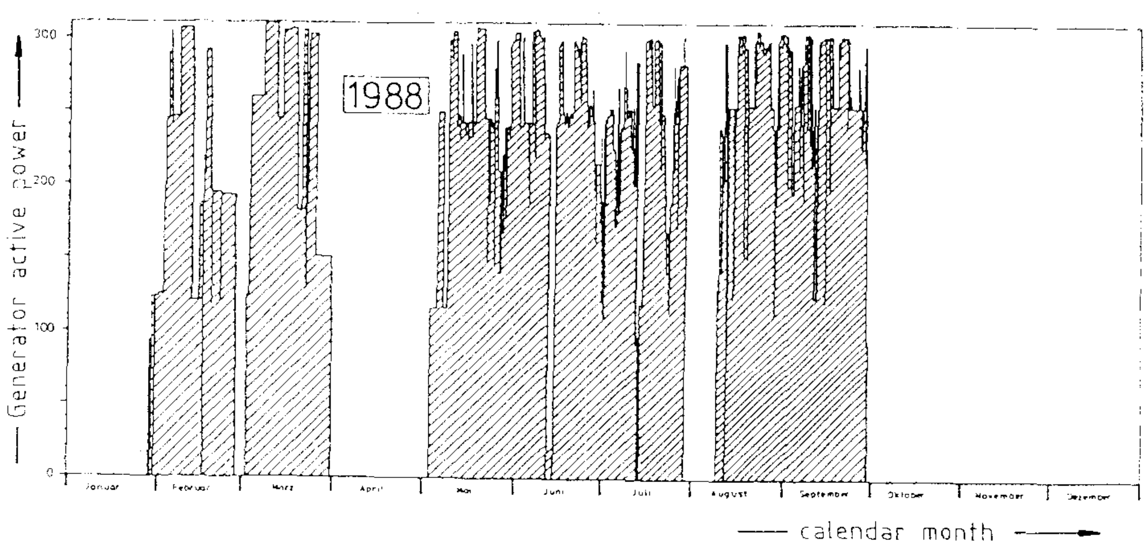
Until today the plant was in operation 16 410 h and has generated 2 891 068 MWh. The time availibilty has been 61 % in 1987 and 52 % in 1988.

The diagram of the previous operating history is a spike curve which is characterized by frequent power changes and several prolonged plant downtimes.

HKG THTR 300	Electric power output diagram during operating phase between Nov. 16, 1985 and Dec. 31, 1988	





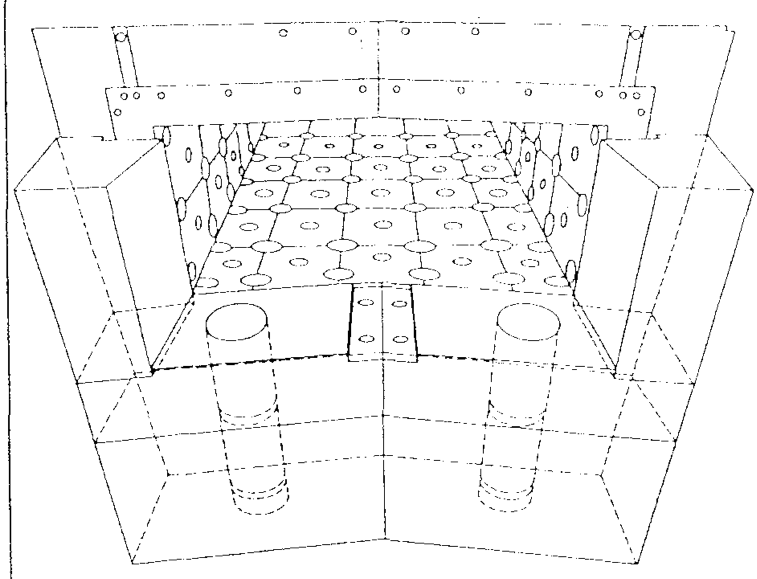


The power changes were initially caused by difficulties arising in the withdrawal of spherical elements from the reactor. In the beginning of the plant operation spheres could be withdrawn only at reduced plant power, since only with a reduced helium mass flow which is partly passed in countercurrent to the fuel element flow direction for cooling the fuel element discharge pipe, withdrawal of the spherical elements was possible. This defect was eliminated during the 1987/88 plant inspection. Further downtimes resulted from jamming of spheres in the singulizer disk of a helical damaged-spheres separator in the refuelling system and from the necessity to exchange the casks which collect the damaged spherical elements. Finally power reduction was repeatedly required in summer 1988 to keep the exhaust air temperature in those parts of the reactor hall within the permissible limits, which accommodate the components of the steam/feedwater circuit, e.g. the steam generator ring rooms. On September 29, 1988 the power plant was shut down for the scheduled 1988 inspection.

On the occasion of a routine inspection, we inspected - as a precautionary measure - a hot gas duct, the duct through which the hot helium passes from the reactor core to the steam generator. The figure shows an internal view of a hot gas duct with its rectangular passage through the graphite side reflector. The lower graphite blocks of the hot gas duct are each fixed to the respective carbon block by a graphite dowel. In the outer wall of the side reflector these dowels are positioned in bore holes penetrating the blocks. The figure shows the front part of the metallic section of the hot gas duct showing the inner insulation which consists of metal foil blankets, covered by 30 cm x 30 cm cover plates which are each held down and fixed by 4 corner bolts and 1 central bolt. After the inspection of the first duct had revealed damage on some attachment fixtures (central bolts), we decided to inspect all the 6 ducts, and it was detected that out of the approx. 2600 bolts 35 bolts heads had come off. In addition it was detected that several graphite dowels installed for holding in position the lower outer blocks of the hot gas duct had been displaced.

The damage has been thorougly analysed and the following causes have been determined: The bolt heads failed due to stresses which had concentrated in the range of the bolt head as a result of differential thermal expansion of the materials of the metal foil insulation consisting of 18 layers and the structure of the attachment fixture bolts. In addition a reduction in the material ductility as a result of thermal neutron irradiation in the temperature range above 500 °C was observed.

HKG THTR 300	Front part of the hot gas duct	



After thorough analyses we and the plant supplier have jointly come to the result that further operation of the THTR 300 is justified in spite of the existing damage.

Since the damage is essentially concentrated on the central bolts, the thermal insulation in the metal part of the hot gas duct is held down by the corner bolts as before. Thus the functional capability of the thermal insulation is safely ensured also in the present situation. In case that parts of the insulation were detached afterall, this would be detected by the operational monitoring of the process parameters mass flow and pressure loss. We have, however, the intention to observe the situation in future by inspecting the hot gas ducts in shorter intervals.

During the overall operation until shutdown of the power plant on September 29, 1988 for the 1988 inspection the plant has generated 2 891 068 MWh. For generating this electrical gross output the plant had to be operated for 423 full power days including the commissioning period.

In the following the main results of the plant operation are presented.

HKG THTR 300

THTR - Operating experience

Safety-relevant conclusions

Operation

Normal operation	Shutdowns Plant outages	Inspections
Design	Shutdowns / Decay heat removal	Radiological protection data
Core dynamics	Shutdown rods	Graphite dust
Temperature distribution	Penetration isolation valves	Activity
Refueling/ spheres damage	Emergency power supply	Inspection manual
Coolant gas activity		
Non-active impurities in the coolant gas		
Thermodynamics		
Measuring methods		

The evaluation of the operating data can be subdivided into three sections:

power operation,
plant downtimes including shutdown procedures, and
inspections.

From all three sections important information has been obtained which will be discussed in the paragraphs below.

2. Evaluation of Operating Experience

2.1 Design Data and Power Operation

HKG THTR 300	Comparison of measured and calculated operating data at 100 % power output on February 9, 1988		
	Unit	Measured valve	Calculated valve
Thermal power of core and internals	MW	756	755
Circulator speed	min $^{-1}$	5407	5380
Helium flow rate	kg/s	48,26	49,12
Feedwater flow rate	t/h	151,6	151,7
Mass flow through reheater	t/h	144,7	144,1
Hot gas temperature at SG inlet	°C	750,3	750
Cold gas temperature at SG outlet	°C	246,8	246,3
Main steam temperature	°C	534,2	535
Main steam pressure	bar (abs)	186,9	186,7
Reheat steam temperature	°C	530,6	527
Reheat steam pressure	bar (abs)	48,5	48,4
Generator active power	MW	304,3	304,3
Cooling water temperature	°C	26,7	26,5

The design data which had been specified for the THTR power operation have been confirmed by measurements during operation. This fact is not evident for a prototype plant. It shows that the theoretical bases for the design of hightemperature reactors are available. From the point of view of safety engineering the following aspects are interesting in this context:

2.1.1 Core Dynamics, Control Behaviour, Power Distribution

The core power output can be controlled at all power levels and under all core conditions without any problems. Power changes are possible in the range between 40 % and 100 % power output in any steps desired. Power changes are performed by ramps of 8 % per minute within the main operation range.

The previous operation has, on the one hand, confirmed the design values for the core and the operational and safety procedures and, on the other hand, it has verified the functional capability of the control equipment and the components of the primary and secondary system.

During power changes the electrical unit output, the main steam pressure, the main steam temperature and the cold gas temperature are controlled. The control variables for this purpose are the helium mass flow, the position of the reflector rods and the feed water quantity.

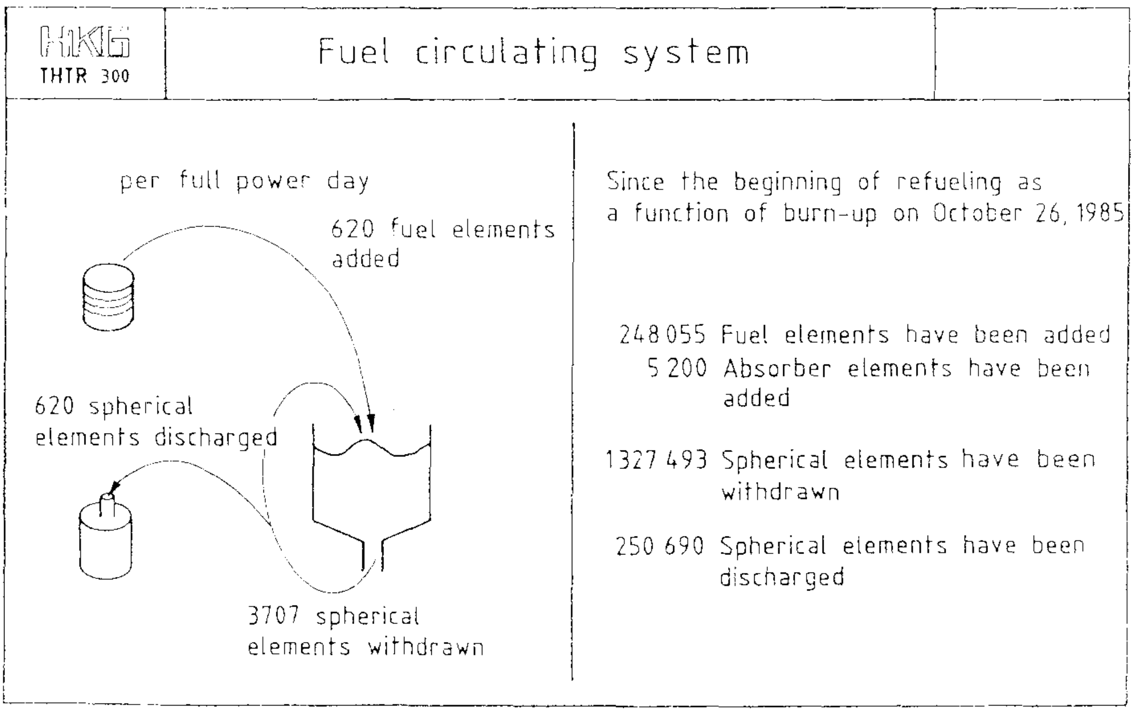
The control concept especially controls also upset operating conditions, such as the automatic power reduction to about 70 % in the event of failure of one circulator turboset, load rejection to plant auxiliary power, or turbine scram. Instabilities of the core behaviour never occur during such control porcedures, nor fluctuations of the power distributions (e.g. xenon fluctuations). The temperature coefficient of the THTR is negative in all power ranges. It is between Γ = - 12 mN/K and - 4 mN/K. For demonstrating the negative feed-back, the power and temperature curves were recorded at a thermal power of several per cent in the course of a controlled intentional "return to criticality" of the reactor. The curves showed the expected slow changes of power and temperature thus confirming the design calculations. The inherent safety of the THTR and its "good-natured" control behaviour has thus been verified experimentally.

2.1.2 Temperature Distribution in the Core

The requirements for the temperature distribution in the core result from the maximum permissible temperature of the fuel elements as well as from the maximum permissible insertion depth of the incore rods, which, in turn, results from the rod temperature which must not exceed the specified design values.

The permissible fuel element temperatures can be observed without any difficulties by manoevering the incore rods and the reflector rods so as to prevent power concentration in the lower core region. Another possibility of indirect control of the permissible temperatures is obtained by monitoring the hot gas temperature in the bottom reflector. Observance of the maximum incore rod tip temperatures is more difficult. For this purpose it is necessary to perform design calculations on the temperature and power distribution in the core in parallel with the operation. During the running-in phase these distributions continously change. Due to potential uncertainties in the calculated maximum rod tip temperatures in practice conservative safety margins for the permissible insertion depth are required. This sets a limit to the possibility of using the incore rods. Due to high excess reactivities, which occur for example after prolonged plant downtimes, relatively deep insertion of the incore rods is required also during power operation. This may result in power restrictions for a limited period (approx. 2 weeks) to ensure that the maximum incore rod tip temperatures are not exceeded.

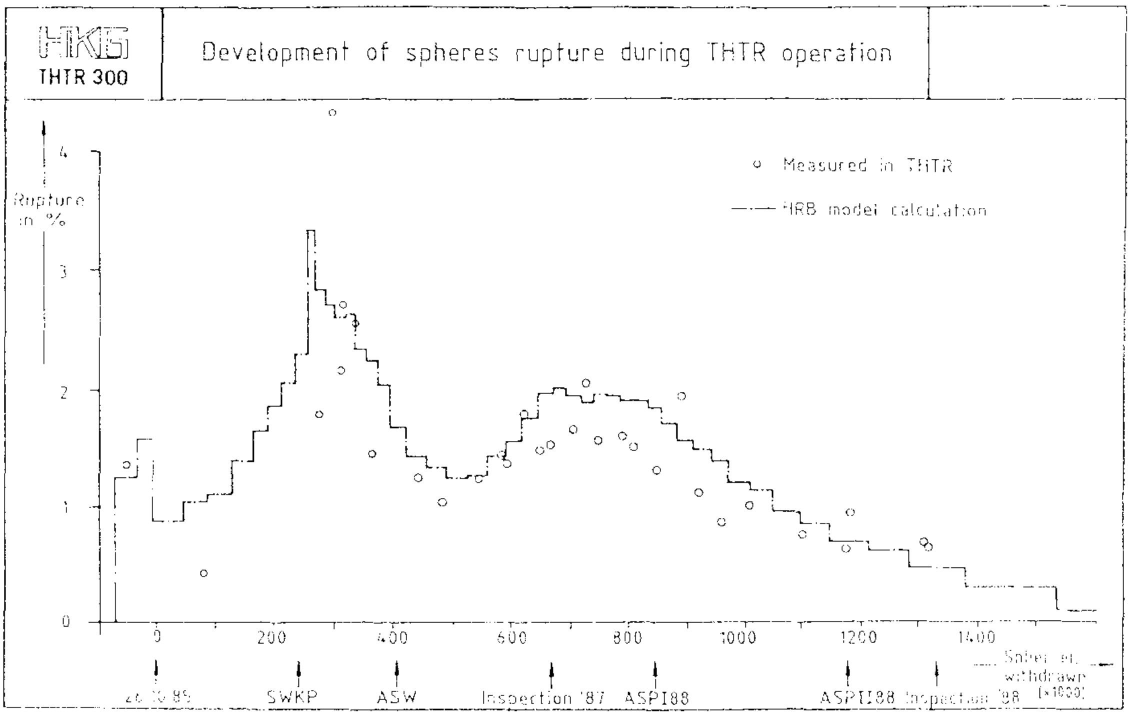
2.1.3 Refueling and Damage of Spherical Elements



A special characteristic of the THTR is continuous refueling. 3707 spherical elements are withdrawn from the reactor core per full power day. 620 spherical elements are discharged from the circuit, the rest is returned into the reactor core. The 620 spherical elements withdrawn are replaced by 620 fresh fuel elements.

Up to 29.09.1988 a total of 1,3 million spherical elements from the core have been drawn off, from this figure 235 000 spherical elements taken away and replaced by a correspondant number of fresh spherical elements. Essential for the safety of reactor operation is the correct, i.e. refueling of the reactor core according to design. The spherical elements are added to the core according to a refueling strategy calculated in advance. This procedure has proven to be successful in previous refueling practice. The subsequent calculations will, however, require new reference data for calculations to actual measured values. In this aspect the calculation model can certainly be further improved, e.g. by using measured values on the flow behaviour of the spherical elements and the measured burn-up spectrum of the fuel elements discharged. The observance of the safety-relevant design data such as excess reactivity, power distribution and temperature distribution and, thus, the guaranty of the rod worths does not pose any problems. These data are continously verified experimentally and are thus ensured at any time independent of the calculations.

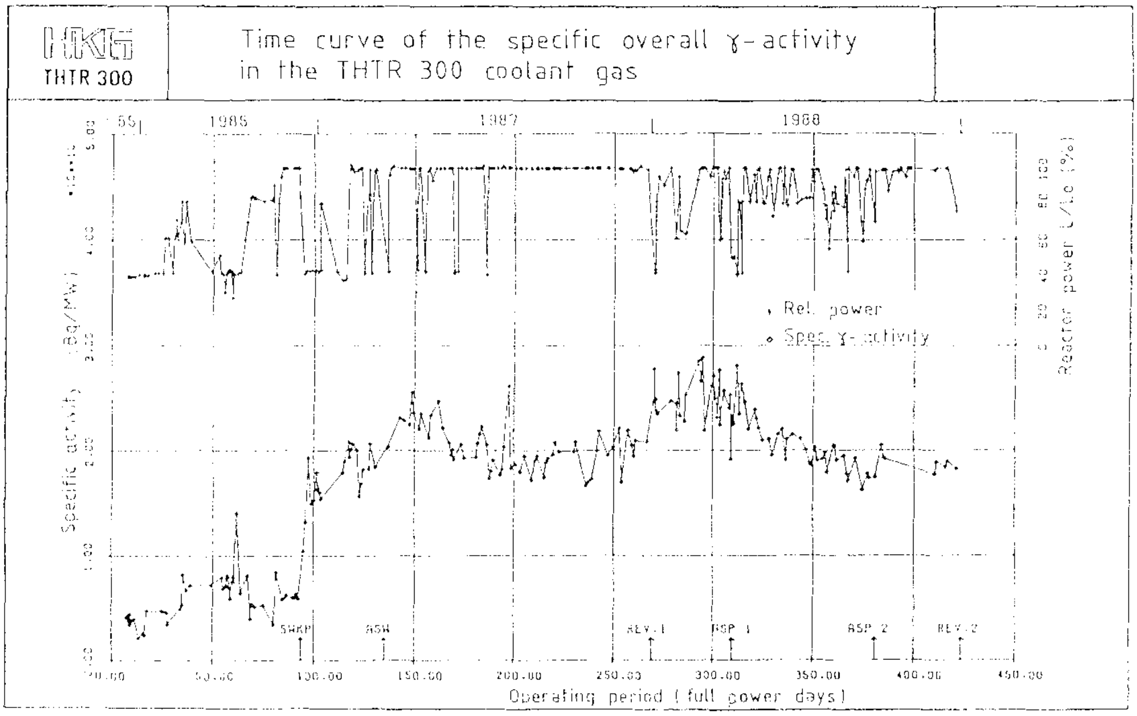
The practical performance of the refueling procedure met with some difficulties. They had no safety relevance and were eliminated as was described earlier. This applies as well to the unexpected high number of damaged sherical elements, which were sorted out by the helical scrap separator during withdrawal of the spherical elements from the reactor core. Up to the present time 10 casks have been filled with approx. 17.000 damaged spherical elements. The share of damaged spherical elements in the total amout of spherical elements withdrawn was about 1.5 % in the beginning of the refueling operation and is continously drecreasing. Recently the rate reached 0.6 %.



A model calculation was performed based on the assumption that the damage was mainly caused by frequent and deep insertion of the incore rods during the THTR commissioning phase. This assumption has been confirmed by the agreement with the experimental data. Since the damage in most cases only concerns the graphite shell in which the fuel is embedded, i.e. the coated fuel particles in the damaged fuel elements are intact in their greatest part, retention of the fission products is ensured as before. The flow behaviour of the spherical elements in the reactor core and the insertion of the incore rods is not impaired by the damaged spherical elements. Therefore the damage of spheres has no safety relevance.

Elimination of the disturbances of the process described above requires, however, a great effort, e.g. the exchange of casks for damaged spherical elements requires complete depressurization of the prestressed concrete reactor vessel. Therefore it is intended - in particular also for economic reasons - to change the mode of manoevering the incore rods so that damage of further spherical elements is reduced to a minimum, R + E work is carried out for an evaluation of the mode of spheres rupture and the mechanical behaviour of the pebble bed in order to obtain an exact analysis of all the effects occurring.

2.1.4 Coolant Gas Activity in the Primary Circuit



The coolant gas activity of the THTR does not exceed the expected values. The overall development of the coolant gas activity is shown in the figure. As had been expected, the coolant gas activity increased during the commissioning phase with increasing reactor power reaching almost constant values at continuous full power operation. It remains clearly and constantly below the design values. As for the AVR, the fission product retention capability of the fuel elements has thus been confirmed also for the THTR in power operation.

2.1.5 Non-Radioactive Impurities in the Coolant Gas

The impurities contained in the coolant gas, H_2O, CO_2, H_2 and in some rare cases also traces of O_2, which have an oxidizing effect on graphite, have removed 65 kg of carbon from the spherical elements and the graphite internals up to the present time.

This carbon quantity has to be considered in relation to the overall carbon inventory of the core which is 728 tons. The helium purification system of the THTR has been able to cope with all concentrations of impurities without any problems. The primary circuit with its auxiliary systems does not pose any problems with regard to chemical and radiochemical parameters.

HKG THTR 300

Impurities in the THTR 300 Coolant Gas

		undisturbed	after injection of ammonia
H_2O	µbar/vpm	≤ 0,5/< 0,01	< 2/< 0,05
H_2	µbar/vpm	30 / 0,8	up to 4000/ up to 100
CH_4	µbar/vpm	4 / 0,1	up to 200/ up to 5
CO_2	µbar/vpm	8 / 0,2	
CO	µbar/vpm	16 / 0,4	
N_2	µbar/vpm	< 4 /< 0,1	up to 2000/ up to 50
O_2	µbar/vpm	n.n.	
Ar	µbar/vpm	n.n.	
Kr, Xe	Bq/m^3 i.N.	3×10^8	
^3H	Bq/m^3 i.N.	$5 - 20 \times 10^5$	
^{14}C	Bq/m^3 i.N.	100 - 300	
Aerosole	Bq/m^3 i.N.	< 100	
NH_3	µbar/vpm	n.n.	up to 3800/ up to 100

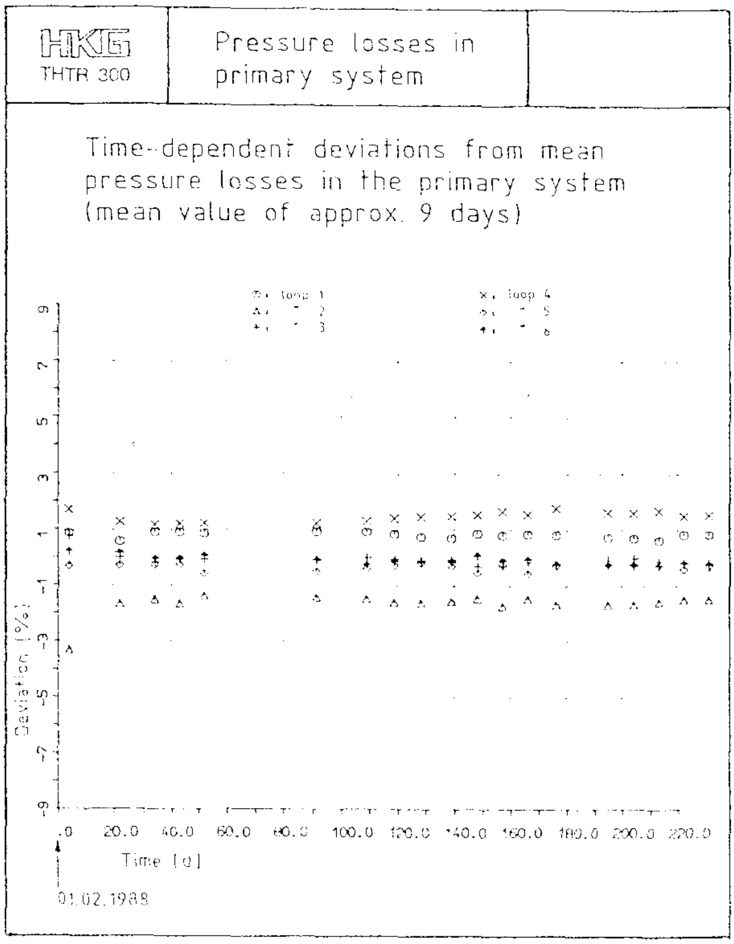
In general the impurities in the coolant gas (H_2O, H_2, CH_4, CO_2, Co und N_2) are low. During steady-state operation they sum up to a maximum of 80 µbar (= 2vpm). Only during start-up the contents of hydrogen and nitrogen may rise by NH_3 decay up to the level of a few mbar. During shutdown of the reactor ammonia is fed into the core to reduce the friction factor of the incore rods in the pebble bed core.

2.1.6 Thermodynamic Parameters of the Primary System

In addition to the operational data quoted at the beginning of this chapter, which are directly included into the power calculation, a number of additional data are measured to describe the primary system. This has shown that the bypass of the helium mass flow is higher than expected. It is defined to be being 18 % instead of 7 %, which had been expected. The core outlet temperature which has therefore to be higher by about ten per cent is below the design values for fuel elements and graphite internals even at full power operation, thus it does not pose any problems. In connection with the damage of the attachment fixtures of the hot gas duct insulation reference should briefly be made to another group of thermodynamic data of the primary system. Apart from the temperatures, these are the helium mass flows and the pressure losses of the 6 steam generator/circulator units. These data are continuously recorded and evaluated in the THTR. In addition, derived values such as e.g. the pressure loss coefficients are continuously determined.

These values are observed, on the one hand as mean values of all the six steam generator/circulator units for detecting uniform changes in all the 6 hot gas ducts and, on the other hand, they are evaluated as relative deviations from the mean value for determining irregular changes in individual hot gas ducts.

Evaluations performed during the latest year of operation have shown that changes of the above-mentioned data in the primary system are detectable with an accuracy of 1 %.



These statements show that in addition to the measured values for the reactor core itself also the thermodynamic data of the primary system are stable and reproduceable. Therefore safety-relevant changes which may occur can be detected safely and early enough. Thus it is demonstrated that the design has been confirmed and that the components such as e.g. the helium circulator and the steam generator have proven their functional capability.

2.1.7 Measuring Methods

Another condition for safe plant operation is the correct acquisition and reliable processing of all the measured values required for plant safety and plant operation. The instrumentation concept of the THTR - including the elimination of incore instrumentation - and the practical application of the measuring facilities have proved to be efficient. This applies also to the special measuring facilities necessary for a prototype plant, such as neutron flux instrumentation, temperature measurements of the metal and ceramic internals, instrumentation for measurements in the helium circulators and steam generators, spheres counting equipment and burn-up measurement facility. The information on the plant required for safety reasons has been available at any time.

2.2 Conclusions from Shutdown Procedures and Plant Downtimes

2.2.1 Shutdown Procedures, Decay Heat Removal Systems

As shown earlier in the operational diagram, the THTR has been shut down relatively frequently during the commissioning phase and the power operation. Part of the shutdown procedures were scheduled and maintenance and repair measures, especially in-service inspections. In addition, especially during the trial operation, the excitation of the two automatic shutdown procedures was repeatedly triggered by the Plant Protection System: reactor scram (11 x, 4 of them as tests during the commissioning phase) or Decay Heat Removal 45 procedure (20 x). The causes were a too narrow adjustment of the limiting values, (this was eliminated during the comissioning phase), defective instruments, errors in detail planning of release logics and operator errors. The greatest part of the releases were not required for safety reasons. In all the shutdown procedures heat removal from the core and from the internals was effected according to the design principles. Minor irregularities in the procedures were never of safety relevance and were eliminated in the course of the commissioning phase. Experience has shown up to now that the decay heat removal systems which are partly identical with operational systems have a sufficient availability, an appropriate process design, and have proven their fuctional capability in practice. In the course of the overall operating period including the shutdown procedures several hundred measuring data are being recorded and evaluated in sections by the continuously operating long-term recording program of the process computer system. The "service life consumption" of the steam generators and the associated piping amounts only a few percent. Only some solid parts which could be exchanged, have reached a life time consumption of about 10 % up to now. Assuming a "normal" further power operation, there are no restrictions or safety-relevant problems to be expected from today's point of view for a further long-term operation.

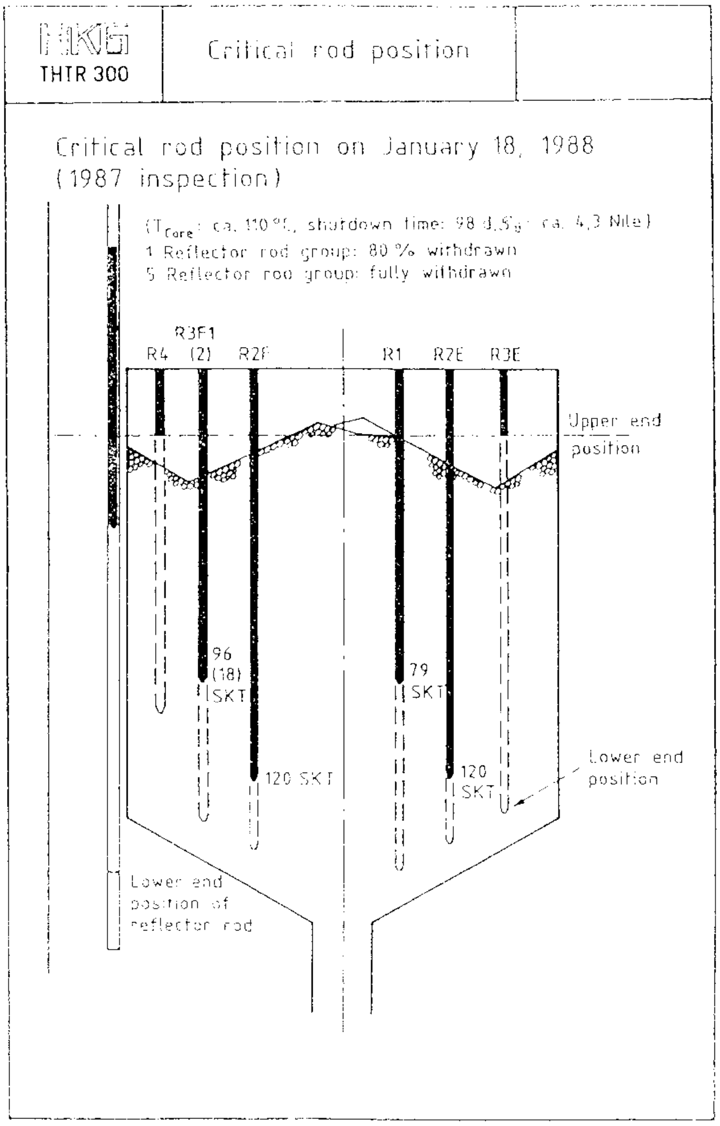
The cooldown procedure "Heat Removal 5" designed to come into action in the event of major disturbances, or the measures for resumption of heat removal after a prolonged interruption of decay heat removal (LUNWA) have not come into action up to now. Therefore it can be stated that the previous operating experience does not give rise to any new safety requirements with regard to detection of disturbances and release and sequence of cooldown procedures. It is currently being investigated, whether there is a possibility of simplifying the excitation logics of the Plant Protection System and improving the sequence of the cooldown procedures. The use of the absorber rods could be reduced, as will be demonstrated in the section below.

2.2.2 Shutdown Systems

The THTR ist equipped with two independent shutdown systems, the reflector rods (6 groups of 6 rods each) and the incore rods (7 groups of 6 rods each). Four reflector rod groups represent the shutdown system, the incore rods are inserted for long-term shutdown. In order to ensure sufficient subcriticality, it was claimed that during the running-in phase in the event of reactor scram in addition to the reflector rods a group of incore rods (group R3E) should be automatically inserted by the long-stroke pistion drive.

This claim has proven to be unnecessary at an early date, since it has been demonstrated during the commissioning phase on the occasion of scram tests from power operation that the reactor is still subcritical after 30 minutes by insertion of the reflector rod shutdown groups alone without additional insertion of the incore rod group and that the reactor remaine subcritical over the period of xenon build-up. This situation is maintained even under the most adverse conditions by definition (start-up after prolonged standstill, no xenon, low helium temperature). The claim for automatic insertion of an incore rod group in the event of reactor scram can therefore be eliminated.

For automatic long-term shutdown it was envisaged to insert all the 42 incore rods to their lower end position. Also for these conditions it has been repeatedly demonstrated that the measures for long-term shutdown of the reactor need not be applied to the extent originally envisaged. Even with the boundary conditions of maximum excess reactivity, low helium temperature, long-term subcriticality after prolonged operation, i.e. with full protactinium conversion, it is sufficient to insert 4 incore rod groups to a depth about 1 m above the lower end position. The figure below shows as an example the critical rod position after the 1987 Inspection.



The long-term shutdown system was designed too conservatively so that it is overdimensioned. For this reason the incore rods are only inserted in that number and depth which is required for safety reasons to ensure sufficient subcriticality during prolonged plant downtimes. Since the incore rods have to be considered to be the cause of the increased rate of damaged spherical elements, it is expected that this measure will result in a marked reduction of spheres damage.

From a process design aspect the incore rods and the reflector rods have proven to be efficient safety systems. The dropping times of the reflector rods corresponded to the design values, the insertion times and insertion depths of the incore rods when automatically inserted by the long-stroke pistons have ensured subcriticality of the reactor at any time.

2.2.3 Penetration Isolation System

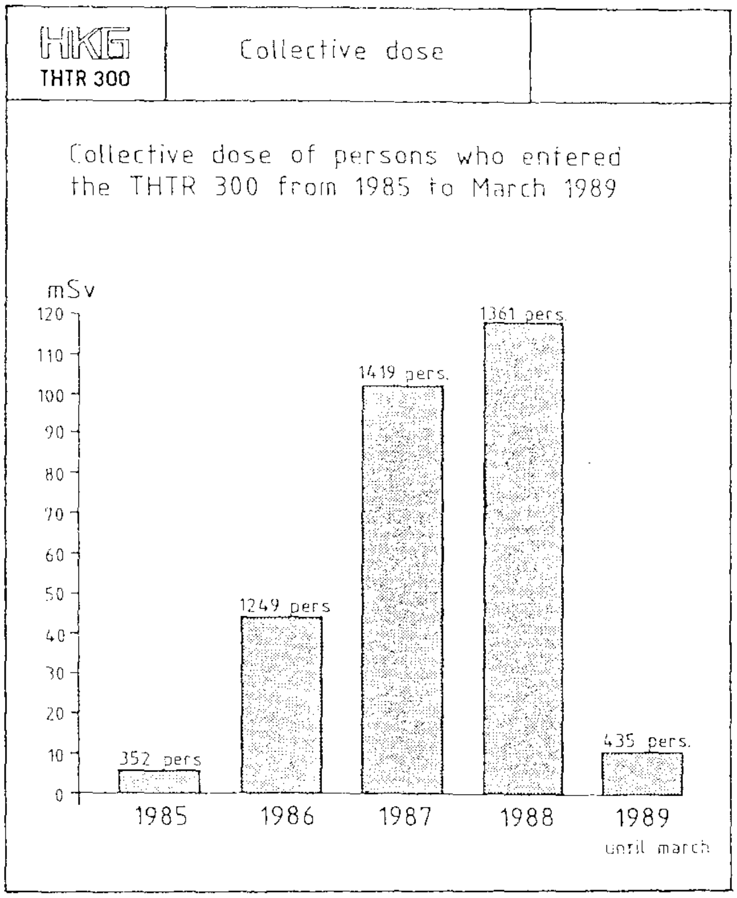
The penetration isolation system consists of shut-off valves equipped with diverse drive systems. Each pipe penetrating the PCRV and carrying primary gas is shut off by these valves to ensure activity confinement. Each line is equipped with two valves which close in case of demand upon excitation by the Plant Protection System. In the course of the THTR operation no disturbances have occurred up to now which would have required an activation of the penetration isolation system. Modifications or backfitting of these active engineered safety systems has not become necessary as a result of the previous operation.

2.2.4 Emergency Power Supply

The only case of emergency power supply occurred in the beginning of the commissioning phase. It was initiated by the attempt to switch over the electrical feed water pump form a supply line to a redundant line within approximately 1 second. This resulted in shutdown of the supplying transformer. This distrubance gave rise to several modifications of details, optimizations and definitions of process design. In principle, however, the concept for detection and activation of the emergency power supply system has been confirmed.

2.3 Inspections

2.3.1 Radiological Protection Data Referring to Plant Personnel



Radiation exposure of the THTR plan personnel is very low. The radiation exposure values for the previous operating period are indicated in the figure. The data demonstrate that the plant concept with a prestressen concrete reactor vessel has proven to be successful.

This fact applies also to the conditions prevailing in the event of maintenance or repair work on components of the primary. By using special disassembly facilities and tools and observance of the sequences of work planned in detail, these activities can be carried out with a low collective as well as single dose. When in April 1988 repair work on one of the helical damaged-spheres separators of the fuel circulating system had to be performed, the overall collective dose was 2.71 mSv and the maximum single doses were less than 0.2 mSv. We assume that such favorable values can be maintained also in future.

2.3.2 Graphite Dust

It has been detected on piping carrying primary gas and on components disassembled from the primary system that surfaces of components which are part of the helium circuits and the fuel circulation system are contaminated by radioactive graphite dust (mass deposition about 1 mg/cm^2). The specific activity of the dust was determined to be 2×10^8 Bq/g at a maximum. It is mainly caused by the radionuclides Co-60, Nb/Zr-95, Hf-181 und Pa 233. The overall quantitiy of graphite dust detected corresponds to the expected weigth loss of the spherical elements during circulation by abrasion. Under the aspect of radiological protection it does not pose any problems for disassembly work. The only effects of the graphite dust on the operation of systems were noticed in the beginning of the commissioning phase, when individual moisture sensors in the moisture monitoring system of the steam generators failed. This source of failure was eliminated by installing simple dust filters upstream the sensors. It can thus be stated that the graphite dust does not pose any problems, neither with regard to operation nor to safety.

Measurements on piping carrying primary gas have shown that also in the event of a depressurization accident the graphite dust does not cause an increased release of activity.

2.3.3 Activity Release with Vent Air

HKG THTR 300	Activity release	

Activity release with exhaust air
1988

	Release in Bq	Licensed annual limit value in Bq	Release in % of annual limit value
Inert gases	2,504E11	6,66E14	0,037 %
Aerosols	8,968E07	3,7 E08	24,2 %
Jodine	1,086 E07	3,7 E08	2,9 %
H3-Control area	3,471E12	8,14E12	42,8 %
C 14	2,682 E10	7,4 E12	0,36 %

The activity release with vent air measured in 1988 is presented in the figure. It was no problem during power operation to remain below the low limiting values specified in the THTR license, because at that time only minor repair work was performed on components of the helium circuit.

To reduce the release of radioactive aerosols to the environment, i.e. the release of activity carried by graphite dust, it has proved necessary in the course of the commissioning phase to provide all exhaust paths with filters. This has been done and has proved to be a successful solution.

Contrary to normal operating conditions, during inspections the PCRV is often depressurized and open to perform some work on integrated components. To maintain a specified flow direction, the PCRV is kept under a slightly negative pressure during the performance of the above-mentioned repair work. For this purpose a small partial quantity of the helium inventory is withdrawn from the PCRV and released to the atmosphere with the vent air. Since the graphite internals still contain tritium after depressurization, which in case of moisture enters the gas phase via exchange reactions, the gas mixture withdrawn form the PCRV has to be passed through catalysers and a molecular sieve before it is released to the atmosphere. By this measure it is ensured that even in the event of complete ventilation of the PCRV no safety problems will arise.

3. Experience Expected form Further Operation of the THTR-300

A further operation of the THTR-300 is expected to furnish essential know-how in addition to the present operating experience and would thus allow to come to a valuable completion of the research contract. It is especially expected by us that it will be possible to extend and confirm by experiments the know-how on core design, spheres damage rate, and the activity release from the spheres.

Another objective is the verification of the long-term performance of the prototype components. The hot gas duct is an example which is significant at the present moment, but also the long-term behaviour of other prototype components such as shutdown rods, PCRV and graphite internals is of great interest.

A further task which could be persued during a further operation of the THTR is the development of disassembly and repair equipment for the components installed within the PCRV. The previous operation has demonstrated that the problem of accessiblity is of utmost importance to the operating company and that the development of disassembly equipment is urgently required. In our opinion it is another task of a prototype to verify the easy repairability of High-Temperature Reactors.

For initiating these tasks it is, however, necessary to obtain a new definition of the financial basis for the THTR-300 project.

4. The Risk Participation Contract and Covering of Financial Risks

As early as in 1971 the partners cooperating in the THTR-300 project had realized that because of the prototype character of the THTR-300 and the research objectives pursued with this reactor it would not be possible to achieve a commercial operation of the plant from the very beginning. For this reason a risk participation contract was negotiated and concluded already at the beginnig of the project earmarking a liability sum of DM 450 million to cover the economic risks of the plant operation and the decommissioning risks of the plant operation and the decommissioning costs. Two thirds of this sum was furnished by the Federal Government and one third by the Federal State Government of North Rhine Westphalia. DM 270 million are reserved for compensating losses from plant operation, and DM 180 million are presently envisaged for decommissioning of the plant.

It is further stipulated in the contract that during the first 3 years 10 % of the operating deficit is covered by the HKG partners and 90 % is furnished form the sum guarranteed in the risk participation contract.

After three years the share assumend by the HKG partners increases to 30 %. Since the latest up-dating of the risk participation contract in 1983 the costs of decommissioning (dismantlement) of the plan have increased compared to the costs earmarked in the risk participation contract. Based on an expert opinion the costs of dismantlement of the plant, quoted at DM 180 million in the existing risk participation contract, have now increased to about DM 450 million.

As a result of new risks affecting the THTR-300 project from external sources, which might result in plant outages, the HKG partners are of the opinion that the guaranteed sum of DM 450 million is not sufficient. All these new risks came up in concrete form late in 1988.

In the following they will be briefly characterized:

Risk of Standstill due to Fuel Element Supply Problems

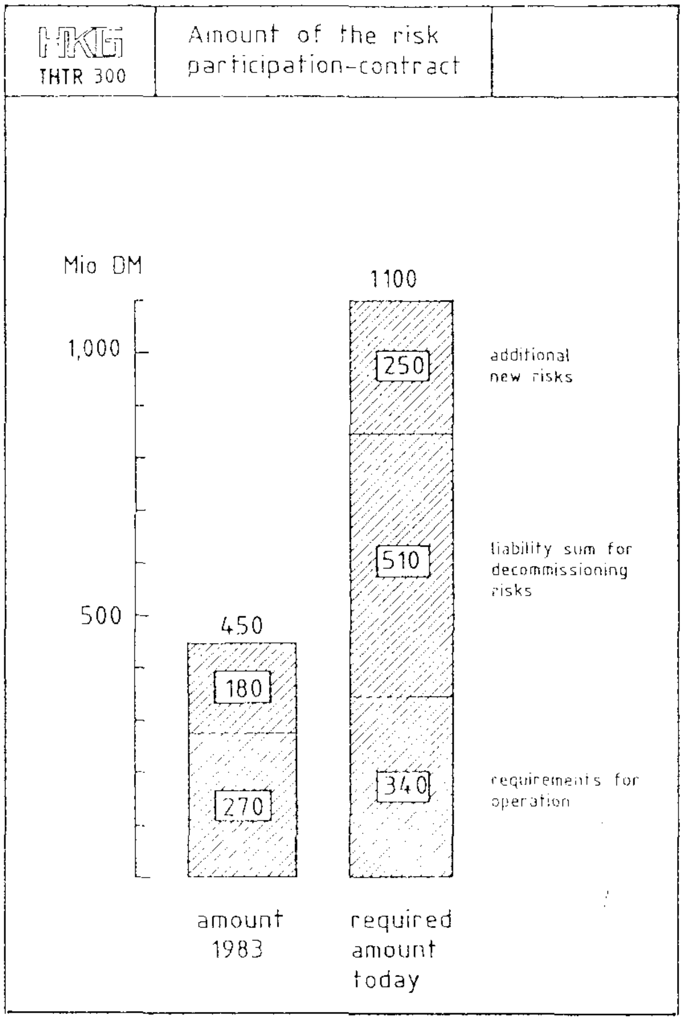
The fuel elements for the THTR-300 fabricated by NUKEM up to the present time are sufficient for an operating period until end 1991. On December 31, 1988 NUKEM terminated the fabrication of the spherical fuel elements. Continuation of the fuel element fabrication in due time is currently not ensured.

Risk of Standstill due to Fuel Element Disposal Problems

It is claimed in the operating license for the THTR-300 that it has to be given evidence at the end of 600 full power days, this would be some time early in 1990, that external intermediate storage facilities for the spent fuel elements are available and that the license has been obtained for the transport preparation hall for storing low-activity waste on the THTR plant site. Both conditions have not yet been met at the present time.

Risk of Standstill due to Problems Regarding the Permanent Operating License

The present operating license for the THTR-300 covers 1100 full power days, i.e. it will expire in mid 1992. The subsequent permanent operating license requires another licensing procedure. At the moment it cannot be predicted which will be the requirements and criteria of this licensing procedure. In any case there is a high probability that the competent nuclear licensing authority will perform a detailed safety investigation before granting a license for further plant operation. In view of this situation the HKG partners have asked the partners of the risk participation contract to increase the contractual amount guaranteed to DM 1.1 billion. In evaluating the increase of the sum guaranteed it has to be emphasized that it is intended to cover a financial risk which must not occur with certainty. If for example a further operation of the THTR-300 at an availability of 70.4 % was possible within a long-term program, the sum guaranteed would be claimed only to a maximum of DM 340 million.



The figure shows the individual items of the risk participation contract and the increase considered necessary by the HKG partners.

5. Summary

The evaluation of the operating experience gained from the THTR up to now comes to an absolutely positive result. The principal design data have been confirmed.

The THTR-300 represents the successful connection link between the 15 MW_{el} AVR experimental reactor and a future commercial plant. On the basis of the present know-how obtained from the THTR operation another optimized high-temperature reactor can be designed and constructed thus representing a further step towards commercialization of advanced reactors.

It is evident that the necessity to increase the risk participation contract does not arise from safety considerations but exclusively from economic factors affecting the THTR from outside.

TÜVIS	Sicherheitskriterien für Kernkraftwerke	Entwurf September 1980

Sicherheitskriterien für Anlagen zur Energieerzeugung mit gasgekühlten Hochtemperaturreaktoren

- Entwurf September 1980 -

erstellt
für den Bundesminister des Inneren

E S S E N
15. September 1980

siehe auch BMI - Schreiben vom 4.11.1980

Inhaltsverzeichnis

1. Vorbemerkung

Der Bundesminister des Innern beauftragte [1] die TÜV-Arbeitsgemeinschaft Kerntechnik West im Unterauftrag über die Gesellschaft für Reaktorsicherheit (GRS) mbH, einen ersten Entwurf zu Sicherheitskriterien für gasgekühlte Hochtemperaturreaktoren (HTR) unter Mitarbeit der GRS zu erstellen.

Entsprechend der Beauftragung wurden als Grundlage und Gliederungsvorschlag die Sicherheitskriterien für Kernkraftwerke vom 21.10.1977 herangezogen. Die Entwürfe zu den einzelnen Kriterien wurden im Sinne von Grundsätzen allgemein gehalten. Unverändert wurden die Kriterien 1.1, 2.3, 2.6 bis 2.10 und 9.1 übernommen. An anderen Stellen wurden Ergänzungen vorgenommen, zum Beispiel wurde ein Kriterium "Bauliche Schutzvorkehrungen zur Rückhaltung radioaktiver Stoffe" (Kriterium 8.5) formuliert. Übergeordnet wurde das Kapitel "Einzelfehlerkonzept" aufgenommen. Als Text wurde die Interpretation zu den Sicherheitskriterien für Kernkraftwerke, "Einzelfehlerkonzept - Grundsätze für die Anwendung des Einzelfehlerkriteriums - "[2] übernommen unter Streichung der Fußnote 4 und Änderung des Punktes (2).

1) Schreiben des BMI an die GRS vom 21. Oktober 1977, Gesch.-Z. RS I6-513 301/3;
Schreiben der GRS an die TÜV-ARGE Kerntechnik West vom 10. November 1977, Az.: gu/hos;
Schreiben der GRS an die TÜV-ARGE Kerntechnik West vom 16. Februar 1978, Az.: gu-br/81403

2) Bekanntmachung des BMI vom 26. Oktober 1978 GMBL 1978, 631

Für einen Erstentwurf, der im November 1978 dem Auftraggeber vorgelegt wurde, erfolgte die Bearbeitung der Kriterien 2.6 bis 2.10 und 10.3 durch Sachverständige der GRS, die der Kriterien 2.1 bis 2.5 und 9.1 durch Sachverständige des TÜV-Rheinland, die übrigen Kriterien wurden von Sachverständigen des RWTÜV bearbeitet. Kommentare und Begründungen zu einzelnen Kriterien, Hinweise zu zusätzlichen Gesichtspunkten und das Ergebnis einer Literaturrecherche nach in- und ausländischen Regeln und Richtlinien, die auf gasgekühlte Hochtemperaturreaktoren angewendet werden können, sind im Entwurf November 1979 [3] aufgeführt.

Der vorliegende Kriterienentwurf entstand aus 3) durch Einarbeiten von Stellungnahmen, die der Bundesminister des Innern von dem Länderausschuß für Atomkernenergie, dem Deutschen Gewerkschaftsbund, der Reaktorsicherheitskommission, dem Ingenieurbüro Zerna/Schnellenbach, der Vereinigung der Großkesselbetreiber, dem Zentralverband der elektrotechnischen Industrie sowie der TÜV-Leitstelle Kerntechnik eingeholt hat.

Der Kriterienentwurf enthält Grundsätze für die sicherheitstechnischen Anforderungen, die der Auslegung von Kernkraftwerken mit Hochtemperaturreaktoren zugrunde zu legen sind. Für andere Anlagen zur Energieerzeugung mit Hochtemperaturreaktoren (z. B. Anlagen zur Prozeßwärmeerzeugung) gelten die Kriterien in den nichtanlagenspezifischen Forderungen ohne Einschränkungen, in anlagenspezifischen Forderungen sinngemäß.

3) TÜV Arbeitsgemeinschaft Kerntechnik West
Sicherheitskriterien für gasgekühlte Hochtemperaturreaktoren
Entwurf November 1979
Essen, 2. November 1979

2. Kriterienentwurf

2.1 Inhalt

1) Diese Kriterien wurden aus den Sicherheitskriterien für Kernkraftwerke vom 21.10.1977 (verabschiedet im Länderausschuß für Atomkernenergie am 12. Oktober 1977) unverändert übernommen.

Abschnitt 9	
Kriterium 9.1	Lüftungstechnische Anlagen [1)]
Abschnitt 10	
Kriterium 10.1	Strahlenschutzüberwachung
Kriterium 10.2	Aktivitätsüberwachung in Fortluft und Abwasser
Kriterium 10.3	Umgebungsüberwachung
Abschnitt 11	
Kriterium 11.1	Handhabung und Lagerung von Kernbrennstoffen und sonstigen radioaktiver Stoffen

1) Dieses Kriterium wurde aus den Sicherheitskriterien für Kernkraftwerke vom 21.10.1977 (verabschiedet im Länderausschuß für Atomkernenergie am 12. Oktober 1977) unverändert übernommen.

2.2 Definitionen

1. Ableitung radioaktiver Stoffe

Die Abgabe flüssiger, aerosolförmiger oder gasförmiger radioaktiver Stoffe aus der Anlage auf hierfür vorgesehenen Wegen.

2. Bestimmungsgemäßer Betrieb

(1) Betriebsvorgänge, für die die Anlage bei funktionsfähigem Zustand der Systeme (ungestörter Zustand) bestimmt und geeignet ist (Normalbetrieb);

(2) Betriebsvorgänge, die bei Fehlfunktion von Anlagenteilen oder Systemen (gestörter Zustand) ablaufen, soweit hierbei einer Fortführung des Betriebes sicherheitstechnische Gründe nicht entgegenstehen (anomaler Betrieb);

(3) Instandhaltungsvorgänge (Inspektion, Wartung, Instandsetzung).

3. Einzelfehler [1)]

Ein Einzelfehler ist ein Fehler, der durch ein einzelnes Ereignis hervorgerufen wird, einschließlich der durch den Fehler entstehenden Folgefehler.

1) Zur Anwendung des Einzelfehlers innerhalb der Kriterien siehe Kapitel "Einzelfehlerkonzept".

4. Freisetzung radioaktiver Stoffe

Das Entweichen radioaktiver Stoffe aus den vorgesehenen Umschließungen in die Anlage oder in die Umgebung.

5. Grenzwert

Grenzwerte im Sinne dieser Kriterien sind diejenigen Werte der Zustandsgrößen von Anlagenteilen, Systemen oder darin enthaltenen Medien, bei deren Einhaltung ein Versagen sicherheitstechnisch wichtiger Einrichtungen mit angemessenem Sicherheitsabstand ausgeschlossen ist.

6. Inhärente sicherheitsgerichtete Eigenschaften

Inhärente sicherheitsgerichtete Eigenschaften einer Reaktoranlage sind solche, die aufgrund naturgesetzlicher Zusammenhänge auch ohne Eingreifen des Sicherheitssystems Störungen, Störfälle oder Unfälle entweder verhindern oder deren Auswirkungen verzögern, mildern oder begrenzen.[2)]

7. Prozeßveriable

Die Prozeßvariable ist eine unmittelbar im Prozeß meßbare physikalische Größe.

2) Beispiele für inhärente sicherheitsgerichtete Eigenschaften sind:
- Phasenstabilität des Kühlmittels Helium,
- negativer Temperaturkoeffizient der Reaktivität,
- Wärmeleistungs-, Wärmespeicherungseigenschaften und thermische Stabilität der Graphiteinbauten.

8. Reaktorschutzsystem

Das Reaktorschutzsystem ist ein System, das die Werte der für die Sicherheit der Reaktoranlage und Umgebung wesentlichen Prozeßvariablen zur Erfassung von Störfällen überwacht, verarbeitet und Schutzaktionen auslöst, um den Zustand der Reaktoranlage in sicheren Grenzen zu halten.

9. Redundanz

Vorhandensein von mehr funktionsbereiten technischen Mitteln, als zur Erfüllung der vorgesehenen Funktion notwendig ist.

10. Sicherheitssystem

Das Sicherheitssystem ist die Gesamtheit aller Einrichtungen einer Reaktoranlage, die die Aufgabe haben, die Anlage vor unzulässigen Beanspruchungen zu schützen und bei auftretenden Störfällen deren Auswirkungen auf das Betriebspersonal, die Anlage und die Umgebung in vorgegebenen Grenzen zu halten.

11. Störfall

Ereignisablauf, bei dessen Eintreten der Betrieb der Anlage oder die Tätigkeit aus sicherheitstechnischen Gründen nicht fortgeführt werden kann und für den die Anlage ausgelegt ist oder für den bei der Tätigkeit vorsorglich Schutzvorkehrungen vorgesehen sind.

12. Unfall [3]

Ereignisablauf, der für eine oder mehrere Personen eine die Grenzwerte übersteigende Strahlenexposition oder Inkorporation radioaktiver Stoffe zur Folge haben kann, soweit er nicht zu den Störfällen zählt.

[3] Übernommen aus der Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (Strahlenschutzverordnung - StrlSchV) vom 13. Oktober 1976 (BGBL I (1979) 2905), Anlage I.

2.3 Einzelfehlerkonzept

In den Kriterien

5.2: Nachwärmeabfuhr nach Störfällen,

6.1: Reaktorschutzsystem,

7.1: Elektrische Energieversorgung des Sicherheitssystems

wird die Annahme eines Einzelfehlers gefordert.

(1) Beim Einzelfehler handelt es sich um einen zufälligen, nicht als Folge des zu betrachtenden Anforderungsfalls im bestimmungsgemäßen Betrieb oder bei Störfällen auftretenden und vor Eintritt des Anforderungsfalls nicht bekannten zusätzlichen Fehler in den Sicherheitseinrichtungen. Ein Fehler liegt vor, wenn ein Systemteil [1)] bei Sicherheitseinrichtungen seine Funktion bei Anforderung nicht erfüllt. Eine betrieblich mögliche Fehlbedienung, die einen Fehler in den Sicherheitseinrichtungen zur Folge hat, ist einem Einzelfehler gleichzusetzen.

Gründe für den unterstellten Fehler müssen im allgemeinen nicht angegeben werden.

1) Der Begriff "Systemteil" umfaßt alle Teile der Funktionseinheit selbst und der zu ihrer sicherheitstechnisch richtigen Funktion notwendigen und - ggfs. auch redundanten - Versorgungs-, Steil- und Hilfseinrichtungen.

(2) Die Annahme des Einzelfehlers (Einzelfehlerkonzept) ist ein deterministisches Konzept für die Auslegung der Sicherheitseinrichtungen in Kernkraftwerken. Sie dient neben anderen Verfahren und Maßnahmen, wie z.B. die probabilistische Analyse (Zuverlässigkeitsanalyse) und die Qualitätssicherung zur Sicherheitsvorsorge.

Die sichere Erfüllung der im jeweiligen Kriterium geforderten Systemfunktion der genannten Sicherheitseinrichtungen muß ohne Inanspruchnahme der Redundanzen gewährleistet werden, die zur Abdeckung des gemäß Einzelfehlerkonzept im jeweiligen Anforderungsfall zu unterstellenden Einzelfehlers zusätzlich erforderlich sind.
Beim Nachweis der erforderlichen Verfügbarkeit der genannten Einrichtungen mit Hilfe probabilistischer Analysen werden die möglichen Fehler durch die Ausfallraten der Komponenten erfaßt.

(3) Einzelfehler werden grundsätzlich sowohl bei aktiven als auch bei passiven Systemteilen unterstellt.
Für passive Systemteile ist das Versagen im Rahmen des Einzelfehlerkonzeptes dann nicht zu unterstellen, wenn nachgewiesen wird, daß sie gegen die bei allen für sie zu unterstellenden Anforderungsfällen maximal zu erwartenden Beanspruchungen unter Berücksichtigung der im Betriebszeitraum vorhersehbaren Veränderungen der Werkstoffeigenschaften mit ausreichenden Sicherheitszuschlägen ausgelegt sind, aus einem für den Verwendungszweck geeigneten Werkstoff gefertigt werden und unter einer umfassenden Qualitätssicherung hergestellt, montiert, errichtet, geprüft und betrieben werden, so daß eine ausreichende Zuverlässigkeit gesichert ist.

Die hierbei anzuwendenden Maßnahmen und die Sicherheitszuschläge sind auch entsprechend der sicherheitstechnischen Bedeutung (orientiert am Schadensausmaß bei unterstelltem Versagen) der Sicherheitseinrichtungen festzulegen.

(4) Müssen zur Beherrschung eines zu unterstellenden Anforderungsfalls mehrere der in den eingangs genannten Kriterien geforderten Systeme gleichzeitig oder auch zeitlich nacheinander ihre Funktion erfüllen, so ist das Auftreten eines Einzelfehlers für die Summe der Systeme nach Maßgabe der Grundsätze des Einzelfehlerkonzeptes zu unterstellen, nicht aber für mehrere der benötigten Systeme gleichzeitig.

Davon abweichend ist in der Störfallanalyse bei Annahme des Ausfalls der ersten Anregung des Reaktorschutzsystems das gleichzeitige Auftreten eines Einzelfehlers an aktiven Systemteilen zu unterstellen, bei gleichzeitigem Instandsetzungsfall jedoch erst nach einem Zeitraum von 100 Stunden.

(5) Für jedes der in den Sicherheitskriterien 5.2, 6.1 und 7.1 geforderten Systeme ist das Auftreten eines Einzelfehlers auch während Instandhaltungsvorgängen zu unterstellen. Dies gilt nicht für Inspektionen, wenn die Funktionsbereitschaft des betroffenen Systemteils im Anforderungsfall rechtzeitig wiederhergestellt werden kann.

Mit der Instandsetzung ist unverzüglich nach der Schadenserkennung zu beginnen.
Instandhaltungsarbeiten an redundanten Sicherheitseinrichtungen, während derer das jeweilige System nicht funktionsbereit ist, sind ohne besondere, seine Funktion ersetzende oder seine Funktionsbereitschaft überflüssig machende Maßnahmen (z. B. Abschaltung, Leistungsminderung) nur zulässig, wenn für die Dauer der Instandhaltungsarbeiten das Einzelfehlerkonzept erfüllt ist und wenn außerdem zulässige Instandhaltungszeiten eingehalten werden. Die Inspektionsintervalle sowie die ohne besondere Maßnahmen zulässigen Wartungs- und Instandsetzungszeiten (Zeit ab Schadenserkennung bis Abschluß der Instandsetzung) sind unter Verwendung der für die genannten redundanten Systeme durchgeführten Zuverlässigkeitsanalysen so festzulegen, daß die Zuverlässigkeiten dieser Systeme durch die Instandhaltungsarbeiten nicht unter die zur Störfallbeherrschung erforderlichen Zuverlässigkeiten herabgesetzt werden.

Während kurzzeitiger Instandsetzungen braucht das zusätzliche Auftreten eines Einzelfehlers an Systemteilen nicht unterstellt zu werden, wenn wegen der Kürze der Instandsetzungsdauer die Zuverlässigkeit der betrachteten Sicherheitseinrichtung nicht wesentlich herabgesetzt wird.

(6) Für Gebäude und Teile von Gebäuden und für passive Teile des Sicherheitsbehälters ist ein Versagen im Rahmen des Einzelfehlerkonzepts nicht zu unterstellen, wenn die unter (3) geforderten Nachweise erbracht werden.

(7) Anlageninterne Störfälle und Einwirkungen von außen sind grundsätzlich gleichzusetzen. Dabei ist das Einzelfehlerkonzept wie beschrieben anzuwenden.

Bei äußeren Einwirkungen mit sehr geringer Eintrittswahrscheinlichkeit (wie z.B. Flugzeugabsturz und Explosionsdruckwelle) ist das gleichzeitige Auftreten eines Einzelfehlers nicht zu unterstellen; auch ein gleichzeitiger Instandsetzungsfall wird nicht postuliert.

Erfordert die Beherrschung einer derartigen äußeren Einwirkung die Funktion von Sicherheitseinrichtungen eher als nach einer Zeit von 30 Minuten, so ist ein Einzelfehler in den aktiven Systemteilen zu unterstellen. Bei der Betrachtung der Langzeit-Nachkühlphase ist nachzuweisen, daß erforderlichenfalls an den für die Langzeit-Nachkühlphase benötigten Sicherheitseinrichtungen rechtzeitg Instandsetzungsmaßnahmen durchgeführt werden können.

(8) Fehler infolge derselben Ursache an mehreren zueinander redundanten Systemteilen und Auslegungsfehler werden durch das Einzelfehlerkonzept nicht abgedeckt. Fehler dieser Art müssen durch geeignete Maßnahmen vermieden werden, wie z.B.

2.4 Abschnitte 1 bis 11

Abschnitt 1

Kriterium 1.1: Grundsätze der Sicherheitsvorsorge

Die Anlage muß so beschaffen sein und so betrieben werden, daß die Reaktoranlage jederzeit im bestimmungsgemäßen Betrieb und bei Störfällen sicher abgeschaltet und in abgeschaltetem Zustand gehalten, die Nachwärme abgeführt und die Strahlenexposition des Personals und der Umgebung unter Beachtung des Standes von Wissenschaft und Technik auch unterhalb derjenigen Dosisgrenzwerte so gering wie möglich gehalten werden kann, die durch die Vorschriften des Atomgesetzes und der aufgrund des Atomgesetzes erlassenen Rechtsverordnungen festgesetzt sind. Die hierzu nach dem Stand von Wissenschaft und Technik erforderliche Sicherheitsvorsorge ist nach folgenden Grundsätzen vorzunehmen:

1. Der erste und vorrangige Grundsatz wird gebildet durch hohe Anforderungen an die Auslegung und die Qualität der Anlage sowie an die Qualifikation (Fachkunde und Zuverlässigkeit) des Personals. Bereits dadurch muß auch ohne Inanspruchnahme der Sicherheitseinrichtungen ein möglichst störfallfreier und umweltverträglicher Betrieb der Anlage gewährleistet sein. Zu diesem Zweck sind sicherheitsfördernde Auslegungs-, Fertigungs- und Betriebsgrundsätze anzuwenden. Insbesondere sind zu verwirklichen:
   - Berücksichtigung ausreichender Sicherheitszuschläge bei der Auslegung der Systeme und Anlagenteile;

- Verwendung überprüfter Werkstoffe;
- Instandhaltungsfreundlichkeit von Systemen und Anlagenteilen unter besonderer Berücksichtigung der Strahlenexposition des Personals;
- ergonomische Maßnahmen an den Arbeitsplätzen;
- umfassende Qualitätssicherung bei Fertigung, Errichtung und Betrieb;
- Durchführung von wiederkehrenden Prüfungen in angemessenem Umfang;
- sichere Überwachung der Betriebszustände;
- Aufzeichnung, Auswertung und sicherheitsbezogene Verwertung von Betriebserfahrungen.

Nach allgemeiner technischer Erfahrung können während der Lebensdauer einer Anlage Fehlfunktionen von Anlagenteilen oder Systemen (anomale Betriebszustände) auftreten. Zur Beherrschung dieser anomalen Betriebszustände sind Systeme zur Betriebsführung und -überwachung vorzusehen. Diese Systeme sind so auszulegen, daß Störfälle als Folge von anomalen Betriebszuständen mit ausreichender Zuverlässigkeit [1] vermieden werden.

1) Anmerkung zur Methodik:
Zur Überprüfung der Ausgewogenheit des Sicherheitskonzeptes sind - in Ergänzung der Gesamtbeurteilung der Sicherheit der Anlage aufgrund deterministischer Methoden - die Zuverlässigkeiten sicherheitstechnisch wichtiger Systeme und Anlagenteile mit Hilfe probabilistischer Methoden zu bestimmen, soweit dieses nach dem Stand von Wissenschaft und Technik mit der erforderlichen Genauigkeit möglich ist.

2. Als zweiter Grundsatz sind darüber hinaus Maßnahmen zur Beherrschung von Störfällen zu treffen. Hierfür sind ausreichend zuverlässige [1] technische Sicherheitseinrichtungen vorzusehen. Diese Sicherheitseinrichtungen sind so auszulegen, daß sie das Personal und die Bevölkerung vor den Auswirkungen von Störfällen schützen. Dazu sind folgende Auslegungsgrundsätze anzuwenden:
   - Redundanz, Diversität, weitgehende Entmaschung von Teilsystemen, räumliche Trennung redundanter Teilsysteme;
   - sicherheitsgerichtetes Systemverhalten bei Fehlfunktion von Teilsystemen oder Anlagenteilen;
   - Bevorzugung passiver gegenüber aktiven Sicherheitsfunktionen.

   Darüber hinaus sind in angemessenem Umfang vorsorglich organisatorische und technische Maßnahmen innerhalb und außerhalb der Anlage zur Feststellung und Eindämmung von Unfallfolgen vorzusehen.

[1] Fußnote siehe oben.

Abschnitt 2

Kriterium 2.1: Qualitätssicherung

Planung, Erzeugung, Erhaltung und Nachweis der Qualität sind in Art und Umfang entsprechend der sicherheitstechnischen Bedeutung der Systeme und Anlagenteile durch technische und organisatorische Maßnahmen zu sichern. Diese Qualitätssicherung hat nach Grundsätzen und Verfahren zu erfolgen, die nach dem Stand von Wissenschaft und Technik den besonderen Erfordernissen der Sicherheit in der Kerntechnik angemessen sind.

Kriterium 2.2: Prüfbarkeit

Alle Anlagenteile müssen grundsätzlich so beschaffen und angeordnet sein, daß sie entsprechend ihrer sicherheitstechnischen Bedeutung vor ihrer Inbetriebnahme und danach in regelmäßigen Zeitabständen in hinreichendem Umfang geprüft und gewartet werden können. Wenn an Anlagenteilen regelmäßig wiederkehrende Prüfungen nach dem Stand der Technik nicht in dem für die Erkennung etwaiger Mängel erforderlichen Umfang durchgeführt werden können, so sind für die Erhaltung des einwandfreien Zustandes oder der einwandfreien Funktion des Anlagenteils besondere Maßnahmen zu treffen [1])

1) Zu diesen Maßnahmen können gehören:
- zusätzliche Sicherheitszuschläge bei der Auslegung,
- besondere Anforderungen an die Werkstoffe, wie Reinheitsgrad usw.,
- Fertigungsqualität,
- konstruktive Gestaltung, z.B. redundante Strukturen,
- Begrenzung und Kontrolle der Betriebsparameter,
- geplanter Austausch von Komponenten.

Die Folgen etwaiger noch zu unterstellender Mängel müssen so beschränkt werden, daß die Reaktoranlage auch bei den unter diesen Umständen in Betracht zu ziehenden Ereignissen sicher abgeschaltet und in abgeschaltetem Zustand gehalten werden kann, die Nachwärme abgeführt und die Ableitung oder eine etwaige Freisetzung radioaktiver Stoffe unter Beachtung der Regeln von Wissenschaft und Technik auch unterhalb der zugelassenen Werte so gering wie möglich gehalten wird.

Kriterium 2.3: Strahlenexposition in der Umgebung

Zum Schutz der Umgebung vor den Auswirkungen der Anlage muß gewährleistet sein, daß alle sicherheitstechnisch wichtigen Anlagenteile so ausgelegt sind und sich in einem solchen Zustand befinden und gehalten werden, daß die Strahlenexposition in der Umgebung durch Direktstrahlung aus der Anlage sowie Ableitung und etwaige Freisetzung radioaktiver Stoffe unter Beachtung des Standes von Wissenschaft und Technik auch unterhalb der zugelassenen Werte so gering wie möglich gehalten wird. Zu diesem Zweck müssen diese Anlagenteile so beschaffen und gegen Einwirkungen geschützt sein, daß sie im bestimmungsgemäßen Betrieb und bei Störfällen ihre sicherheitstechnischen Aufgaben erfüllen können.

Kriterium 2.4: Strahlenexposition in der Anlage

Alle Anlagenteile, die radioaktive Stoffe enthalten oder enthalten können, müssen so beschaffen, angeordnet und abgeschirmt sein, daß die Strahlenexposition von Personen bei allen im bestimmungsgemäßen Betrieb erforderlichen Tätigkeiten unter Beachtung der Regeln von Wissenschaft und Technik auch unterhalb der zugelassenen Werte so gering wie möglich ist. Zur Erfüllung dieses Grundsatzes müssen die Anlagenteile insbesondere auch instandhaltungsfreundlich beschaffen und angeordnet sein.

Alle Anlagenteile sind grundsätzlich so zu gestalten, daß Aus- und Einbauarbeiten bei Ersatz, Instandsetzungs- und Wartungsvorgängen und wiederkehrenden Prüfungen unter Einhaltung der oben genannten Forderungen zügig durchgeführt werden können. Hierfür erforderliche Maßnahmen sind vorzusehen [1].

Kriterium 2.5: Arbeitsgestaltung nach ergonomischen Erkenntnissen

Arbeitsaufgaben, Arbeitsplätze, Arbeitsumgebung, Arbeitsorganisation und Arbeitsmittel in der Anlage sind unter Berücksichtigung gesicherter ergonomischer Erkenntnisse so zu gestalten, daß sie die Voraussetzung für ein sicherheitstechnisch optimales Verhalten der Beschäftigten bieten.

1) Diese Maßnahmen können z.B. sein:
- Ausbaueinrichtungen,
- Transporteinrichtungen,
- Lagereinrichtungen,
- Abstelleinrichtungen,
- Überwachungseinrichtungen,
- Abschirmung und der notwendige Platz dazu.

Kriterium 2.6: Einwirkungen von außen

Alle Anlagenteile die erforderlich sind, den Kernreaktor sicher abzuschalten, ihn in abgeschaltetem Zustand zu halten, die Nachwärme abzuführen oder eine etwaige Freisetzung radioaktiver Stoffe zu verhindern, müssen so ausgelegt sein und sich in einem solchen Zustand befinden und gehalten werden, daß sie ihre sicherheitstechnischen Aufgaben auch bei naturbedingten Einwirkungen, soweit sie in Betracht zu ziehen sind, wie Erdbeben, Erdrutsch, Sturm, Hochwasser, Sturmflut, sowie möglichen Einwirkungen von biologischen Organismen (z.B. Vogelschwärme, Muschelbewuchs in Kühlwasserleitungen) oder sonstige Einwirkungen von außen wie Störmaßnahmen Dritter, Flugzeugabsturz, Einwirkungen von gefährlichen, insbesondere explosionsfähigen Stoffen und Bergschäden, erfüllen können. Der Auslegung dieser Anlagenteile sind zugrunde zu legen:

1. die jeweils folgenschwersten naturbedingten Einwirkungen oder sonstigen Einwirkungen von außen, die nach dem Stand von Wissenschaft und Technik an dem betreffenden Standort berücksichtigt werden müssen;

2. Kombinationen mehrerer naturbedingter Einwirkungen oder sonstiger Einwirkungen von außen wie Kombinationen dieser Einwirkungen mit Störfällen, soweit das gleichzeitige Eintreten auf Grund der Wahrscheinlichkeit und des Schadensausmaßes in Betracht gezogen werden muß.

Die erkennbare zukünftige Entwicklung der Eigenschaften des Standortes muß berücksichtigt werden.

Kriterium 2.7: Brand- und Explosionsschutz

Es sind die erforderlichen Maßnahmen zur Verhütung von Bränden und Explosionen in der Anlage zu treffen. Die sicherheitstechnisch wichtigen Anlagenteile müssen so beschaffen und angeordnet sein, daß die Erfüllung ihrer Aufgaben durch Brände und Explosionen nicht verhindert wird.
Geeignete Einrichtungen zur frühzeitigen Erkennung und Bekämpfung von Bränden und Explosionsgefahren müssen vorhanden sein. Sie müssen so beschaffen und gesichert sein, daß sie nicht ihrerseits bei Störungen und Schäden an ihnen oder bei Fehlbedienung die Funktionsfähigkeit sicherheitstechnisch wichtiger Anlagenteile - gegebenenfalls unter Berücksichtigung von deren Redundanzen - beeinträchtigen.

Kriterium 2.8. Zugangskontrolle, abzusperrende Bereiche

Das gesamte Anlagengelände und zusätzlich Anlagenbereiche innerhalb und außerhalb desselben, die besonders schutzbedürftig sind, müssen gegen den Zutritt Unbefugter gesichert sein. Die Zugänge zu diesen Bereichen müssen so eingerichtet sein, daß eine lückenlose Überwachung des Personen- und Güterverkehrs durchführbar ist.

Kriterium 2.9: Fluchtwege und Kommunikationsmittel

Die Anlage muß einfache, deutlich und dauerhaft gekennzeichnete und ausfallsicher beleuchtete Fluchtwege haben.
Es müssen geeignete Alarmeinrichtungen und Kommunikationsmittel vorhanden sein, durch die allen in der Anlage anwesenden Personen von mindestens einer zentralen Stelle aus Anweisungen für das Verhalten bei Störfällen gegeben werden können.
Die für die Sicherheit des bestimmungsgemäßen Betriebs, die Beherrschung von Störfällen und darüber hinaus auch bei unvorhersehbaren Ereignisabläufen erforderliche Kommunikation innerhalb der Anlage und nach außerhalb muß jederzeit gewährleistet sein.

Kriterium 2.10: Stillegung und Beseitigung

Die Anlage muß so beschaffen sein, daß sie unter Einhaltung der Strahlenschutzbestimmungen stillgelegt werden kann. Ein Konzept für eine Beseitigung nach der endgültigen Stillegung unter Einhaltung der Strahlenschutzbestimmungen muß im Verlauf der Planung und der Errichtung der Anlage erstellt werden.

Abschnitt 3

Kriterium 3.1: Reaktorkernauslegung

Der Reaktorkern muß hinsichtlich seines Aufbaus und seiner Leistungsentwicklung so ausgelegt und hergestellt sein, daß im Zusammenwirken mit den übrigen Systemen der Gesamtanlage die für den bestimungsgemäßen Betrieb und für Störfälle jeweils spezifizierten Grenzwerte der Aktivitätsfreisetzung aus den Brennelementen und der Belastung sicherheitstechnisch wichtiger Anlagenteile und Systeme, insbesondere des Kernaufbaus und der Kerneinbauten, im Hinblick auf Abschalt- und Kühlbarkeit des Reaktorkerns eingehalten werden.

Bei der Auslegung des Reaktorkerns ist von den für den jeweils betrachteten Auslegungsfall ungünstigsten Leistungsdichteverteilungen, Wärmespeicherungs- und Wärmetransportvorgängen sowie Last- oder Belastungsvorgeschichten der Anlage auszugehen.
Weiterhin sind bei der Festlegung der Auslegungsdaten Sicherheitszuschläge zu berücksichtigen, die die Fehler verwendeter Daten, Rechenmodelle oder Messungen ausreichend abdecken.

Kriterium 3.2: Rückkopplungseigenschaften des Reaktorkerns

Der Reaktorkern muß so ausgelegt sein, daß aufgrund inhärenter Rückkopplungseigenschaften Leistungsexkursionen, die sich durch in Betracht zu ziehende Reaktivitätsanstiege ergeben, soweit abgefangen werden, daß im Zusammenwirken mit anderen inhärenten Eigenschaften und dem Sicherheitssystem sicherheitstechnisch bedeutsame Schäden im Reaktor, am Aktivitätseinschluß und im Kühlmittelkreislauf nicht eintreten.

Bei der Auslegung ist insbesondere dafür Sorge zu tragen, daß der prompte Anteil des Temperaturkoeffizienten der Reaktivität im bestimmungsgemäßen Betrieb und bei Störfällen hinreichend negativ ist.

Kriterium 3.3: Einbauten des drucktragenden Behälters

Die Einbauten des drucktragenden Behälters müssen so beschaffen und angeordnet sein, daß im bestimmungsgemäßen Betrieb die jeweils spezifizierten Grenzwerte für ihre Belastung nicht überschritten werden. Darüber hinaus müssen die Einbauten so beschaffen sein, daß bei störfallbedingten Belastungen die sichere Abschaltung des Reaktors, die ausreichende Abfuhr der Nachwärme und die sichere Einschließung des Reaktorkühlmittels gewährleistet sind.

Kriterium 3.4: Systeme zur Steuerung und Abschaltung des Kernreaktors [1)]

Die Systeme zur Steuerung und Abschaltung des Kernreaktors sind so auszulegen, daß alle im bestimmungsgemäßen Betrieb und bei Störfällen auftretenden Reaktivitätsänderungen so beherrscht werden, daß die für diese Fälle jeweils spezifizierten Grenzwerte für die Reaktoranlage bei den in Betracht zu ziehenden Transienten nicht überschritten werden.
Reaktivitätshub und Reaktivitätsrampe von reaktivitätssteuernden Einrichtungen sind so zu begrenzen, daß bei fehlerhaftem Fahrbefehl die jeweils spezifizierten Grenzwerte für die Reaktoranlage im Zusammenwirken mit dem Sicherheitssystem eingehalten werden. Der Reaktorkern und die Systeme zur Steuerung müssen so aufeinander abgestimmt sein, daß Schwankungen des Neutronenflusses, die zu einem Überschreiten der für sicherheitstechnisch wichtige Komponenten und für die Aktivitätsfreisetzung aus den Brennelementen spezifizierten Grenzwerte führen können, entweder nicht möglich sind oder zuverlässig festgestellt und so begrenzt werden, daß die genannten Grenzwerte nicht überschritten werden.
Es sind zwei Abschaltsysteme vorzusehen. Beide können zur Steuerung des Kernreaktors herangezogen werden, soweit dadurch die Erfüllung der nachfolgenden Aufgaben nicht beeinträchtigt wird.

Die Auslegung der Abschaltsysteme hat im Hinblick auf die Reaktivität für den ungünstigsten Abbrandzustand zu erfolgen.

1) Eine Präzisierung dieses Kriteriums im Hinblick auf den möglichen Ausfall des ersten Abschaltsystems bei Betriebstransienten ist vorgesehen.

(1) Erstes Abschaltsystem:

Das erste Abschaltsystem muß für sich allein in der Lage sein, den Kernreaktor aus dem bestimmungsgemäßen Betrieb heraus und bei Störfällen auch bei Ausfall von reaktivitätswirksamsten Komponenten [2)] in einer durch die Störfallanalyse begründeten Zeit unterkritisch zu machen und solange zu halten, daß die jeweils spezifizierten Grenzwerte der Reaktoranlage nicht überschritten werden. Die zum Zeitpunkt der Abschaltung aufzubringende Abschaltreaktivität ist so zu bemessen, daß die Einhaltung der jeweils spezifizierten Grenzwerte durch die automatisch eingeleiteten Maßnahmen ausreichend lang gewährleistet wird.

Der Ausfall von reaktivitätswirksamsten Komponenten ist zu berücksichtigen, soweit er durch eine einzelne Störung im Abschaltsystem verursacht werden kann; es ist wenigstens ein Steuerelement als nicht verfügbar anzunehmen. Bei der physikalischen Auslegung sind ausreichende Sicherheitszuschläge zu berücksichtigen.

Der Ausfall von reaktivitätswirksamsten Komponenten braucht nicht berücksichtigt zu werden, wenn beide Abschaltsysteme einschließlich der Anregung durch das Reaktorschutzsystem, insbesondere hinsichtlich der Abschaltcharakteristik, der Wirksamkeit und des Zeitverhaltens gleichwertig sind.

2) Beispiel für Komponente: Steuerelement

(2) Zweites Abschaltsystem:

Das zweite Abschaltsystem ist von dem ersten Abschaltsystem unabhängig und konstruktiv verschiedenartig auszuführen. Es muß für sich allein in der Lage sein, den Kernreaktor aus jedem Normalbetriebszustand heraus unterkritisch zu machen und ihn in dem für die Reaktivitätsbilanz ungünstigsten Zustand, der unter den in Betracht zu ziehenden Umständen im System möglich ist, beliebig lange unterkritisch zu halten. Dabei ist im abgeschalteten Zustand eine ausreichende Abschaltreaktivität während der Kernlebensdauer unter Berücksichtigung von Sicherheitszuschlägen bei der physikalischen Auslegung zu gewährleisten.

Ist das erste Abschaltsystem allein nicht in der Lage, den Reaktorkern im für die Reaktivitätsbilanz ungünstigsten Zustand beliebig lange unterkritisch zu halten, so ist der für diese Aufgabe zusätzlich zum ersten Abschaltsystem benötigte Teil des zweiten Abschaltsystems auf die dabei in Betracht zu ziehenden Störfallbedingungen auszulegen. In diesem Fall gelten hinsichtlich des Ausfalls von Abschaltelementen die Ausführungen des Punktes (1) für die Gesamtheit der benötigten Abschaltsysteme [3].

[3] Nach Maßgabe der jeweiligen Störfallbedingungen ist festzulegen, ob ein Teil des zweiten Abschaltsystems automatisch angeregt werden muß.

Inhärente sicherheitsgerichtete Eigenschaften können bei der Auslegung der Abschaltsysteme in dem Umfang als Ersatz für anlagentechnische Maßnahmen dienen, wie ihre Wirksamkeit nachgewiesen wird.

Wenn die Abschaltsysteme gemeinsame Komponenten mit dem System zur Steuerung des Kernreaktors haben, ist sicherzustellen, daß weder eine Funktion des Steuerungssystems noch ein Fehler im Steuerungssystem das bestimmungsgemäße Funktionieren der Abschaltsysteme verhindert.

Abschnitt 4

Kriterium 4.1: Einschließung des Reaktorkühlmittels

Zur Verhinderung einer Freisetzung von radioaktiven Stoffen ist eine Einschließung des Reaktorkühlmittels vorzusehen.
Grundsätzlich hat diese - abhängig vom gewählten Anlagenkonzept - zu umfassen:

- die dichtende Membran (Liner) des drucktragenden Behälters (siehe Kriterium 4.3),
- die Durchführungen durch den drucktragenden Behälter einschließlich ihrer Abschlüsse, die Dicht- oder Drucktragefunktion übernehmen,
- die Reaktorkühlmittel führenden Rohrleitungen, die den Liner oder die Behälterabschlüsse durchdringen, einschließlich der ersten Absperrarmatur,
- die Sperrmedienleitungen einschließlich der ersten Absperrarmatur,
- die Rohrleitungen, die sich innerhalb der dichtenden Membran befinden und von außen mit Reaktorkühlmittel beaufschlagt werden (z.B. Wärmetauscher im Primärkreislauf), bis zum Anschluß an den Behälterabschluß oder Liner.

Für den Bereich der dichtenden Membran hat der drucktragende Behälter die drucktragende Funktion zu übernehmen (siehe Kriterium 4.3).

Kriterium 4.2: Auslegungsgrundlagen der Einschließung des Reaktorkühlmittels

Die Einschließung des Reaktorkühlmittels muß mit hinreichendem Sicherheitsabstand so ausgelegt werden, daß sie den während des bestimmungsgemäßen Betriebes und bei Störfällen auftretenden maximalen Belastungen standhält und die erforderliche Dichtheit besitzt. Alle Komponenten der Einschließung sind konstruktiv so zu gestalten, daß die erforderlichen erstmaligen Prüfungen bei der Herstellung oder am Aufstellungsort möglich sind. Mindestens die Komponenten der Einschließung, die drucktragende Funktion haben, sind konstruktiv so zu gestalten, daß sie gemäß Kriterium 2.2 wiederkehrend prüfbar sind.

Durch Werkstoffwahl, sachgerechte Formgebung, Schweißung und gegebenenfalls Wärmebehandlung muß an allen Stellen der Einschließung im bestimmungsgemäßen Betrieb und bei Störfällen ein ausreichend zäher Werkstoffzustand während der Lebensdauer der Anlage erhalten bleiben. Die Durchdringungen des drucktragenden Behälters sind gegen Austreiben zu sichern. Sind nach dem Stand von Wissenschaft und Technik Schäden an den Abschlüssen zu unterstellen, so ist der zulässige störfallbedingte maximale Leckquerschnitt zu spezifizieren; die Begrenzung auf diesen Wert ist durch technische Maßnahmen sicherzustellen. Die Komponenten der Einschließung müssen so ausgelegt werden, daß ein Versagen, welches zu einer unzulässigen Beeinträchtigung der Funktionsfähigkeit sicherheitstechnisch wichtiger Anlagenteile führen könnte, ausgeschlossen werden kann. Es sind Einrichtungen für eine Überwachung auf etwaige Leckagen aus der Einschließung des Kühlmittels während des Betriebes vorzusehen.

Kriterium 4.3: Drucktragender Behälter in vorgespannter Konstruktion

An den drucktragenden Behälter in vorgespannter Konstruktion werden folgende Anforderungen gestellt:

(1) Der Behälter ist so auszulegen, daß die drucktragende Funktion im bestimmungsgemäßen Betrieb und bei Störfällen sichergestellt ist.

(2) Der drucktragende Behälter ist so auszulegen, daß bei den im bestimmungsgemäßem Betrieb und bei Störfällen auf ihn einwirkenden Lasten der Aktivitätseinschluß durch die Komponenten der Einschließung des Reaktorkühlmittels (siehe Kriterium 4.1) in Verbindung mit dem drucktragenden Behälter gewahrt bleibt.

(3) Der drucktragende Behälter ist vor unzulässigen Temperaturbeanspruchungen zu schützen. Dazu ist erforderlichenfalls ein ausreichender Wärmeschutz vorzusehen, dessen Wirksamkeit zu überwachen ist.

(4) Der drucktragende Behälter ist durch eine Betriebsinstrumentierung auf die Einhaltung sicherheitstechnisch bedeutsamer Auslegungswerte zu überwachen.

(5) Durch geeignete Maßnahmen ist sicherzustellen, daß bei möglichen Kühlmittelleckagen durch den Liner die drucktragende Funktion des Behälters nicht gefährdet wird.

(6) Bei der Auslegung des drucktragenden Behälters sind induzierte Belastungen aus Einwirkungen von außen (Druckwelle, Flugzeugabsturz, Auslegungs- und Sicherheitserdbeben) zu berücksichtigen.

(7) Ein Überschreiten der Tragfähigkeitsgrenze des drucktragenden Behälters muß mit ausreichendem Sicherheitsabstand ausgeschlossen werden. Hierzu ist zusätzlich zu den Anforderungen (1) bis (6) der Grenztragfähigkeitsnachweis für den drucktragenden Behälter zu erbringen.

Abschnitt 5

Kriterium 5.1: Nachwärmeabfuhr im bestimmungsgemäßen Betrieb

Ein zuverlässiges System mit ausreichender Redundanz zur Nachwärmeabfuhr im bestimmungsgemäßen Betrieb muß vorhanden sein. Es muß so beschaffen sein, daß die Einhaltung der für den bestimmungsgemäßen Betrieb spezifizierten Grenzwerte für die Aktivitätsfreisetzung aus den Brennelementen und für die Belastung der Komponenten der Kühlgasführung, der Abschaltung, der Einschließung des Reaktorkühlmittels und des Sicherheitseinschlusses während ihrer gesamten Einsatzzeit gewährleistet ist. Das System für die Nachwärmeabfuhr im bestimmungsgemäßen Betrieb kann teilweise oder vollständig identisch sein mit dem Nachwärmeabfuhrsystem nach Störfällen, falls dieses hierfür geeignet und ausgelegt ist.

Kriterium 5.2: Nachwärmeabfuhr nach Störfällen

Ein zuverlässiges und redundantes System für die Nachwärmeabfuhr nach Störfällen muß vorhanden und so beschaffen sein, daß bei den zu unterstellenden störfallauslösenden Ereignissen und den in Betracht kommenden Ausgangszuständen und Störfallrandbedingungen (z.B. Bruchgrößen, Bruchlagen und Transienten im Reaktorkühlsystem) die Nachwärme so abgeführt wird, daß die für Störfälle spezifizierten Grenzwerte [1)] für die Aktivitätsfreisetzung aus den Brennelementen und für die Belastung der sicherheitstechnisch wichtigen Anlagenteile, die zur weiteren Beherrschung der Störfälle notwendig sind, nicht überschritten werden.

1) Die Grenzwerte sind unter Berücksichtigung der jeweils erwarteten Eintrittshäufigkeit der Störfälle zu spezifizieren.

Im einzelnen müssen folgende Grundsätze erfüllt werden:

- Das System muß auch während Instandhaltungsvorgängen bei gleichzeitigem Auftreten eines Einzelfehlers seine sicherheitstechnische Aufgabe erfüllen können (vergleiche das Einzelfehlerkonzept). Kann aufgrund inhärenter sicherheitstechnischer Eigenschaften des Reaktors die Nachwärmeabfuhr längere Zeit unterbrochen werden, ohne daß spezifizierte Grenzwerte überschritten werden, so braucht wähend Instandsetzung und Wartung, die innerhalb dieser Zeit abgeschlossen werden, das Auftreten eines Einzelfehlers nicht unterstellt werden.

- Das System ist so auszulegen, daß bei Störfällen mit Fremdmedieneinbruch in den Reaktorkern chemische Reaktionen und ihre Auswirkungen so begrenzt [2] werden, daß die Funktion des Systems und anderer sicherheitstechnischer Einrichtungen nicht unzulässig beeinträchtigt wird.

[2] Begrenzung z.B. durch Temperaturabsenkung

- Teile des Betriebssystems können in das Nachwärmeabfuhrsystem einbezogen werden, wenn

(1) ihre Funktionsbereitschaft während des bestimmungsgemäßen Betriebes überprüft werden kann,

(2) die benötigten Teilsysteme die erforderliche Zuverlässigkeit besitzen,

(3) die Betriebsbereitschaft der für das Nachwärmeabfuhrsystem vorgesehenen Teile des Betriebssystems innerhalb einer durch den Störfall vorgegebenen Zeitspanne nach Störfalleintritt gewährleistet ist,

(4) und diese Teile entsprechend ihrer Bedeutung für das Nachwärmeabfuhrsystem ausgelegt, gefertigt und ausreichend wiederkehrend prüfbar sind.

Bei der Berechnung der Nachwärmeabfuhr nach einem Druckentlastungsstörfall ist nach der Ausströmphase entweder vom Atmosphärendruck oder von dem Ausgleichsdruck auszugehen, der sich unter den im ungünstigsten Fall zu unterstellenden Bedingungen [3] und bei Abzug eines Sicherheitsabschlages vom so berechneten Druck ergibt. Kann vornehmlich unter Ausnutzung inhärenter sicherheitsgerichteter Eigenschaften die Nachwärme abgeführt oder die Nachwärmeabfuhr längere Zeit unterbrochen werden, ohne daß spezifizierte Grenzwerte überschritten werden, so können diese Eigenschaften bei der Auslegung des Nachwärmeabfuhrsystems berücksichtigt werden und an Stelle von systemtechnischen Maßnahmen treten.

3) Im ungünstigsten Fall zu unterstellende Bedingungen sind z.B.:
- minimal zu unterstellende Nachwärme,
- maximal spezifizierte Leckrate aus dem Sicherheitseinschluß,
- verstärkter Wärmeübergang in die Strukturen.

Abschnitt 6

Kriterium 6.1: Reaktorschutzsystem

Die Anlage muß mit einem zuverlässigen [1] Reaktorschutzsystem ausgerüstet sein, das bei Erreichen festgelegter Werte für Prozeßvariablen Schutzaktionen auslöst.
Es muß so beschaffen sein, daß es auch während Instandhaltungsvorgängen bei gleichzeitigem Auftreten eines Einzelfehlers im System seine sicherheitstechnische Aufgabe erfüllen kann. Von Hand oder durch die betriebliche Steuerung und Regelung gegebene Befehle dürfen notwendige Schutzaktionen weder beeinträchtigen noch verhindern können.

Das Reaktorschutzsystem ist so auszulegen und aufzubauen, daß es die durch die Störfallanalyse gestellten Anforderungen erfüllt.
Für jedes zu beherrschende Ereignis sollen grundsätzlich mindestens zwei Anregekriterien zur Verfügung stehen. Als Anregekriterien sollen verschiedene Prozeßvariablen herangezogen werden.

1) Als Mittel zur zuverlässigen Auslegung des Reaktorschutzsystems sollen vorzugsweise angewendet werden:
- redundante Auslegung von Komponenten, Baugruppen und Untersystemen, räumlich getrennte Installation entsprechend dem Wirkungsbereich möglicher versagensauslösender Ereignisse,
- Fehler oder Ausfälle sollten weitestgehend selbstmeldend sein,
- Verwendung von Geräten unterschiedlicher Bauart (Diversitätsprinzip),
- Anpassung der Komponenten an die möglichen Umgebungsbedingungen.

Ist die Forderung nach zwei (x) Anregekriterien nicht zu erfüllen, so muß die Meßwerterfassung der allein herangezogenen Prozeßvariablen im Verhältnis zur Meßwerterfassung bei Heranziehen verschiedener Prozeßvariablen entsprechend sicherheitstechnisch höherwertig [2) aufgebaut bzw. ausgelegt sein.
Prozeßvariablen, die aus anderen Prozeßvariablen abgeleitet sind oder erst im Zusammenwirken mit weiteren Prozeßvariablen (z.B. UND-Verknüpfung) die Anregekriterien für Schutzaktionen ergeben, sind als eine Sicherheitsvariable anzusehen.

Die mechanischen und elektrischen Geräte der Anregekanäle (Meßfühler bis einschließlich Grenzwertgeber) des Reaktorschutzsystems dürfen grundsätzlich nicht für Funktionen im Rahmen der Reaktorregelung verwendet werden. Ausnahmen sind nur zulässig, wenn sie aufgrund der technischen Eigenart des Reaktorschutzsystems oder der Meß-, Steuer- und Regelsysteme erforderlich sind, und wenn im Rahmen der Störfallanalyse nachgewiesen wurde, daß der gemeinsame Ausfall der Meßkanäle für Steuer- und Regelsysteme und für das Reaktorschutzsystem als Störfall beherrscht wird.

(x) auf verschiedenen Prozeßvariablen basierenden

2) Als Maßnahmen zum höherwertigen Aufbau oder zur höherwertigen Auslegung sollen vorzugsweise angewendet werden:
- Verwendung von Geräten unterschiedlicher Bauart (Diversitätsprinzip),
- Grenzbelastungsprüfungen, Prüfzyklen.

Abschnitt 11

Kriterium 11.1: Handhabung und Lagerung von Kernbrennstoffen und sonstigen radioaktiven Stoffen

In der Anlage müssen Einrichtungen vorhanden sein, die eine sichere Handhabung und Lagerung der Kernbrennstoffe und sonstiger radioaktiver Stoffe ermöglichen. Diese Einrichtungen müssen so beschaffen, angeordnet und abgeschirmt sein, daß ein Kritikalitätsstörfall und eine unzulässige Strahlenexposition des Personals und in der Umgebung ausgeschlossen werden.

Durch entsprechende Gestaltung der Handhabungseinrichtungen muß sichergestellt sein, daß Kernbrennstoffe sicher in den Kern eingebracht und am vorgesehenen Ort positioniert werden können.
Eine unzulässige Freisetzung radioaktiver Stoffe aus bestrahlten Kernbrennstoffen während der Handhabung sowie aus den Handhabungseinrichtungen im bestimmungsgemäßen Betrieb und bei Störfällen ist durch geeignete Maßnahmen zu verhindern. Bereiche, an denen die Möglichkeit von Leckagen besteht, sind zu überwachen.

Die Einrichtungen zur Lagerung bestrahlter Kernbrennstoffe müssen über ausreichende Lagerkapazität verfügen. Bei der Festlegung der Lagerkapazität sind Gesichtspunkte, wie eine eventuell erforderliche Auslagerung von Brennelementen aus dem Reaktorkern, wiederkehrende Prüfungen und Instandsetzungsvorgänge, zu berücksichtigen. Es ist sicherzustellen, daß die Nachwärme aus den Lagereinrichtungen im bestimmungsgemäßen Betrieb und bei Störfällen zuverlässig abgeführt wird.

Diejenigen Komponenten der Heliumreinigungsanlage sind nach den Auslegungsgrundlagen der Einschließung des Reaktorkühlmittels auszulegen (vergleiche Kriterien 4.2), deren Versagen zu unzulässigen radiologischen Auswirkungen führt.

Kriterium 10.3: Umgebungsüberwachung

Es müssen die personellen, organisatorischen und apparativen Voraussetzungen gegeben sein, um eine Strahlenschutzüberwachung der Umgebung im bestimmungsgemäßen Betrieb, bei Störfällen und Unfällen im erforderlichen Umfang hinreichend schnell, genau und zuverlässig durchführen zu können (siehe Kriterium 6.3). Insbesondere müssen vorhanden sein:

(1) Einrichtungen und Geräte zur Bestimmung von Dosis, Dosisleistung, Aktivitätskonzentration und Oberflächenkontamination sowie zur Bestimmung von Nukliden während des bestimmungsgemäßen Betriebs;

(2) Einrichtungen und Geräte zur Ermittlung der erforderlichen Information über Ortsdosen, Aktivitätskonzentrationen, Oberflächenkontaminationen und Nukliden bei etwaigen Freisetzungen radioaktiver Stoffe;

(3) Einrichtungen zur Messung meteorologischer Daten, die zur Bestimmung der Ausbreitungsverhältnisse erforderlich sind (vgl. Kriterium 6.3).

Kriterium 10.2: Aktivitätsüberwachung in Fortluft und Abwasser

In der Anlage müssen die personellen, organisatorischen und gerätetechnischen Voraussetzungen gegeben sein, um im erforderlichen Umfang Art, Menge und Konzentration der mit der Fortluft und dem Abwasser abzuleitenden radioaktiven Stoffe hinreichend genau und zuverlässig messen, registrieren sowie die Ableitung erforderlichenfalls begrenzen zu können. Insbesondere müssen ortsfeste Einrichtungen vorhanden sein, mit denen es möglich ist, die Ableitung und Freisetzung flüssiger, gasförmiger und aerosolförmiger radioaktiver Stoffe im bestimmungsgemäßen Betrieb kontinuierlich und getrennt zu überwachen und zu registrieren. Es müssen Einrichtungen vorhanden sein, die bei Störfällen die Ableitung oder Freisetzung flüssiger, gasförmiger und aerosolförmiger radioaktiver Stoffe messen und registrieren, so daß eine Berechnung der Auswirkungen auf die Umgebung möglich ist. Alle Meßwerte sollten an den Meßeinrichtungen angezeigt werden. Sie müssen in der Warte registriert werden sowie entweder angezeigt oder abgefragt [1)] werden können.

1) Das Abfragen soll in Abweichung von den Regeln KTA 1503.1 und 1504 erlaubt sein.

(5) Einrichtungen zur Messung von Personendosen sowie der Kontamination von Personen und Gegenständen,

(6) geeignete Laboreinrichtungen zur Auswertung und Analyse radioaktiver Proben.

Meßwerte der unter (1) bis (3) genannten ortsfesten Einrichtungen sind in der Warte oder in einem Wartennebenraum zu registrieren, müssen vor Ort angezeigt werden und in der Warte entweder angezeigt oder abgefragt [1] werden können. Das Überschreiten des Gefahrenmeldewertes und der Ausfall des Gerätes sind vor Ort und in der Warte anzuzeigen.

1) Das Abfragen soll in Abweichung von der Regel KTA 1501 erlaubt sein.

Abschnitt 10

Kriterium 10.1 Strahlenschutzüberwachung

In der Anlage müssen die personellen, organisatorischen, räumlichen und apparativen Voraussetzungen gegeben sein, um eine Strahlenschutzüberwachung in der Anlage bei bestimmungsgemäßem Betrieb, bei Störfällen und bei Unfällen im erforderlichen Umfang hinreichend genau und zuverlässig zu gewährleisten.
Insbesondere müssen vorhanden sein:

(1) ortsfeste Einrichtungen zur Messung von Ortsdosisleistungen,

(2) ortsfeste Einrichtungen zur Messung der Konzentration radioaktiver Stoffe in der Raumluft von Raumgruppen oder Räumen, in denen eine entsprechende Überwachung zum Schutze von Personen oder zur frühzeitigen Entdeckung etwaiger freigesetzter radioaktiver Stoffe notwendig ist,

(3) ortsfeste Einrichtungen zur Messung der Konzentration radioaktiver Stoffe in Kreisläufen, in denen eine entsprechende Überwachung zur frühzeitigen Entdeckung etwaiger freigesetzter radioaktiver Stoffe notwendig ist,

(4) tragbare Meßgeräte zur Ermittlung von Ortsdosisleistungen sowie Konzentrationen und Art radioaktiver Stoffe in Luft und Wasser,

Die lüftungstechnischen Anlagen müssen so ausgelegt und beschaffen und mit den Eigenschaften der übrigen Anlagenteile so abgestimmt sein, daß im bestimmungsgemäßen Betrieb und bei Störfällen die hierfür jeweils als zulässig spezifizierten Werte für die Raumluftzustände und für die Ableitung oder etwaige Freisetzung radioaktiver Stoffe nicht überschritten werden können. Umluftanlagen sind in geeigneter Weise mit Fortluftanlagen zu kombinieren, so daß die Strahlenexposition von Personen innerhalb und außerhalb der Anlage unter Beachtung der Regeln von Wissenschaft und Technik auch unterhalb der zugelassenen Werte so gering wie möglich gehalten wird.
Soweit die Konzentration radioaktiver Stoffe in der Luft bestimmter Räume so groß werden kann, daß jeweils als zulässig spezifizierte Werte überschritten werden, müssen die zugehörigen lüftungstechnischen Anlagen über Luftfilteranlagen verfügen. Eine Schaltung der lüftungstechnischen Anlagen so, daß die Abluft nur im Bedarfsfall über Filteranlagen geführt wird, ist zulässig. Die Luftfilteranlagen müssen hinreichend zuverlässig und so beschaffen sein, daß sie unter den jeweiligen Einsatzbedingungen den erforderlichen Abscheidegrad haben. Zur Überprüfung ihres Zustandes müssen die erforderlichen Einrichtungen vorhanden sein.

Abschnitt 9

Kriterium 9.1: Lüftungstechnische Anlagen

Die Anlage muß über zuverlässige lüftungstechnische Anlagen für folgende Räume verfügen:

1. Räume, in denen im bestimmungsgemäßen Betrieb oder bei Störfällen im Jahresdurchschnitt im Kubikmeter der Raumluft höhere Aktivität als
   - für Radionuklide und Radionuklidgemische, bei denen die Inkorporation grenzwertbestimmend ist, 1/7300 der Werte der Anlage IV Tabelle IV 1 und IV 2, Spalte 5
   - für Radionuklide, bei denen die Submersion grenzwertbestimmend ist, die Werte der Anlage IV Tabelle IV 4, Spalte 5

   der Strahlenschutzverordnung auftreten können; Ausnahmen sind zulässig, wenn die Vorschriften der §§ 45 und 46 Abs. 1 bis 3 und 5 der Strahlenschutzverordnung eingehalten werden;

2. Räume, in denen für den bestimmungsgemäßen Betrieb als zulässig spezifizierte Werte für die Raumluftzustände anders nicht eingehalten werden können, oder in denen sicherheitstechnisch wichtige Anlagenteile mit Luftkühlung auch bei Störfällen arbeiten müssen;

3. Räume, in denen die Luft durch ein Inertgas ersetzt ist, oder in denen aus Gründen des Arbeitsschutzes bestimmte Raumluftzustände eingehalten werden müssen.

Kriterium 8.5: Bauliche Schutzvorkehrungen zur Rückhaltung radioaktiver Stoffe

Gebäudeteile, in denen radioaktiv kontaminierte Flüssigkeiten anfallen können, sind innenseitig mit einer flüssigkeitsdichten Sperre zu versehen, so daß diese Flüssigkeiten, die z.B. beim Versagen von Behältern und Rohrleitungen freigesetzt werden, zurückgehalten werden. Die Sperre ist dekontaminierbar zu gestalten.

Kann das Versagen der innenliegenden Sperre bei Störfällen von innen oder bei äußeren Einwirkungen nicht ausgeschlossen werden und ist kein Sicherheitseinschluß im Sinne von Kriterium 8.1 vorhanden, so sind bauliche Schutzvorkehrungen vorzusehen, die hinsichtlich der Rückhaltung kontaminierter Flüssigkeiten eine ausreichend zuverlässige Sperre zum Baugrund darstellen.
Zur Sicherstellung der andauernden Funktionsfähigkeit dieser Bauteile sind bei der Errichtung zusätzliche qualitätssichernde Maßnahmen für Bauart, Werkstoffe und Herstellung erforderlich (vergleiche Kriterium 2.2).

Die Funktionsfähigkeit der sicherheitstechnisch wichtigen Rohrleitungs- und Kabeldurchführungen muß auch unter störfallbedingten Umgebungsbedingungen und Störfallfolgelasten gewährleistet sein.

Die Absperrarmaturen müssen fernbetätigt oder selbsttätig schließen. Die Stellung der Absperrarmaturen muß von der Warte aus überwacht werden können. Die Redundanz von Absperrarmaturen muß sich in ihrer Energieversorgung fortsetzen.
Eine ausreichende räumliche Trennung ist erforderlich. Jede Absperrarmatur einer Redundanzgruppe muß für sich allein in der Lage sein, den Abschluß der betreffenden Rohrleitung zu gewährleisten.

Die Absperrarmaturen müssen auch bei störfallbedingten Umgebungsbedingungen und Störfallfolgelasten ihre sicherheitstechnische Aufgabe erfüllen.

Schleusen, Lüftungsklappen und Dichtkästen sind auf Leckagen zu überwachen.
Falls aus Gründen des Strahlenschutzes erforderlich, sind Leckagen kontrolliert abzuleiten.

Kriterium 8.4: Durchführungen durch den Sicherheitseinschluß

Rohrleitungen, die den Sicherheitseinschluß durchdringen und in Verbindung mit dem Kühlmittel oder als Bestandteil der Einschließung des Reaktorkühlmittels in Berührung mit dem Kühlmittel stehen, sowie Rohrleitungen, die den Sicherheitseinschluß durchdringen und in Verbindung mit der Innenatmosphäre des Sicherheitseinschlusses stehen, müssen grundsätzlich zwei Absperrarmaturen haben. Davon ist eine Armatur innerhalb und eine außerhalb in der Nähe des Sicherheitseinschlusses anzuordnen. Ausnahmen von diesen Forderungen sind zulässig, wenn dies wegen der technischen Eigenart oder der Betriebsweise der betreffenden Rohrleitung notwendig ist und die sicherheitstechnische Funktion des Sicherheitseinschlusses nicht beeinträchtigt wird.

Rohrleitungen, die den Sicherheitseinschluß durchdringen und den o.g. nicht zuzuordnen sind, müssen mindestens eine außerhalb des Sicherheitseinschlusses liegende Absperrarmatur haben. Die Auslegung der Absperrarmaturen und der betreffenden Rohrleitungen bis zur äußeren Absperrarmatur muß mindestens der Auslegung des Sicherheitseinschlusses entsprechen.

Rohrleitungsdurchführungen durch den Sicherheitseinschluß müssen denselben Auslegungsanforderungen genügen, die für den Sicherheitseinschluß selbst gelten. Diese Forderung gilt entsprechend für Kabeldurchführungen.

Die für die Beherrschung von Störfällen notwendigen Einrichtungen innerhalb des Sicherheitseinschlusses sind auf störfallbedingte Umgebungsbedingungen und Störfallfolgelasten auszulegen.

Kriterium 8.3: Druck- und Dichtheitsprüfungen des Sicherheitseinschlusses

Der Sicherheitseinschluß muß so ausgelegt und beschaffen sein, daß bei der Erstprüfung eine Druckprüfung beim Prüfdruck und eine Dichtheitsprüfung beim Auslegungs- oder Prüfdruck durchgeführt werden kann.

Der Prüfdruck für die erstmalige Druckprüfung muß grundsätzlich gleich dem Auslegungsdruck sein; der Prüfdruck kann nach Maßgabe des technischen Konzeptes höher angesetzt werden.

Wiederkehrende Prüfungen müssen bei solchen Drücken durchgeführt werden können, bei denen ein ausreichender Rückschluß auf die Leckrate bei den Auslegungsbedingungen möglich ist. Falls diese nicht mit dem Druck der erstmaligen Prüfung erfolgen, ist bei der erstmaligen Prüfung zu Vergleichszwecken die Leckrate beim für die wiederkehrenden Prüfungen vorgesehenen Druck aufzunehmen.

Kriterium 8.2: Auslegungsgrundlagen des Sicherheitseinschlusses

Der Sicherheitseinschluß ist einschließlich aller Durchführungen, Schleusen und Hilfseinrichtungen, soweit ihre Funktion zur Beherrschung von Störfällen notwendig ist, so auszulegen, daß er den statischen, dynamischen und thermischen Belastungen im bestimmungsgemäßen Betrieb sowie bei Störfällen soweit standhält, wie es zur Erfüllung seiner sicherheitstechnischen Aufgabe erforderlich ist. Der Sicherheitseinschluß muß bei Einwirkungen von außen seine Integrität [1)] bewahren.

Der Auslegungsdruck für den Sicherheitseinschluß ist ausgehend vom ungünstigsten Betriebszustand unter Berücksichtigung von Störfällen mit einem ausreichenden Sicherheitszuschlag festzulegen.

Die Bildung von explosionsfähigen Gasgemischen oder die Auswirkungen der Reaktion explosionsfähiger Gasgemische auf den Sicherheitseinschluß sind bei Störfällen soweit zu beschränken, daß die Erfüllung der sicherheitstechnischen Aufgabe des Sicherheitseinschlusses gewährleistet bleibt.

1) Die Forderung, daß der Sicherheitseinschluß seine Integrität bei Einwirkung von außen bewahren muß, bedeutet:

- Dicht- und Tragfähigkeit des Sicherheitseinschlusses sind zu gewährleisten, wenn der Nachweis, daß als Folge des Ereignisses unter Berücksichtigung betrieblicher Leckagen die Bestimmungen der Strahlenschutzverordnung für Störfälle eingehalten werden, nur unter der Voraussetzung der Dichtheit erbracht werden kann.
- Nur die Tragfähigkeit des Sicherheitseinschlusses braucht gewährleistet zu werden, wenn der Nachweis gelingt, daß auch ohne die Dichtheit die Bestimmungen der Strahlenschutzverordnung eingehalten werden.

Abschnitt 8

Kriterium 8.1: Sicherheitseinschluß des Kernreaktors [1)]

Die Anlage muß einen Sicherheitseinschluß besitzen, der seine sicherheitstechnische Aufgabe im bestimmungsgemäßen Betrieb und bei Störfällen erfüllen kann.

Der Sicherheitseinschluß muß im Zusammenwirken mit der Einschließung des Kühlmittels (siehe Kriterium 4.1) und weiteren Rückhaltebarrieren für radioaktive Stoffe gewährleisten, daß bei der zu unterstellenden Ableitung oder Freisetzung radioaktiver Stoffe in die Umgebung im bestimmungsgemäßen Betrieb und bei Störfällen die Forderungen des Atomgesetzes in Verbindung mit der Strahlenschutzverordnung eingehalten werden.

Die Einschließung des Reaktorkühlmittels (siehe Kriterium 4.1) muß im Sicherheitseinschluß untergebracht sein. Alle anderen Anlagenteile, die radioaktive Stoffe enthalten, müssen ebenfalls innerhalb des Sicherheitseinschlusses untergebracht werden, soweit die Forderungen der Strahlenschutzverordnung nicht durch andere geeignete Maßnahmen erfüllt werden.

Es muß ein zuverlässiger und zur Erreichung des Schutzzieles ausreichend schneller Abschluß der Durchführungen durch den Sicherheitseinschluß gewährleistet sein.

1) Zum Sicherheitseinschluß zählen das Bauwerk, Schleusen, Durchführungen und erforderlichenfalls Hilfssysteme zur Rückhaltung und Filterung etwaiger Leckagen aus dem Sicherheitseinschluß.

[redacted]

Abschnitt 7

Kriterium 7.1: Elektrische Energieversorgung des Sicherheitssystems

Für die elektrische Energieversorgung des Sicherheitssystems müssen Einspeisemöglichkeiten aus zwei Netzanschlüssen, aus dem Hauptgenerator und bei deren Ausfall aus einem autarken, zuverlässigen und redundanten Notstromsystem vorhanden sein.

Die Zuverlässigkeit der elektrischen Energieversorgung muß der Zuverlässigkeit entsprechen, die für die zu versorgenden Systeme gefordert wird.

Für die Energieversorgung des Sicherheitssystems müssen voneinander unabhängige, räumlich getrennte, redundante Notstromstränge mit Notstromerzeuger und Verteileranlagen vorhanden sein, so daß auch während Instandhaltungsvorgängen bei gleichzeitigem Auftreten eines Einzelfehlers eine sicherheitstechnisch ausreichende Notstromversorgung gewährleistet ist (vergleiche das Einzelfehlerkonzept).

Die Redundanz der Notstromstränge mit Notstromerzeuger und Verteileranlagen muß der Redundanz der maschinentechnischen Systeme entsprechen.

Bei Einwirkungen von außen muß die Funktionsfähigkeit einer ausreichenden Anzahl der redundanten Notstromstränge durch räumliche Trennung oder bauliche Schutzmaßnahmen gewährleistet (vergleiche das Einzelfehlerkonzept) und die Notstromversorgung auch hinreichend lange durch ausreichende Kraftstoffvorräte und sichere Verbrennungsluftansaugung sichergestellt sein.

Es muß gewährleistet sein, daß vor Ablauf der für den unterbrechungslosen Dauerbetrieb der Notstromerzeuger zulässigen Zeit der Notstrombedarf anderweitig gedeckt werden kann.

Die Schaltwarte und die Notsteuerstelle müssen so voneinander räumlich getrennt sein, voneinander unabhängig mit Energie versorgt werden und derart gegen Einwirkungen von außen geschützt sein, daß sie nicht gleichzeitig außer Funktion gesetzt werden können.

Die Schaltwarte ist so anzuordnen, zu gestalten, abzuschirmen, zu belüften und mit Notstrom zu versorgen, daß sich das Personal auch bei Störfällen, die von der Warte aus beherrscht und überwacht werden müssen, in der Schaltwarte aufhalten, sie verlassen und betreten kann. Diese Anforderungen gelten sinngemäß für die Notsteuerstelle.

von außen geschützten Anlagenbereich sind zusätzlich die Anzeigen und Aufzeichnungen der Störfallfolgeinstrumentierung vorzusehen, die nach einer Einwirkung von außen zur Erfüllung der an die Störfallfolgeinstrumentierung gestellten Aufgabe benötigt werden.

Kriterium 6.4. Schaltwarte und Notsteuerstelle

Es muß eine Schaltwarte vorhanden sein, von der aus die Anlage im bestimmungsgemäßen Betrieb sicher betrieben werden kann und von der aus Maßnahmen ergriffen werden können, um sie in einem sicheren Zustand zu halten oder sie in einen solchen zu überführen.

Die für den Betrieb des Sicherheitssystems und für die Beherrschung von Störfällen erforderlichen Steuerungsmaßnahmen und Schalthandlungen müssen grundsätzlich von der Schaltwarte aus vorgenommen werden können. Falls Maßnahmen vor Ort durchgeführt werden sollen, ist zusätzlich zu der Zeitspanne 1), in der alle erforderlichen Maßnahmen automatisch einzuleiten sind, eine ausreichende Zeitspanne als Wege- und Montagezeit unter Berücksichtigung der Störfallbedingungen einzuplanen.

Auch bei Ausfall der Einrichtungen der Schaltwarte oder deren Nebenräume, wie z.B. Rangierverteiler oder Elektronikraum, muß der Reaktor sicher abgeschaltet und nachgekühlt sowie der abgeschaltete Zustand und die Nachwärmeabfuhr überwacht werden können. Die hierfür erforderlichen Steuer- und Überwachungseinrichtungen sind in einer Notsteuerstelle unterzubringen.

1) Als Zeitspanne werden derzeitig 0,5 Stunden als angemessen angesehen.

Es ist eine Gliederung der Störfallinstrumentierung in Störfallablaufinstrumentierung und Störfallfolgeinstrumentierung vorzusehen. Eine Vermaschung ist zulässig.

Die Störfallablaufinstrumentierung ist so auszulegen, daß die zur Feststellung eines Störfallablaufs ausgewählten Sytem- und Komponentendaten für eine spätere Aufklärung der Ursache und der Belastungen während des Störfalls übersichtlich und dauerhaft dokumentiert werden.

Für die Störfallablaufinstrumentierung sind redundante Einrichtungen zur Registrierung ausgewählter System- und Komponentendaten in der Warte oder in Wartennebenräumen vorzusehen. Die Nichtverfügbarkeit einer Komponente der Störfallablaufinstrumentierung darf nicht dazu führen, daß die Registrierung verhindert wird.

Die Störfallfolgeinstrumentierung ist so auszulegen, daß die Daten, die nach Eintreten eines Störfalls oder eines Unfalls für die Beurteilung der Anlagensicherheit, der Wirksamkeit des Sicherheitssystems und für die Entscheidung über Notfallmaßnahmen innerhalb der Anlage bis hin zu Notfallschutzmaßnahmen notwendig sind, zuverlässig und ausreichend genau angezeigt sowie im erforderlichen Umfang dokumentiert werden.

Die Anzeige und Aufzeichnung jeder Meßgröße der Störfallfolgeinstrumentierung muß in der Schaltwarte der Anlage erfolgen. In einem gegen Einwirkungen [redacted] [redacted]

Kriterium 6.2: Betriebsführungs-, Überwachungs- und Meldeeinrichtungen

Die Anlage muß Einrichtungen zur Betriebsführung (Steuerung und Regelung), Überwachung und Meldung haben, die im bestimmungsgemäßen Betrieb jederzeit eine ordnungsgemäße Betriebsführung und einen ausreichenden Überblick über den Betriebszustand der Anlage ermöglichen.
Alle sicherheitstechnisch wichtigen Zustandsgrößen sind durch geeignete Einrichtungen zu registrieren.
Die verfahrenstechnische Redundanz ist in der Regel in den Einrichtungen zur Betriebsführung, Überwachung und Meldung fortzusetzen.

Es müssen Gefahrenmeldeeinrichtungen vorhanden sein, die Veränderungen des Betriebszustandes, aus denen sich eine Verminderung der Sicherheit ergeben könnte, rechtzeitig anzeigen.

Kriterium 6.3: Störfallinstrumentierung

Es ist eine Störfallinstrumentierung vorzusehen, die folgende allgemeine Anforderungen erfüllen muß:

- Sie muß vor, während und nach einem Störfall oder einem Unfall einen ausreichenden Überblick über den Betriebszustand ermöglichen und alle den Anlagenzustand beschreibenden wesentlichen Daten sowie die wichtigsten Wetterdaten anzeigen und ihre Dokumentation in der zeitlichen Reihenfolge gewährleisten.
- Sie muß eine Abschätzung der Auswirkungen auf die Umgebung ermöglichen (siehe auch Kriterium 10.2).
  Die Einrichtungen der Störfallinstrumentierung sind an eine unterbrechungslose Notstromversorgung anzuschließen.

Redundante Teile des Reaktorschutzsystems sollen grundsätzlich voneinander unabhängige Einrichtungen zur Meßwerterfassung und Signalverarbeitung besitzen; Verknüpfungsstellen dürfen die Redundanz und Auslösesicherheit des Systems nicht verschlechtern.
Die redundanten Teile des Reaktorschutzsystems sind räumlich so voneinander zu trennen, derart mit elektrischer Energie und den erforderlichen Medien zu versorgen, daß Störungen innerhalb eines der Teilsysteme nicht gleichzeitig die Funktion der übrigen Systeme beeinträchtigen.

Das Reaktorschutzsystem muß so ausgelegt, ausgeführt und betrieben werden, daß es auch bei Störfällen im Reaktorschutzsystem keine Aktionen auslöst, die die Reaktoranlage in einen gefährlichen Zustand überführen können.
Das Reaktorschutzsystem ist in den für den jeweiligen Anlagenzustand erforderlichen Teilsystemen betriebsbereit zu halten.

FORSCHUNGSZENTRUM JÜLICH GmbH KFA

Fortschritte in der Energietechnik

für eine wirtschaftliche, umweltschonende und schadenbegrenzende Energieversorgung

Prof. Dr. Rudolf Schulten
zum 70. Geburtstag

herausgegeben von
K. Kugeler · H. Neis · G. Ballensiefen

The Chernobyl Accident

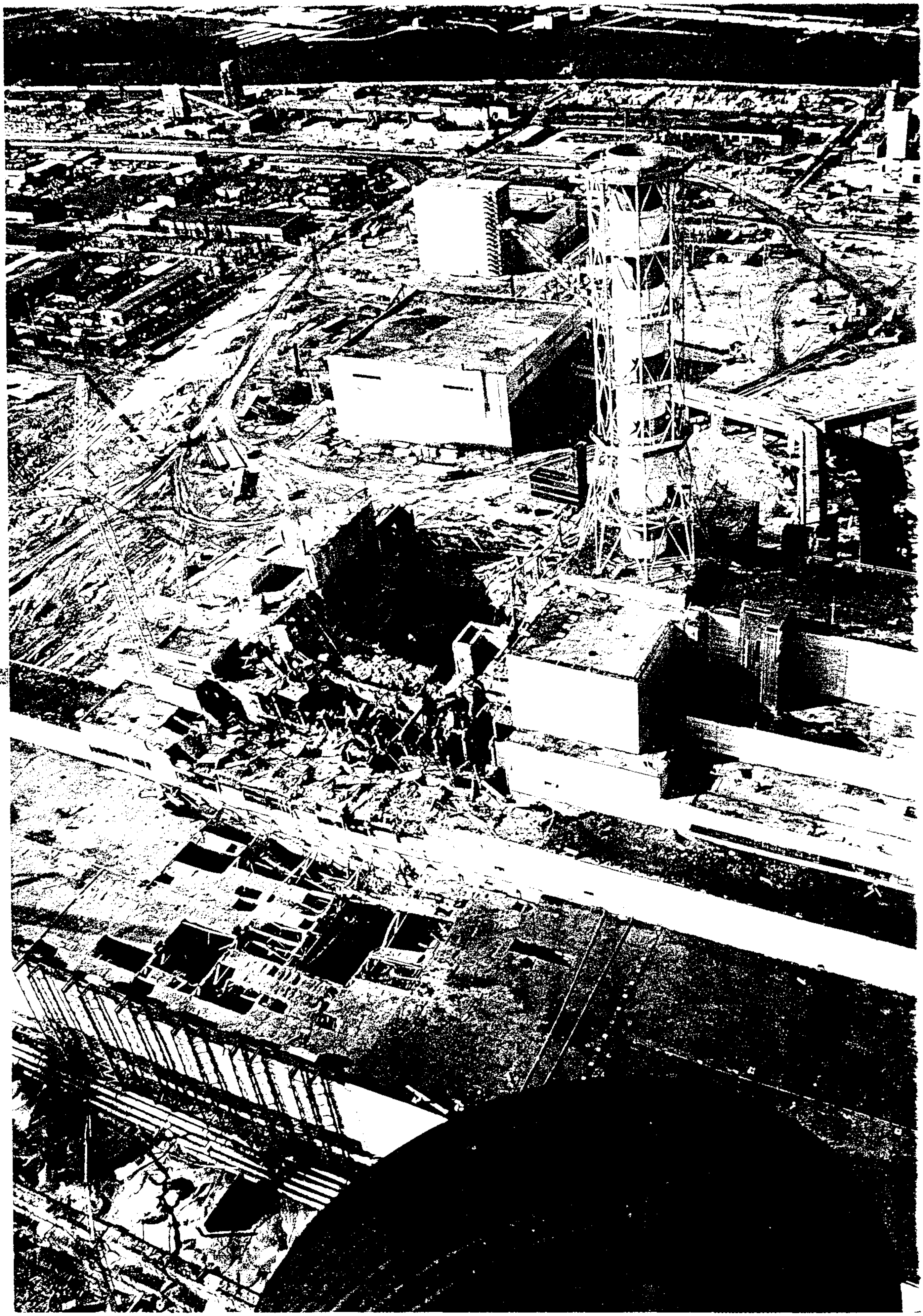
April, 26, 1986 (Saturday)

At 1.23.40 a.m. the head of the unit crew gave the order to extraordinary shutdown of the unit. But as a result of the above-mentioned number of reasons in parallel with the flaw of the control rods contributed significantly to the reactivity, the reactor power increased hundred times without any control.
It caused the abrupt increase of the temperature up to 6000^0C and pressure up to 500atm, segregation of the fuel and heat explosion, which damaged the reactor and part of the plant building involving the escape of radioactive substances to the atmosphere. Thus, according to the reports of different witness who were outside of the forth unit, since about 1.24 a.m. two explosions rang out, the burning pieces and sparks flow up upon the unit. Half of them fell down on the shelter of the turbine hall and caused the fire.

L.M.Vecsler,
V.I.Obodzinsky,
V.K.Popov

THE CHERNOBYL ACCIDENT

Nuclear Society International
Moscow, 1993



L.M.Vecsler,
V.I.Obodzinsky,
V.K.Popov

THE CHERNOBYL ACCIDENT

(accompanying text to the slides)

Nuclear Society International
Moscow, 1993

of the unit finally caused the accident. As a result of a series of violating the operating conditions by the staff of the fourth unit (only the number of the most dangerous mistakes reached six) the reactor was put into its most unstable state and its emergency protecting system intended for extraordinary terminating of the reaction had failed. As for the authors of the project, they hadn't considered the special measures for prevention of the accident having such number of deliberate violations, because such situation was accepted as unlikely. The available flaws of emergency protecting system should be mentioned also.

So, how did the events run during carrying out the test (slide N6)? According to the program the test should be carried out at the power level 700-100 MW (heat), because the continuous work at the lower level is forbidden because of arising instability in reactor's work.

In the 25th of April at 1.00 a.m. (**slide N6, position N1**) the staff began to decrease the power from the rating (3200MW) and at 1.05 p.m. (**slide N6, position N2**) its ratio reached 1600 mW. After that the turbo-generator N7 was turned off. At 2.00 p.m. according to the program of the test the emergency cooling system was turned off. Soon the controller from the "Kievenergo" sent a ban for further decreasing of the power because of the demands for electric energy, which was canceled nine hours after (**slide N6, position N3**).

As power decreased again at 0.28 a.m. on the 26th of April it was demanded to change over the regulating regime of the reactor. But, as a result of the operator's mistake, the power's level drastically decreased to 30MW (**slide N6, position N4**). In such situation the accumulation of $xenon^{135}$ with strong absorption of neutrons in the core takes place. This results in the so-called poisoning of the reactor ending the decrease of reactivity (the ability to the chain reaction). According to the instructions, in this situation the reactor should be shut down and it meant cessation of the test. The staff didn't go to it and decided to increase the power.

At 1.00 a.m. it had been gained to keep the power at the level of 200MW instead of necessary 700-1000MW (**slide N6, position N5**). As a result of removing the control rods for compensation of the poisoning the so-called efficient margin of reactivity providing the ability of the safe shutdown of the reactor became much lower than acceptable. **In other words, the reactor became badly controlled, and its acceleration ability (increasing the power out of control) became**

much more than the ability of the con
reactor. In spite of all contra-indications
the test.

According to the instructions at 1.
alternate main circulation pumps were
working ones (**slide N6, position N6**
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in order to keep the reactor operating
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to start the test. At 1.23.04 a.m. the stop
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(**slide N6, position N7**). In spite of the
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required.

As four pumps linked to the fr
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THE MEASURES TO MITIG CONSEQUE

The first major problem after the
result of explosion in the reactor (s

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According to the instructions at 1.03 a.m. and 1.07 a.m. the two alternate main circulation pumps were turned on in addition to six working ones (slide N6, position N6). After that the reactor run unstable. under these conditions the staff disabled the safety systems in order to keep the reactor operating. After a number of change-overs they gained to keep stable processes in the reactor and decided to start the test. At 1.23.04 a.m. the stop valves of the turbo-generator N8 were closed, and the steam supply of the turbine was stopped (slide N6, position N7). In spite of the instructions, the stuff blocked the emergency systems with turning off both turbines in the hope of repeating the test with turning off the turbo-generator as might be required.

As four pumps linked to the feeding bus-bar of the turbo-generator N8 began to retard revolutions, a flow of water in the reactor reduced and the boiling enhanced. In so far as the RMBK reactor has the positive steam reactivity coefficient (the more steam, the higher reactivity), the reactor power began to increase slowly since 1.23.30 a.m. (slide N6, position N8).

At 1.23.40 a.m. the head of the unit crew gave the order to extraordinary shutdown of the unit. But as a result of the above-mentioned number of reasons in parallel with the flaw of the control rods contributed significantly to the reactivity, the reactor power increased hundred times without any control (slide N6, position N9). It caused the abrupt increase of the temperature up to 6000^0C and pressure up to 500atm, segregation of the fuel and heat explosion, which damaged the reactor and part of the plant building involving the escape of radioactive substances to the atmosphere. Thus, according to the reports of different witness who were outside of the forth unit, since about 1.24 a.m. two explosions rang out, the burning pieces and sparks flow up upon the unit. Half of them fell down on the shelter of the turbine hall and caused the fire.

THE MEASURES TO MITIGATE THE ACCIDENT'S CONSEQUENCES

The first major problem after the accident was fire control. As a result of explosion in the reactor (slide N7) and emission of the

KFA

KERNFORSCHUNGSANLAGE JÜLICH GmbH

Sicherheitstechnische Grundlagen für die Katastrophenschutzplanung am THTR-300

Dieser Bericht gibt die Ergebnisse einer im Auftrag des Ministers für Arbeit, Gesundheit und Soziales des Landes Nordrhein-Westfalen vom Institut für Nukleare Sicherheitsforschung der Kernforschungsanlage Jülich erstellten Studie „Ermittlung von Strahlendosen in der Umgebung des THTR-300 infolge eines angenommenen Coreaufheizunfalles“ in allgemein verständlicher Form wieder.

Sicherheitstechnische Grundlagen
für die Katastrophenschutzplanung
am THTR-300

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Redaktion:	Dipl.-Ing. Siegfried A. Weinhold (verantwortlich) Telefon 0 24 61/61 46 61, Telex 8 3 355 670
Grafik und Einbandgestaltung:	Grafik-Design Hans-W. Schiffer, Jülich
Satz und Lithos:	Robert Rögels, Filmsatztechnik, Mönchengladbach Inh. Rainer Prisack
Druck und Herstellung: November 1984	Grafische Betriebe der KFA Jülich

Sicherheitstechnische Grundlagen für die Katastrophenschutzplanung am THTR-300

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Institut für Nukleare Sicherheitsforschung
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Schwere Reaktorunfälle, bei denen Leben, Gesundheit und Eigentum der Mitbürger ernsthaft in Mitleidenschaft gezogen werden, sind als so extrem unwahrscheinlich anzusehen, daß sie nach den Maßstäben der praktischen Vernunft auszuschließen sind. Wegen der vielfachen Absicherung aller sicherheitstechnisch wichtigen Funktionen des Reaktors müßten so viele Komponenten und auch die Betriebsmannschaft unabhängig voneinander versagen, daß mit derartigen Unfällen allenfalls einmal in einer Million bis zehn Millionen Betriebsjahren zu rechnen ist. Die Betriebserfahrungen mit mehr als 300 Kernkraftwerken in aller Welt, die bislang zusammen über 3 000 Betriebsjahre erbracht haben, bestätigen diese Auffassung. Bislang ist noch kein Kernkraftwerksunfall aufgetreten, der nachweisbare gesundheitliche Folgen für die umwohnende Bevölkerung verursacht hätte. Dies gilt auch für den Unfall in Harrisburg im Jahre 1979. Der eigentliche Reaktor wurde zwar schwer beschädigt, die Bevölkerung war aber keinen nennenswerten Strahlendosen ausgesetzt.

Trotzdem ist für Kernkraftwerke — wie für alle großtechnischen Anlagen — eine Katastrophenschutzplanung vorzusehen, die Vorsorge auch für den extrem unwahrscheinlichen Fall treffen soll. Die zu planenden Schutzmaßnahmen und die Abgrenzung der Zonen, in denen sie vorzusehen sind, richten sich nach den „Rahmenempfehlungen für den Katastrophenschutz in der Umgebung kerntechnischer Anlagen"*) (Tab. 1), die der Bundesminister des Innern veröffentlicht hat. Diese Empfehlungen berücksichtigen allerdings nicht die unterschiedlichen Leistungsgrößen der verschiedenen Kernkraftwerke; d.h., die darin empfohlenen maximalen Radien für die einzelnen Schutzzonen sind so bemessen, daß sie auch für moderne Kernkraftwerke mit Leichtwasserreaktoren bis zu Leistungen von 1 300 MW_{el} sicher ausreichen. Für den mit 300 MW_{el} wesentlich leistungsschwächeren THTR-300, der außerdem gegenüber den Leichtwasserreaktoren andere sicherheitstechnische Eigenschaften besitzt, ist es nicht erforderlich, ohne nähere Prüfung die empfohlenen Maximalwerte der Planung zugrunde zu legen. Deshalb wurde versucht, die bei einem Unfall zu erwartende Strahlenbelastung der Bevölkerung in der Umgebung des Kernkraftwerkes zu ermitteln, um die zu planenden Notfallmaßnahmen daran ausrichten zu können.

Die Ergebnisse der Untersuchungen, die im Auftrag des Ministers für Arbeit, Gesundheit und Soziales vom Institut für Nukleare Sicherheitsforschung der Kernforschungsanlage Jülich ausgeführt wurden, sind in dem Bericht Jül-Spez-257 „Ermittlung von Strahlendosen in der Umgebung des THTR-300 infolge eines angenommenen Coreaufheizunfalls" ausführlich dargestellt. Der vorliegende Artikel gibt eine allgemeinverständliche verkürzte Fassung der wichtigsten Resultate wieder. Einzelheiten können in dem oben zitierten Jül-Spez-Bericht nachgelesen werden.

*) im folgenden als „Rahmenempfehlungen" zitiert.

Der THTR-300

Der THTR-300 gehört zu den Hochtemperaturreaktoren (HTR), die — anders als Leichtwasserreaktoren (LWR) —

Tabelle 1: Vom Bundesminister des Innern empfohlene Dosis-Richtwerte für das Einleiten akuter Notfallmaßnahmen (aus: „Rahmenempfehlungen")

Empfohlene Dosis-Richtwerte für das Einleiten akuter Notfallmaßnahmen bei störfallbedingter Bestrahlung

Ganzkörperbestrahlung von außen und durch Inhalation

Gefährdungsklasse	Ganzkörperdosis bei Aufenthalt im Freien[1]	Empfohlene Notfallmaßnahmen	
		Verbleiben im Haus	Räumung
I	bis 25 rem	zweckmäßig	nein
II	25–100 rem	erforderlich	zweckmäßig
III	über 100 rem	erforderlich bis zur Räumung	erforderlich

[1] Die angegebenen Dosiswerte resultieren aus einer Exposition **im Freien** bei einer Aufenthaltsdauer von einigen Stunden bis zu einigen Tagen unabhängig von Alter oder Geschlecht. Evtl. geringere Wirkungen bei Teilabschirmung des Körpers oder längerer zeitlicher Verteilung der Strahlenbelastung sind außer Acht gelassen.

Bestrahlung der Schilddrüse durch Inhalation von Radiojod und Radiotellur

Gefährdungsklasse	Schilddrüsendosis nach Aufenthalt im Freien[1]	Empfohlene Notfallmaßnahmen		
		Verbleiben im Haus	Jodidtabletten	Räumung
I	bis 25 rem	zweckmäßig	entbehrlich	nein
II	25–500 rem	erforderlich	zweckmäßig bei 100 rem; bei über 100 rem erforderl.	entbehrlich
III	über 500 rem	erforderlich bis zur Räumung	erforderlich auch bei Räumung	zweckmäßig bis 1 000 rem; bei über 1 000 rem erforderlich

Wärme auf einem hohen Temperaturniveau erzeugen können. Dies wird durch eine Reihe technischer Besonderheiten erreicht, die den HTR grundsätzlich vom LWR unterscheiden.

Als *Spaltstoff* dient beim THTR-300 ein Gemisch von Uran- und Thoriumdioxid in Form mohnkorngroßer Partikel. Jede Partikel ist von mehreren Graphit-Schichten umhüllt, die wie winzige Druckkessel alle bei der Kernspaltung entstehenden Spaltprodukte zurückhalten. Mehr als 35 000 solcher Partikel werden, mit Graphitpulver vermengt, zu einer Kugel verpreßt, die mit einer brennstofffreien Graphitschicht umgeben und bei hohen Temperaturen gesintert wird. Die fertige, tennisballgroße Kugel stellt dann das eigentliche *Brennelement* des THTR-300 dar (Bild 1).

Über 650 000 Brennelementkugeln, die von einem aus Graphitblöcken aufgebauten Reaktorgefäß aufgenommen werden, bilden den *Reaktorkern* des Kugelhaufen-Reaktors THTR-300 (Bild 2).

Zum Abtransport der bei der Kernspaltung entstehenden Wärme durchströmt den Reaktorkern das Kühlgas Helium, das von Gebläsen umgewälzt wird. In Dampferzeugern wird seine Wärme zur Erzeugung hochgespannten, heißen Dampfes genutzt, der schließlich die Turbine und den Generator zur Elektrizitätserzeugung treibt.

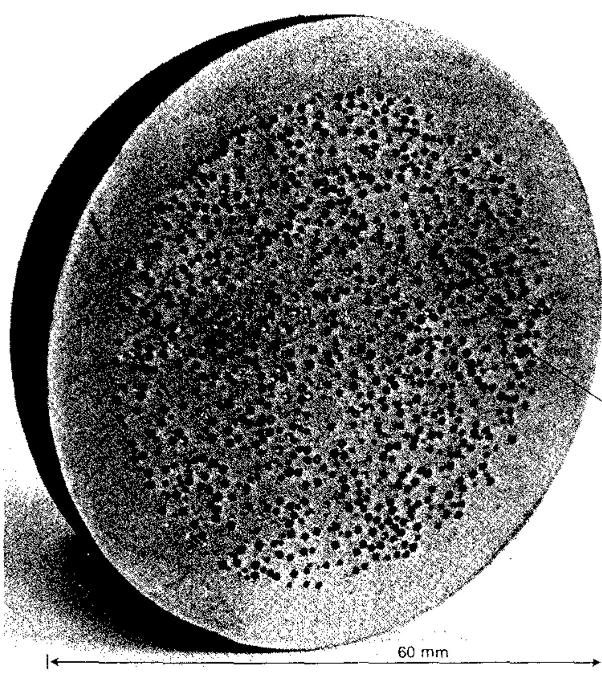
Reaktorkern, Gebläse und Dampferzeuger bilden den *Primärkreislauf*, der von einer gasdichten Stahlhaut, dem sog. Liner, umschlossen wird. Der relativ dünnwandige Liner überträgt den Druck des Kühlgases auf einen dickwandigen Spannbetonbehälter, der gleichzeitig die Strahlung des Reaktorkerns auf einen ungefährlichen Wert reduziert.

Durch den Aufbau des Kerns aus hochschmelzenden keramischen Materialien besitzt der THTR-300 eine hohe „inhärente" *Sicherheit*. Darunter sind gleichsam angeborene Sicherheitseigenschaften zu verstehen, die auch ohne Auslösung aktiver Sicherheitseinrichtungen ständig wirksam sind. Ein Schmelzen des Reaktorkerns ist schon deshalb ausgeschlossen, weil die dazu notwendigen Temperaturen nicht erreicht werden können. Der bis zu sehr hohen Temperaturen wirksame Einschluß der Spaltprodukte in den Brennstoffpartikeln hält das Kühlgas sauber, so daß bei Undichtigkeiten entweichendes Helium nur geringfügig radioaktiv verunreinigt ist und keine Gefahr für die Umgebung darstellt. Die enorme Graphitmasse des Reaktorkerns, die gewaltige Wärmemengen speichern kann, verlangsamt alle Temperaturwechselvorgänge, so daß bei Störungen reichlich Zeit für Gegenmaßnahmen in der Anlage und — falls erforderlich — auch in der Umgebung bleibt. Das daraus resultierende ausgesprochen „gutmütige" Verhalten gasgekühlter Reaktoren bei Betriebsstörungen ist durch langjährige Betriebserfahrungen mit Versuchs- und Prototyp-Reaktoren voll bestätigt worden.

Angenommener Unfallablauf

Betriebsstörungen in Kernkraftwerken können je nach Ursache und zufälligem Versagen von Sicherheitseinrichtungen und Personal einen sehr unterschiedlichen Verlauf nehmen. In der Regel werden sie durch das Reaktorsicherheitssystem und durch geeignete Maßnahmen der Operateure aufgefangen, ohne daß es zu irgendwelchen unerwünschten Folgen käme. Wenn jedoch in unwahrscheinlichen Ausnahmefällen mehrere Sicherheitseinrichtungen und die Operateure versagen sollten, kann sich eine Störung zu einem Störfall (ohne Personenschäden) oder zu einem Unfall (möglicherweise mit Personenschäden) entwickeln. Sofern mit dem Auftreten eines bestimmten Störfalls öfter als etwa einmal in 10 000 bis 100 000 Jahren gerechnet werden muß, ist die Anlage dagegen auszulegen, d.h. im Genehmigungsverfahren ist nachzuweisen, daß die Auswirkungen solcher Störfälle auf die Umgebung unter vorgeschriebenen, sehr niedrigen Grenzwerten bleiben, bei denen Personenschäden sicher vermieden werden. Zu den Störfällen, gegen die beim THTR-300 Vorsorge getroffen ist, gehören auch Erdbeben, Flugzeugabsturz und Brand. Bei noch selteneren Vorfällen wird gefordert, daß das dadurch verursachte Risiko so gering wie möglich gehalten wird: Personenschä-

Bild 1: Das Brennelement des THTR-300 besteht aus einer inneren Graphitkugel von 5 cm Durchmesser, die rund 35 000 Brennstoffpartikel enthält, und einer damit unlösbar verbundenen Graphitaußenschale. Die etwa 1 mm großen Brennstoffpartikel enthalten einen Kern von Uran- und Thoriumdioxid, umgeben von mehreren Graphitschichten, die die Spaltprodukte sicher einschließen.



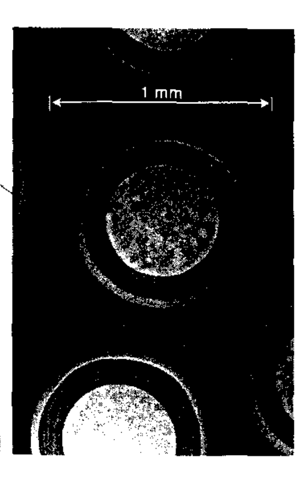
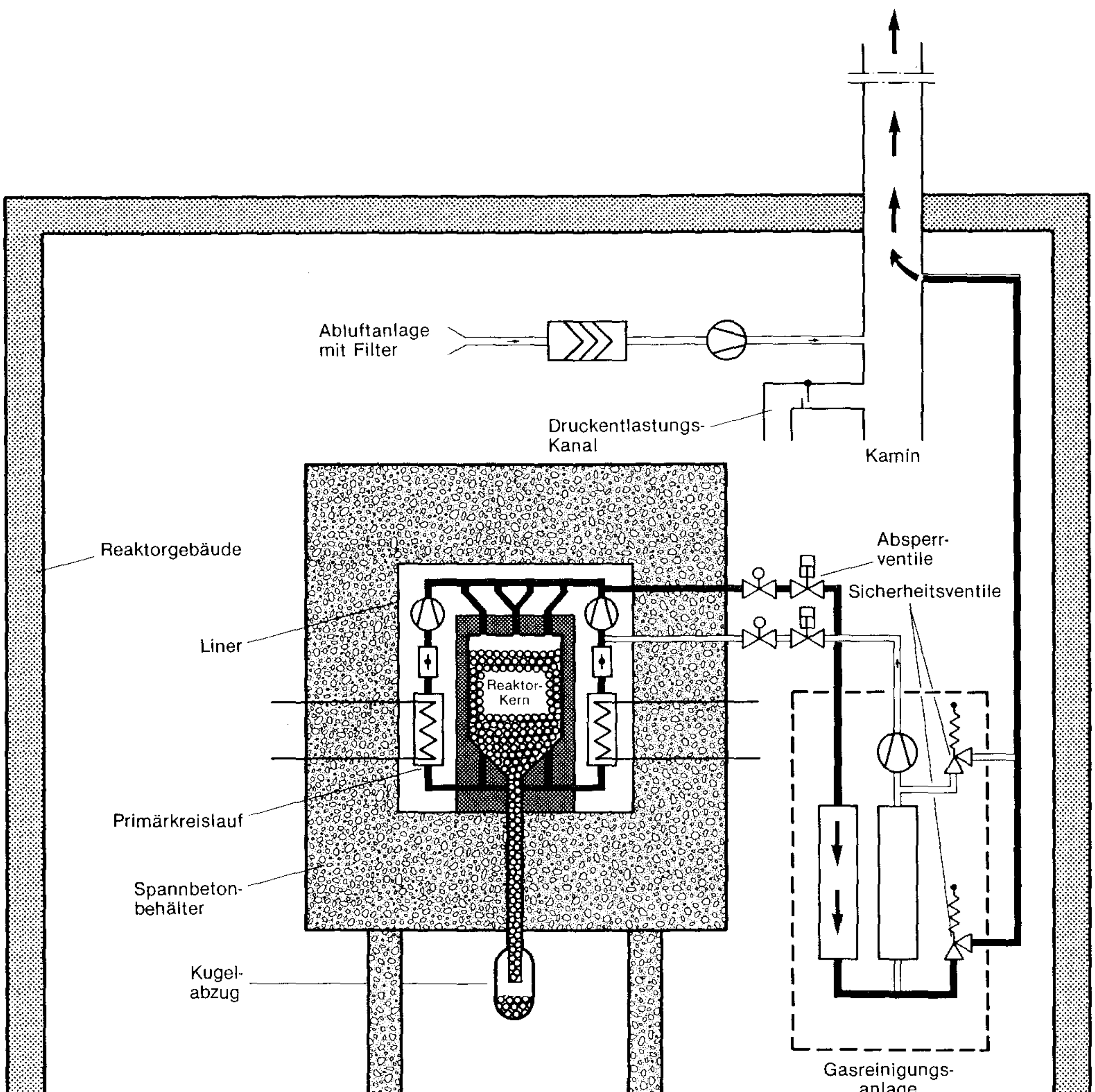


Bild 2: Schematischer Querschnitt durch den THTR-300. Das Bild enthält nur die für den Ablauf des Unfalls wichtigen Komponenten.

▬ Freisetzungspfad bei dem angenommenen Kernaufheizunfall.



den sind jedoch nicht völlig auszuschließen.

Die Gefährdung der Umgebung geht beim THTR-300 wie bei allen Kernkraftwerken von den Spaltprodukten aus, die bei der Kernspaltung unvermeidlich entstehen und im Reaktorkern gespeichert werden. Beim THTR-300 sind sie im Normalbetrieb durch mehrere keramische und metallische Umhüllungen sicher eingeschlossen, die alle versagen müßten, damit Spaltprodukte in die Umgebung gelangen könnten. Eine denkbare Ursache für die Freisetzung von Spaltprodukten aus dem Reaktorkern wäre der totale und langandauernde Ausfall der Reaktorkühlung, der eine Überhitzung des Reaktorkerns zur Folge hätte. Je nach Verlauf könnte es dann zu einem *Kernaufheizstörfall* oder *-unfall* kommen.

Fällt die normale Kühlung des Reaktorkerns aus, so schaltet sich der Reaktor automatisch ab. Dadurch wird die Kernspaltung im Reaktorkern und die damit verbundene Wärmeproduktion sofort unterbrochen. Weil aber die Spaltprodukte durch ihren radioaktiven Zerfall nach wie vor Wärme erzeugen, geht die Wärmeproduktion des Reaktors nicht sofort auf Null zurück, sondern zunächst nur auf einige Prozent der Nennleistung; sie nimmt dann im Verlauf von Tagen und Wochen immer weiter ab. Bei einem Ausfall der betrieblichen Kühlung wird deshalb automatisch eine Notkühlung eingeschaltet, die imstande ist, diese Zerfallswärme abzuführen. Sollte auch die Notkühlung vollständig versagen, so heizt sich der Reaktorkern durch die Zerfallswärme auf. Die Aufheizung wird dadurch begrenzt, daß einerseits infolge der steigenden Temperaturen immer mehr Wärme zur Reaktoroberfläche abfließt, andererseits aber die Wärmeproduktion ständig zurückgeht. Schließlich überwiegt nach einigen Tagen die Wärmeableitung die Wärmeproduktion, und die Temperaturen sinken wieder.

Der Reaktorkern des THTR-300 kann einen solchen Temperaturüberschlag ohne mechanische Schäden überstehen, weil er ganz aus keramischen Materialien — Uran- bzw. Thoriumdioxid und Graphit — aufgebaut ist, die ihre Festigkeit erst bei sehr hohen Temperaturen — 2 800 bzw. 3 600 °C — verlieren. Die gewaltige Masse des Graphits, die hochgeheizt werden muß, sorgt außerdem dafür, daß alle Vorgänge sehr langsam ablaufen, so daß viel Zeit für Gegenmaßnahmen verbleibt. Trotzdem kann es unter Umständen zu einer partiellen Freisetzung von Radioaktivität kommen — etwa ein Promille der im Reaktorkern gespeicherten Spaltprodukte —, die zu einer Umgebungsbelastung führt. Ursache ist einmal, daß die keramische Umhüllung des Spaltstoffs mit steigender Temperatur beginnt, für bestimmte Spaltprodukte (Cäsium, Strontium) durchlässig zu werden. Im Innern des Reaktorkerns erreichen die Temperaturen vorübergehend Werte, bei denen dieser Prozeß schon Bedeutung erlangt. Die Freisetzung hört auf, sobald sich der Reaktorkern wieder abkühlt. Bei noch höheren Temperaturen, die erst nach einigen Tagen in engbegrenzten Bereichen des Reaktorkerns auftreten können, bricht außerdem ein bestimmter Prozentsatz der keramischen Umhüllungen der Brennstoffpartikel, so daß auch noch andere Spaltprodukte (vornehmlich Jod) freigesetzt werden können.

Die aus dem Brennstoff austretenden Spaltprodukte verbleiben zunächst im geschlossenen Primärkreislauf des Reaktors; zum größten Teil lagern sie sich in den kälteren Bereichen des Reaktorkerns und des Primärkreislaufs wieder ab. Zu einem nennenswerten Austrag von Spaltprodukten kommt es nur dann, wenn durch ein kleines Leck im Primärkreislauf ständig ein Gasstrom entweicht, der die freigesetzten Spaltprodukte aus dem Reaktor mitführt. Für die Berechnung der Unfallfolgen wurde deshalb unterstellt, daß der Primärkreislauf des Reaktors zunächst intakt bleibt. Mit steigender Temperatur steigt auch der Druck im Primärkreislauf, bis schließlich nach etwa acht Stunden der Ansprechdruck eines Sicherheitsventils erreicht wird. Es wird angenommen, daß dieses Ventil versagt und nicht wieder schließt, so daß ein ständiger Kühlgasstrom die aus dem Reaktorkern freigesetzten Spaltprodukte in die Umgebung transportiert, wobei der größte Teil wiederum in den kälteren Bereichen des Reaktors zurückgehalten wird. Nach etwa zwei Tagen ist der Kühlgasdruck auf Atmosphärendruck abgesunken; die Spaltproduktemission ist damit praktisch beendet.

Ein derartiger Unfall ist sehr unwahrscheinlich, weil viele Sicherheitseinrichtungen und auch das Betriebspersonal versagen müßten, damit es überhaupt dazu kommt. Ausfallen müßten z. B. das betriebliche Kühlsystem und die Notkühlsysteme, oder die externen Stromversorgungen und die Notstromdiesel, zusätzlich außerdem die Einrichtungen zum Absenken des Kühlgasdrucks; das Sicherheitsventil müßte in Offenstellung versagen. Dabei muß auch noch unterstellt werden, daß Reparaturversuche trotz der zur Verfügung stehenden Zeit von mehreren Stunden erfolglos bleiben.

Es ist deshalb sehr viel wahrscheinlicher, daß die an und für sich schon sehr unwahrscheinlichen Kernaufheizstörfälle einen anderen und milderen Verlauf nehmen. Zum Beispiel kann bei einem Rohrbruch im Primärkreislauf das Kühlgas erheblich schneller entweichen als in dem oben untersuchten Unfall. Die Auswirkungen solcher Kernaufheizstörfälle liegen aber unter denen des oben beschriebenen Unfalls, weil das Kühlgas schon ausgeströmt ist, bevor eine stärkere Freisetzung von Spaltprodukten aus dem Reaktorkern in Gang kommt.

Die Auswirkungen eines Unfalls auf die Umgebung des Kernkraftwerkes hängen in hohem Maße davon ab, wie sich die freigesetzten Spaltprodukte ausbreiten. Dies wird im wesentlichen durch die Witterungsbedingungen bestimmt, die zur Zeit des Unfalls gerade herrschen. Regen z. B. wäscht einen Teil der Spaltprodukte aus der Abgasfahne aus, so daß lokal höhere Strahlendosen auftreten können, als wenn bei stürmischem Wetter die freigesetzten Spaltprodukte über eine größere Fläche verteilt werden. Für die Rechnung werden sehr ungünstige Wetterverhältnisse angenommen, die nur selten — weniger als 20 Stunden im Jahr — auftreten.

Die Direktstrahlung des unfallgeschädigten Reaktors spielt nur in der allernächsten Umgebung des Reaktors innerhalb des Kernkraftwerksgeländes eine Rolle. Die mehrere Meter dicken Wände des Spannbetonbehälters bieten einen ausgezeichneten Schutz vor der Strahlung des Reaktorkerns: ihre Zerstörung durch einen Reaktorunfall — auch durch Erdbeben oder Flugzeugabsturz — ist ausgeschlossen.

Einwirkung radioaktiver Stoffe auf den Menschen

Radioaktive Stoffe — dazu zählen auch die Spaltprodukte — können in vielfacher Weise auf den Menschen einwirken, wenn sie durch einen Unfall aus einem Kernkraftwerk freigesetzt werden. Dies hängt einmal mit der unterschiedlichen Natur der Strahlung zusammen, die von radioaktiven Stoffen ausgesandt wird. *α-Strahlen* haben nur eine sehr kurze Reichweite; ein Blatt Papier hält sie schon vollständig zurück. Sie entfalten ihre schädigende Wirkung nur dann, wenn die Substanzen, die α-Strahlen aussenden, in den Körper gelangen. *β-Strahlen* (identisch mit schnellen Elektronen) durchdringen je nach Energie einige Millimeter bis mehrere Zentimeter menschliches Gewebe. Sie wirken sowohl von außen als auch von innen auf den Körper ein. *γ-Strahlen* schließlich (identisch mit „harter" Röntgenstrahlung) vermögen auch dicke Materieschichten zu durchdringen. Ihre Reichweite in Luft kann mehrere hundert Meter betragen.

Die meisten Spaltprodukte, die bei dem unterstellten Reaktorunfall des THTR-300 freigesetzt werden könnten, emittieren *β-* und *γ-Strahlen*, während *α-Strahlen* in diesem Zusammenhang keine Rolle spielen.

Die chemische Natur der Spaltprodukte hat für ihre biologische Wirksamkeit

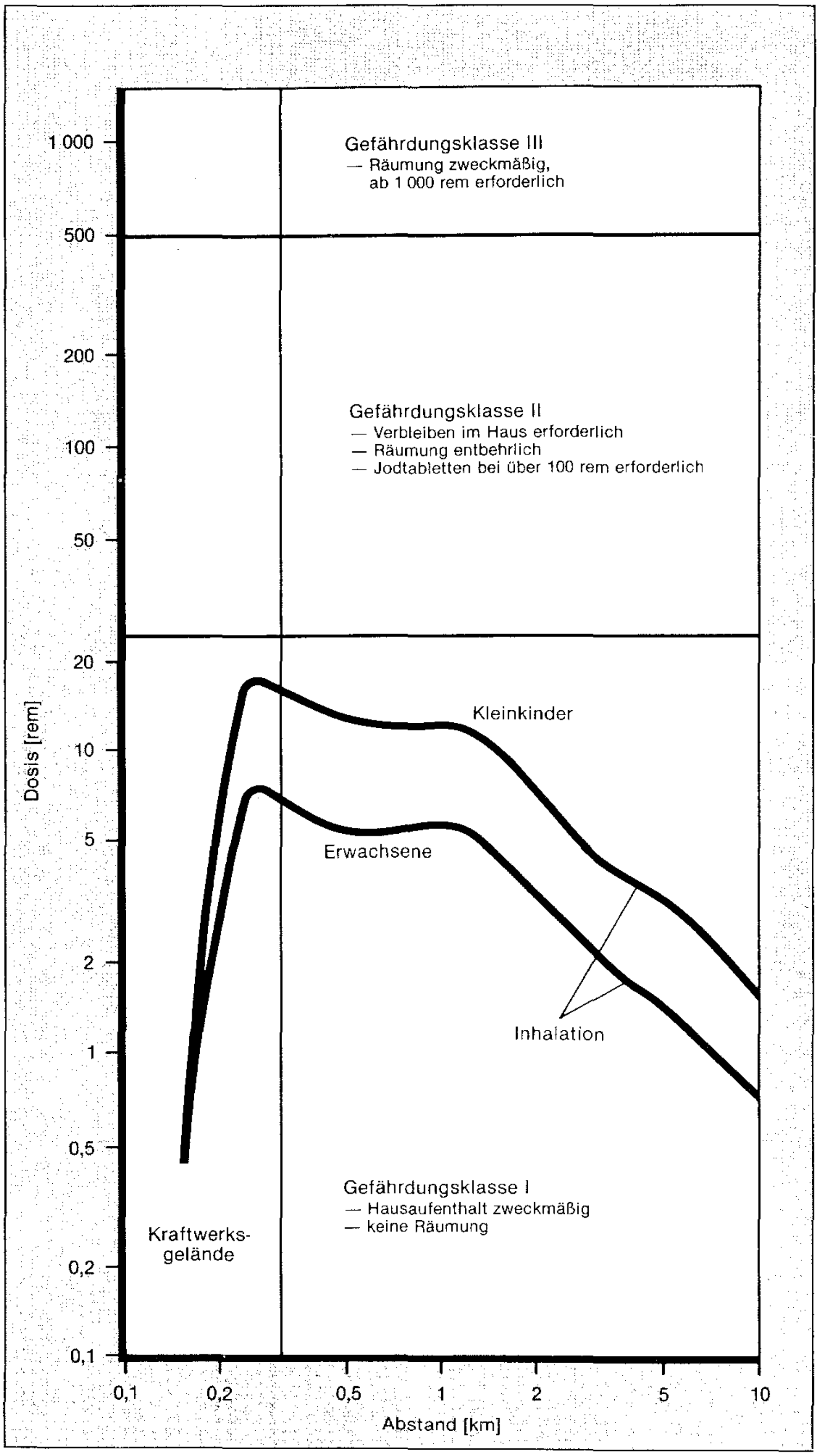


Bild 3: Schilddrüsendosis, verursacht durch Inhalation von Jod und Tellur bei angenommenem Kernaufheizunfall des THTR-300. Verzehrverbot spätestens acht Stunden nach Unfallbeginn. (rem: Maßeinheit für die relative biologische Wirkungsdosis, gültig für alle Strahlenarten)

ebenfalls große Bedeutung. Radioaktive *Edelgase* (Xenon, Krypton) gehen keine chemischen Verbindungen ein und lagern sich auch nicht ab. Sie wirken hauptsächlich durch die weitreichende *γ-Strahlung* aus der Gasfahne, die nach einem Reaktorunfall aus dem Kamin entweicht. Der betroffene Mensch ist gleichsam in diese radioaktive Wolke eingetaucht; die dadurch bewirkte Strahlendosis wird deshalb auch als *Submersionsdosis* bezeichnet. Aus den vorhin genannten Gründen tragen aber die Edelgase bei Reaktorunfällen nur wenig zur gesamten Unfalldosis bei.

Von den *flüchtigen* (d.h. bei höheren Temperaturen gasförmigen) *Spaltprodukten* kommt dem Jod herausragende Bedeutung zu, weil es in der Schilddrüse gespeichert wird. Es kann entweder mit der Nahrung, vor allem der Milch, aufgenommen oder auch eingeatmet werden. Die durch eingeatmete Spaltprodukte verursachte Dosis heißt *Inhalationsdosis.*

Die *metallischen Spaltprodukte* (Cäsium, Strontium) treten meist als Aerosole (luftgetragene staubförmige Partikel) in Erscheinung. Sie werden von der Gasfahne ein Stück weit mitgeführt und sinken dann je nach Witterungsverhältnissen früher oder später zu Boden. Dort verursachen sie die sog. *Bodenstrahlung*, die meist der Submersionsdosis zugeschlagen wird, weil sie sehr ähnlich wirkt. Wegen der Langlebigkeit einiger metallischer Spaltprodukte kann die Bodenstrahlung unter Umständen sehr lange zur äußeren Strahlenbelastung beitragen (Langzeitdosis).

Auf dem Boden abgelagerte Spaltprodukte können auch von den Pflanzen aufgenommen und in das pflanzliche Gewebe eingebaut werden. Handelt es sich um Nahrungsmittel (z.B. Gemüse), so gelangen sie durch Verzehr in das Körperinnere und werden über den Ver-

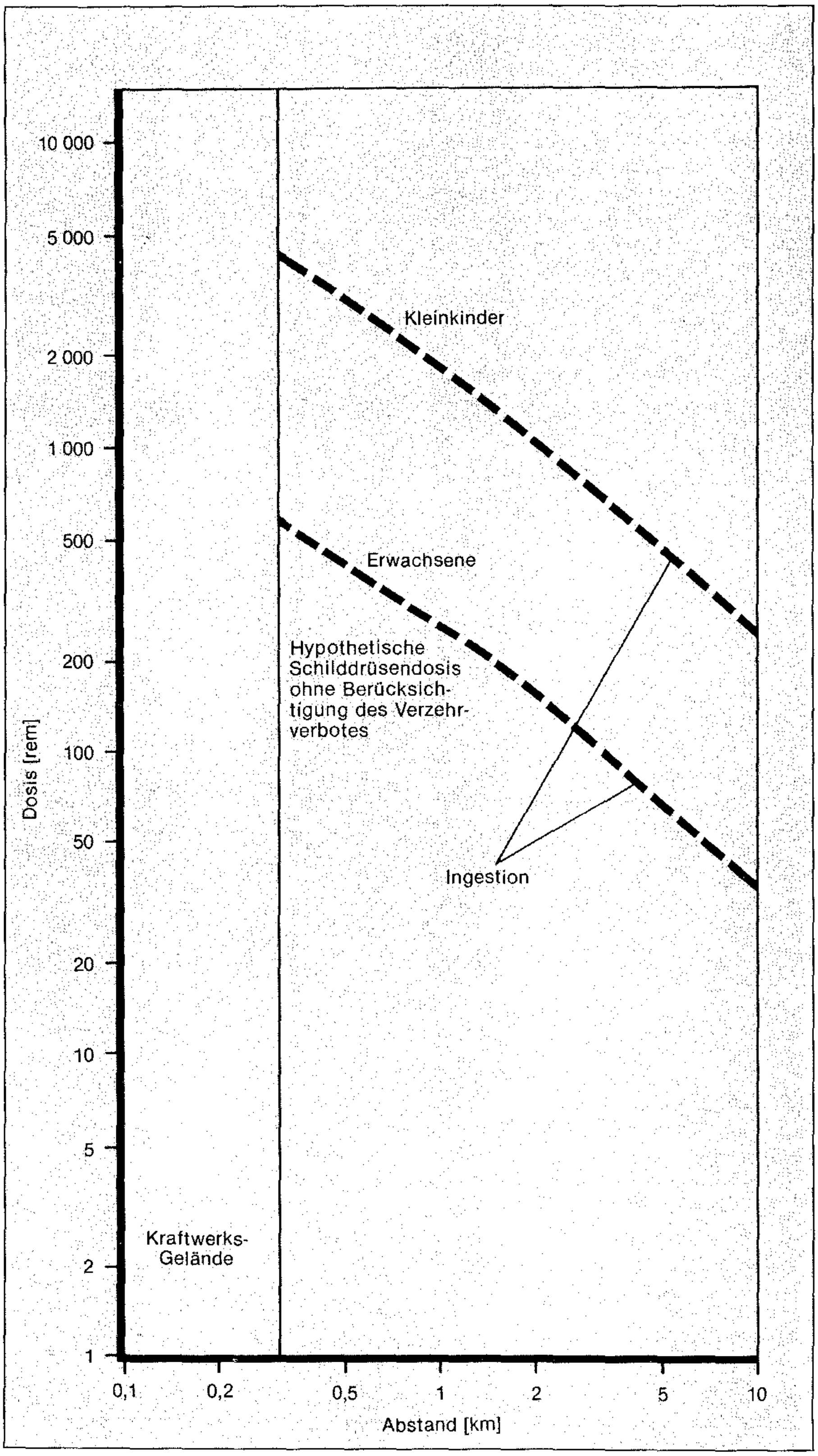
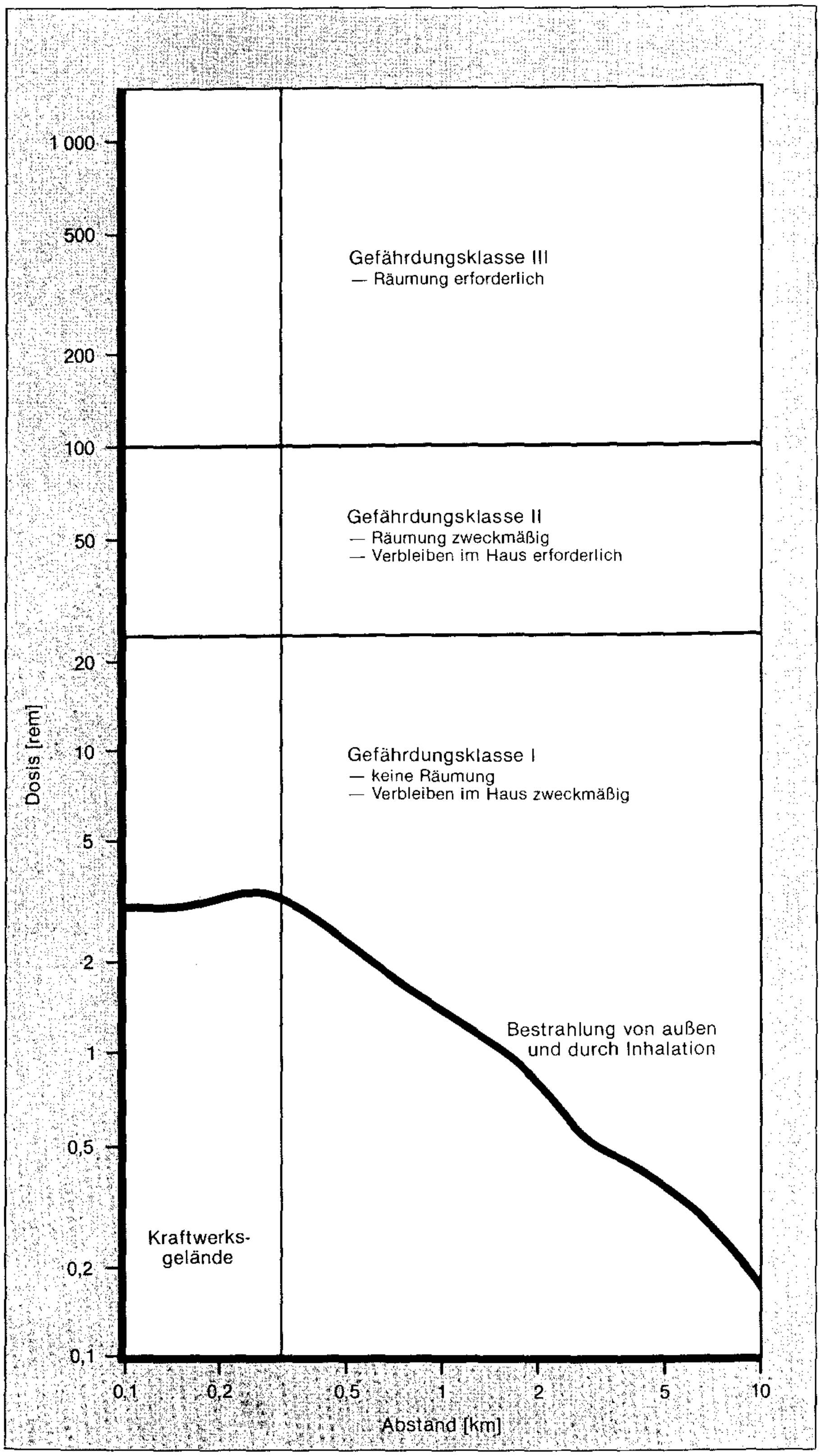


Bild 4: Hypothetische Schilddrüsendosis, verursacht durch den Verzehr kontaminierter Nahrungsmittel innerhalb der ersten 24 Stunden nach Beginn der Jodfreisetzung bei dem angenommenen Kernaufheizunfall des THTR-300, wenn ein Verzehrverbot nicht schon 8 Stunden, sondern erst 32 Stunden nach Unfallbeginn ausgesprochen würde.



dauungsprozeß teils ins Gewebe, teils in die Knochen eingebaut. Die durch die innere Bestrahlung verursachte *Ingestionsdosis* ist insofern besonders wichtig, weil man sie nicht mehr durch Ortswechsel und nur in begrenztem Maße durch andere Maßnahmen beeinflussen kann, sobald die Spaltprodukte vom Körper aufgenommen („inkorporiert") worden sind. Dies gilt auch für die durch Einatmung aufgenommenen Aerosolpartikel, die zur Inhalationsdosis beitragen.

Je nach Unfallverlauf und Witterung tragen die verschiedenen *Belastungspfade* — Submersion, Inhalation und Ingestion — in unterschiedlicher Weise zur Strahlenbelastung bei. Die Schutzmaßnahmen — Verbleib in geschlossenen Gebäuden, Verzehrverbot, Evakuierung — müssen sich nach der relativen Bedeutung der einzelnen Belastungspfade richten.

Die Höhe der Strahlendosen, die die betroffenen Anwohner empfangen, hängt natürlich auch von der *Einwirkungszeit* ab. Im vorliegenden Fall wurde für die Berechnung der Strahlendosen angenommen, daß die Anwohner sieben Tage lang ungeschützt der Strahlung von außen (Luft, Boden) ausgesetzt sind. Die durch Inhalation und Ingestion verursachte innere Bestrahlung wurde hingegen über die Verweilzeit der inkorporierten radioaktiven Substanzen im Körper, maximal über einen Zeitraum von 50 Jahren, berücksichtigt, weil inkorporierte Radioaktivität nur schwer wieder aus dem Körper zu entfernen ist.

Zur Beurteilung der *gesundheitlichen Folgen* einer Strahlenbelastung werden gewöhnlich die Dosen herangezogen, denen der gesamte Körper und einige radiologisch besonders wichtige Organe ausgesetzt waren, unabhängig davon, auf welchen Belastungspfad sie zurückzuführen sind. Neben der *Ganzkörperdosis* interessiert vor allem die *Schilddrüsendosis*. Für diese beiden Dosen geben die „Rahmenempfehlungen" Richtwerte an, die eine Festlegung von

Bild 5: Ganzkörperdosis bei dem angenommenen Kernaufheizunfall des THTR-300. (Hauptbeitrag verursacht durch Inhalation von Strontium)

drei Gefährdungsklassen ermöglichen (Tab. 1). Für jede Gefährdungsklasse werden bestimmte Notfallmaßnahmen empfohlen, so daß die Berechnung der zu erwartenden Dosen eine sinnvolle Planung der Notfallmaßnahmen erlaubt (vgl. Ausführungen Seite 1).

Auswirkungen des angenommenen Reaktorunfalls THTR-300 auf die Umgebung

Bei dem angenommenen Reaktorunfall werden während der ersten acht Stunden nach Unfallbeginn keinerlei Spaltprodukte in die Umgebung emittiert, weil erst dann der Primärkreislauf durch Ansprechen eines Sicherheitsventils geöffnet wird. Zunächst werden hauptsächlich Edelgase und etwas Jod freigesetzt, die aus schadhaften Brennstoffpartikeln stammen. Erst nach mehr als 30 Stunden ist die Temperatur im Reaktorkern so weit gestiegen, daß auch Cäsium emittiert wird. Strontium erscheint erst nach über 50 Stunden, kurz bevor der Emissionsvorgang durch die einsetzende Abkühlung des Reaktorkerns praktisch beendet wird.

Die *Strahlendosen*, die diese Emissionen radioaktiver Stoffe verursachen würden, sind unter sinngemäßer Anwendung der Rechenvorschriften für die Berechnung der Individualdosen nach Auslegungsstörfällen ermittelt worden. Die „Rahmenempfehlungen" berücksichtigen bei der Festlegung der Notfallmaßnahmen nur die Ganzkörperbestrahlung von außen und durch Inhalation sowie die durch Inhalation verursachte Schilddrüsendosis. Dieses Vorgehen entspricht der Zielsetzung der Notfallmaßnahmen, nämlich der Abwendung unmittelbarer Gefahren nach einem Reaktorunfall. Um festzustellen, wie weit im vorliegenden Fall auch Maßnahmen zur Verhinderung des Verzehrs radioaktiv kontaminierter Nahrungsmittel für den Notfallschutz von Bedeutung sein könnten, wurden auch

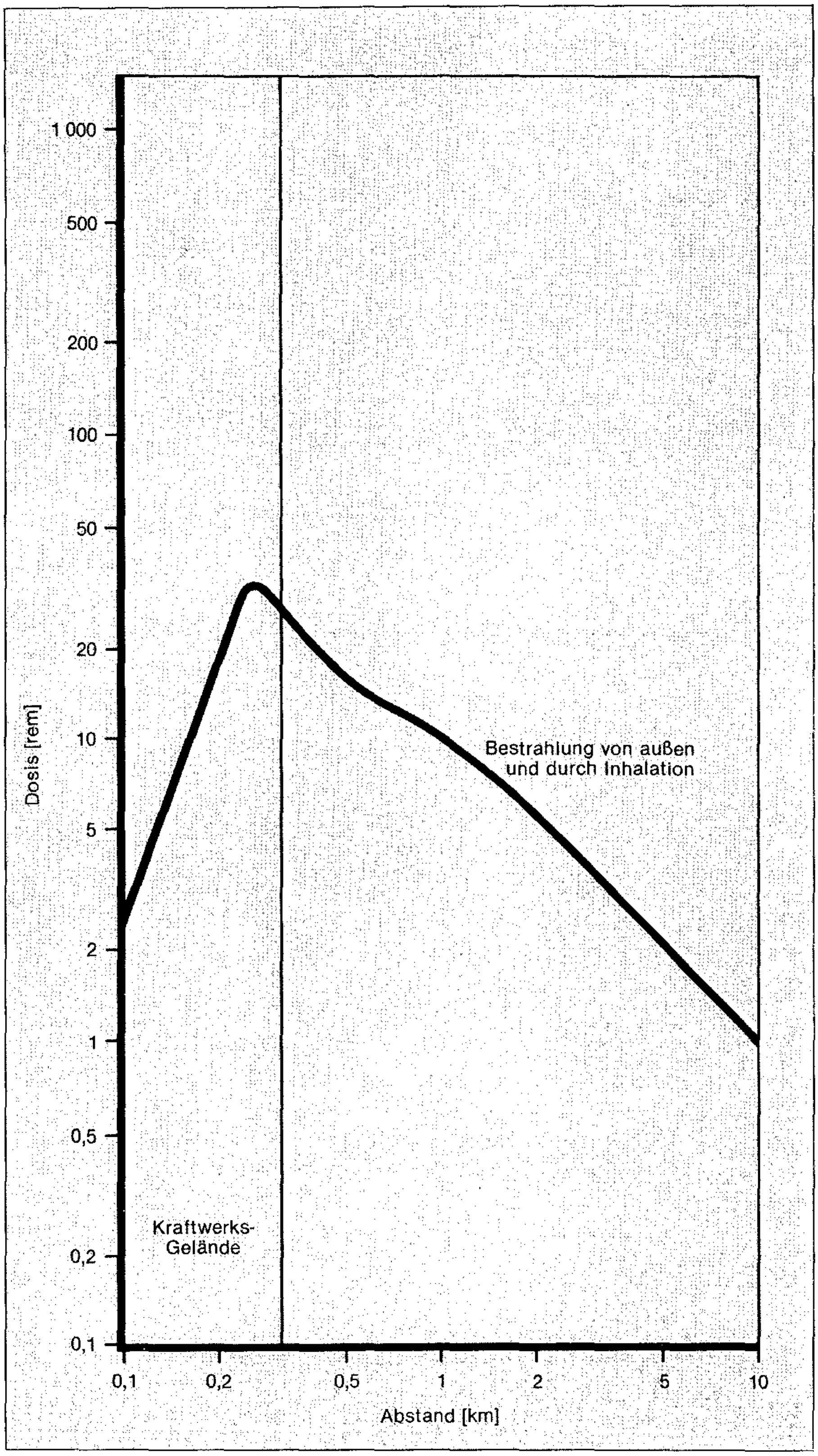


Bild 6: Knochendosis bei dem angenommenen Kernaufheizunfall des THTR-300. (Hauptbeitrag verursacht durch Inhalation von Strontium)

die über den Belastungspfad „Ingestion“ verursachten Strahlendosen bestimmt.

Die Berechnungen der Ganzkörper-, Schilddrüsen- und Knochendosen zeigen, daß der Verzehr radioaktiv kontaminierter Nahrungsmittel entscheidenden Einfluß auf die Höhe der Dosen hat, wenn nicht ein Verzehrverbot spätestens nach dem Beginn der Freisetzung — acht Stunden nach Unfallbeginn — ausgesprochen wird.

Die *Schilddrüsendosis* (Bild 3) bleibt überall unter der Grenze der Gefährdungsklasse I, wenn innerhalb von acht Stunden nach Unfallbeginn ein Verzehrverbot ausgesprochen wird. Die Schilddrüsendosis rührt dann hauptsächlich von der Inhalation von flüchtigem Jod her.

Würde ein Verzehrverbot allerdings erst nach 32 Stunden (24 Stunden nach Emissionsbeginn) wirksam, so würden die oberen Grenzwerte der Gefährdungsklasse II für Kleinkinder innerhalb eines Umkreises von etwa 5 km und für Erwachsene unmittelbar am Kraftwerkszaun überschritten (Bild 4).

Hauptursache für diese Dosisbelastung ist der Verzehr der Milch von Kühen, die in diesem Bereich geweidet haben.

Die „Rahmenempfehlungen“ des BMI gehen allerdings davon aus, daß keine Ingestion (Aufnahme radioaktiver Substanzen mit der Nahrung) stattfindet, sondern gründen ihre Empfehlungen ausschließlich auf die Inhalationsdosis, weil als selbstverständlich vorausgesetzt wird, daß bei einem Reaktorunfall sofort ein Verzehrverbot erlassen wird. In diesem Fall wären keine besonderen Gegenmaßnahmen erforderlich; lediglich ein Verbleiben im Haus wird als zweckmäßig angesehen (Bild 3).

Die *Ganzkörperdosis* (Bild 5) wird hauptsächlich durch innere Bestrahlung verursacht, die von inhaliertem Strontium herrührt. Die äußere Bestrahlung durch die Bodenstrahlung spielt nur eine untergeordnete Rolle, da vorausgesetzt wird, daß innerhalb von sieben Tagen Gegenmaßnahmen ergriffen werden, falls sie erforderlich sein sollten. Die Dosiswerte liegen weit unter dem Grenzwert der Gefährdungsklasse I; Hausaufenthalt ist eventuell zweckmäßig, aber nicht erforderlich.

Die *Knochendosis* (Bild 6) ist ebenso wie die Ganzkörperdosis im wesentlichen auf die Inhalation von Strontium zurückzuführen. Die „Rahmenempfehlungen“ enthalten keine Hinweise auf die Knochendosis. Die Dosiswerte erscheinen jedoch unbedenklich, wenn man sie im Verhältnis zur Ganzkörperdosis sieht und dabei den üblichen Faktor 6 berücksichtigt.

Eine anschauliche Vorstellung von den gesundheitlichen Auswirkungen der zu erwartenden Strahlendosen kann man gewinnen, wenn man sie in Beziehung setzt zu der Dosis, die jeder Mensch durch die natürliche radioaktive Strahlung unvermeidlich erhält. Ein Vergleich zeigt, daß die Ganzkörper-Unfalldosis in einem Kilometer Abstand vom Kernkraftwerk etwa dem Zehnfachen, in fünf Kilometer Abstand etwa dem Doppelten der Jahresdosis aus natürlichen Ursachen entspricht.

Zusammenfassung

Die Untersuchungen haben gezeigt, daß auch bei extrem unwahrscheinlichen Unfällen die Dosisbelastungen außerhalb des Kraftwerksgeländes die oberen Grenzwerte der niedrigsten Gefährdungsklasse der „Rahmenempfehlungen“ nicht überschreiten. Die dort aufgeführten Notfallmaßnahmen sind deshalb nur in eingeschränktem Umfang erforderlich. Insbesondere müssen für die Radien der einzelnen Schutzzonen nicht die in den „Rahmenempfehlungen“ genannten Maximalwerte zugrunde gelegt werden; um den Schutzzweck zu erreichen, genügen wesentlich kleinere Radien. Es ist jedoch erforderlich, Vorkehrungen zu treffen, daß spätestens acht Stunden nach Unfallbeginn der Verzehr von Nahrungsmitteln, die kontaminiert sein könnten, untersagt wird.

Schlußbemerkung

Bei der Umsetzung der errechneten Strahlenbelastungen in konkrete Notfallmaßnahmen sollte berücksichtigt werden, daß die Rechnungen von sehr pessimistischen Annahmen ausgehen und die Rechenverfahren selbst zum Teil erhebliche Sicherheitszuschläge enthalten. So treten z.B. die atmosphärischen Verhältnisse, die den Ausbreitungsrechnungen zugrunde liegen, nur in 0,2 Prozent aller Fälle auf; bei allen anderen Wetterlagen ergeben sich günstigere Resultate. Die Schutzwirkung von Gebäuden wird ebenfalls völlig außer acht gelassen. Obwohl schon der normale Aufenthalt in einem Haus die Dosis um einen Faktor 3...10 reduziert, wird für die Rechnung unterstellt, daß sich alle betroffenen Personen sieben Tage lang ständig im Freien aufhalten.

Die angegebenen Strahlenbelastungen sind daher als eine Art oberer Grenzwerte anzusehen, die nur in extrem seltenen Fällen (Eintrittshäufigkeit für diesen Reaktorunfall mit ungünstiger Wetterlage kleiner als einmal in 100 Millionen Jahren) erreicht werden. Die tatsächlich auftretenden Strahlendosen dürften unter den hier angegebenen Werten liegen. Ausreichend Zeit für das Erkennen der Unfallsituation und die Einleitung adäquater Notfallmaßnahmen steht wegen des langsamen Unfallablaufes in jedem Fall zur Verfügung.

Nuclear Energy, 2000, **39**, No. 4, Aug., 231–236

Status of the pebble bed modular reactor*

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Eskom, South Africa

Eskom is the South African state electricity utility, with an installed capacity of 38 397 MW at the end of 1996 (some 98% of all national generating assets). It is largely coal-based with a small proportion (5%) nuclear. As part of Eskom's long-term planning process, investigations have been made into new power generation options. On reconsidering the nuclear option, Eskom identified two key issues: cost and public acceptance. It was considered that both of these were driven by the safety issues related to potential accidents and the only way to obtain competitive costs with nuclear power was to remove the potential (however remote) for accidents with significant off-site consequences. The only reactor type that was seen to meet this safety standard was the pebble bed modular reactor (PBMR). This paper discusses the PBMR project history, plant performance and design, its benefits, safety features, and current status. It concludes that the PBMR will provide South Africa with a competitive option for coastal generation and, internationally, it will be highly competitive with virtually all other generation options.

Introduction

Eskom is the South African state electricity utility, with an installed capacity of 38 397 MW at the end of 1996 (some 98% of all national generating assets). It is largely coal-based with a small proportion (5%) nuclear. The current nuclear station (Koeberg) has two pressurized water reactor (PWR) units of 965 MW (gross). Koeberg is at Cape Town, some 1400 km from the nearest coal-fired station. Koeberg's units were commissioned in 1984/5. The overall cost of utility operations in 1996 was 2·1 cents/kWh (US). This includes all generation, transmission and distribution costs.

As part of Eskom's long-term planning process, investigations have been made into new power generation options. The option of further light-water reactors (LWRs) was investigated extensively in the late 1980s and rejected due to the cost penalty compared with Eskom's coal-fired options. The average price of coal delivered to Eskom's station is in the order of $7·50/t.

In 1993 Eskom reconsidered the nuclear option (specifically for load centres away from coal fields) and again discounted LWRs. Further investigation was done and two key nuclear issues were identified: cost and public acceptance. It was considered that both of these were driven by the safety issues related to potential accidents and that the only way to obtain competitive costs with nuclear power was to remove the *potential* (however remote) for accidents with significant off-site consequences.

The only reactor type that was seen to meet this safety standard was the small high-temperature reactor (HTR), using coated particle fuel. This design was then investigated in much greater detail and the pebble bed modular reactor (PBMR) concept design was undertaken. The reason for the level of detail in the concept design was to ensure that the safety arguments were valid and that the costing could be effectively done.

The key requirements for such a programme are seen to be

(*a*) adequate technology level in the local industry
(*b*) large enough utility to provide backing for the project
(*c*) non-prescriptive licensing regime
(*d*) cost structure for power generation that excludes current technology.

All these are seen to be met in the South African situation, with Eskom being well placed both in terms of its size and its legal position.

Background

The technological features used in the PBMR are based on the experience gained in a number of projects leading to a high degree of confidence in the basis of the design. The technologies derived from each project differ but in all cases are fundamentally based on

*This paper was first presented at the fourth BNES/BNIF Nuclear Congress, 1–2 December 1999, Royal Lancaster Hotel, London.

extensive research of materials, components, fuel, core, and overall plant technology. Specific examples include the neutronic design of the 265 MWe PBMR reference proposal based on the HTR-MODUL (200 MWe) and HTR-100 (250 MWe) designs by Siemens/Interatom and BBC/HRB respectively. This concept refers to well-proven technology demonstrated by the operational histories of the AVR experimental reactor in Jülich and the THTR demonstration plant in Uentrop-Schmehausen.

The current cost of relevant technology projects worldwide totals billions of US$. Furthermore, the licensing of the HTR-MODUL reactor design in 1987 for commercial operation in Germany demonstrates that the key technologies have been mastered.

As stated earlier, the fundamental concept of the Eskom design is aimed to achieve a plant that has no physical process, however unlikely, that could cause a significant radiation-induced hazard outside the site boundary. This is principally achieved in the PBMR by demonstrating that the integrated heat loss from the reactor vessel exceeds the decay heat production in the post-accident condition and that the peak temperature reached in the core during the transient is below the demonstrated fuel degradation point and far below the temperature at which the physical structure is affected. The prospect of a 'core melt' scenario is therefore zero. Heat removal from the vessel is achieved by passive means.

Present issues

Perceived nuclear risk

It is possible to build a nuclear power station which meets the necessary technical (and licensing) requirements and demonstrates a level of safety far above other industries which still do not have public acceptance. In many ways it is the issue of public acceptance that has been the 'Achilles' heel' of the current generation of nuclear plant. This is based on two perceived problems: disposal of waste and accidents. The waste problem has been technically addressed in terms of feasibility but will need to be 'seen' to be implemented. The reason that the final waste repositories for spent fuel have not yet been fully established is down to economics. It is reasonably cheap to store spent fuel at power stations and the longer the spent fuel is stored the less decay heat generated and therefore the lower the cost of the final repository. The actual risk from such repositories has been shown by Swiss and Swedish work to be extremely low but, until the first operational repository is in place, the perception of a high-risk activity will remain.

The issue of accidents also must be seen to have been solved. The classic question is 'Can the nuclear plant have an accident which could affect the public?' The answer for the current generation of plants is 'Yes, but it is such a remote possibility that . . .'. The only part of this answer that is heard is the first word; the rest is only limited mitigation! To be acceptable, the answer must be 'No'. There must be no physically credible event which can cause off-site actions to be required.

Costs

The costs of the current generation of nuclear power plant increased significantly (in real terms) between 1970 and 1990. This was not due to specific design changes but the increased cost of 'safety', as expressed by QA requirements, design controls, personnel training, etc. The changes in the physical design in this period were not significant (in cost terms), but it was the surrounding infrastructure that caused the cost increases. These cost increases moved the nuclear option from being one of the cheapest options for the first world in 1970 to one of the most expensive in 1990, as the real cost of fossil generation fell. These cost increases are linked to the perceived risk concern noted previously. The problem therefore is that as long as the potential for accidents exists, the costs will always have an upward pressure.

This overall situation led to the position today, where the public believes that nuclear power is more expensive than fossil fuels by its very nature—the opposite of the position in 1970.

PBMR project history

As stated earlier, the PBMR project was launched as part of Eskom's integrated electricity planning supply side working group activities. The initial work in 1993 was due to a request to review potential nuclear options. During this review the possibility of a small, inherently safe reactor based upon German high-temperature reactors was identified. The funding was initially small but grew as the potential economic benefits were increasingly confirmed by the work being done. Throughout the project the targets set were to be competitive on cost with the large Eskom mine-mouth coal-fired power stations, without limitations on siting.

The concept design and costing studies showed that the technology being adopted for the base-line design has been adequately demonstrated to avoid fundamental technical risks. This is supported by technology contracts with the original commercial developers in Germany (Siemens and ABB) through their subsidiary (HTR GmbH) as well as the related research institute (KFA in Jülich). To support other key technology areas there has been detailed involvement of overseas suppliers.

There have been over ten independent reviews of the technical and commercial aspects, both those funded by the project and those requested by potential joint venture partners. The only two specific concerns raised were over the back-end fuel cycle costs (by the International Atomic Energy Agency (IAEA) expert on nuclear costing) and the performance of the recuperator. In the case of the back-end fuel cycle costs the figures for the PBMR are based on the agreed rationale for Koeberg. The current cost of disposal is less than 1% of the fuel cost and therefore even a large (factor of 10) increase would not significantly increase the levelized cost of power. In the case of the recuperator, the concern is principally over the compactness of the design and this has been addressed by doubling the available volume for the recuperator. This has not had a significant impact on the overall costs. During the period of these reviews the available core power was increased, by changes to the fuelling regime, from 226 to 265 MW, while increasing the available margins (i.e. lowering the peak fuel temperature during loss of cooling events). At the same time the reflector structure was made substantially simpler.

These reviews have included a market survey for this class of plant covering 19 countries in depth. This market research indicated a substantially larger overseas market than used for the economic evaluations. This is because of the strong cost advantage of this design over other options where very cheap coal is not available.

The IAEA has been formally requested by the South African government to investigate and advise on the technical and economic feasibility, safety and proliferation aspects of the PBMR. This study is already in progress and several meetings have taken place in South Africa and at the IAEA. It is expected that the final report to the South African government will be made early in the year 2000, and that this will enable the government to make a decision on the matter.

Plant performance

The plant performance figures are shown in Table 1. The dynamic performance of the design has been validated by use of an engineering simulator that has been developed.

Plant design

The aim of this paper is not to explain the reactor design in detail; however, the overall plant layout is shown diagrammatically in Fig. 1. The process cycle used is a standard Brayton cycle with closed-circuit water-cooled inter-cooler and pre-cooler. Separate turbo-compressors and power turbine are used. This separation simplifies the design and qualification process while compressor bypass valves are used for short-term control. All the bearings in the cycle are of the magnetic type which avoids any contaminants in the helium circuit and limits maintenance. The reactor systems are placed inside a reactor pressure vessel and the turbo-machinery and heat exchangers in a power conversion unit.

Table 1. Plant performance figures

Description	Performance
Thermal power	265 MW
Maximum generated power	117 MW
Maximum distributed power	114 MW
Minimum power (high efficiency)	20 MW
Continuous stable power range	0–100% nominal
Ramp rate (0–100 MW)	10 MW/min
Load rejection without trip	100%
General overhauls	30 days per 72 months
Plant life	40 years
Fuel movement	On-load
Staff level for ten-unit site	80
Capital cost	~$100 million/module
Fuel cost	~$4/MWh
Construction period	24 months
Emergency planning zone	400 m

Benefits

In light of this extensive work the project has been analysed from three perspectives to gauge its value to the country, the utility and the investor. These are discussed in the following subsections.

National benefits

In studying the PBMR, Eskom established a scenario (called the base case) to allow analysis. The project has been subjected to an input–output analysis, based upon this base case model for construction of the units only (without the development and fuel projects). The model assumed ten units/year for local construction and 20 units/year for export. The South African content of the local plant is 81% and 50% for export plant. These content values are based upon the plant equipment breakdown and a realistic assessment of current local manufacturing capability.

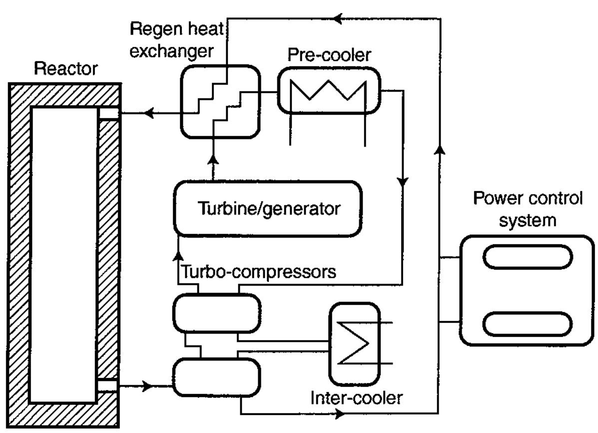

Fig. 1. Overall plant layout

Table 2. Effect of project on South African economy

Description	Local	Export	Total
Permanent jobs created	92 710	111 836	204 546
GDP/year (1998, million rands)	7734	10 597	18 331
BOP/year (1998, million rands)	−791	6488	5697

The choice of 20 units/year for export was based upon an assessment of the world market undertaken for Eskom and would equate to a 2% share of the overall world power market or (over the 20 years considered) approximately 14% of the replacement market for current nuclear plants. The ten units/year for the local market was based on the long-term Eskom growth trend of 3·53% (1980–93), which equates to 1500 MW/year on a 41 000 MW base. This equates to the long-term medium to high growth assumptions of Eskom. In both of these cases (local and export) the impact is on a linear basis, that is, the impact of one unit/year for the local market is 10% of the impact of ten units/year. This evaluation can therefore be scaled to match any assumption basis. To retain the local content, however, there must be an adequate production level (at least five to ten units per year) to maintain the full economies of scale.

This analysis showed that when the project had matured (approximately ten years) the effect on the South African economy from the local and export market was as shown in Table 2.

Utility benefits

The PBMR studies were initiated to meet a future need for distributed generation at a cost competitive to Eskom's current coal generation. The advantages of the PBMR over any other identified options are as follows

(*a*) distributed generation
(*b*) short construction period
(*c*) small unit size
(*d*) excellent load-following
(*e*) low environmental impact
(*f*) competitive economics.

The key impacts of the PBMR option on Eskom's expansion planning would therefore be to

(*a*) allow for the construction of multi-unit power stations near to coastal load centres and therefore limit the need for extensive transmission system strengthening
(*b*) reduce the uncertainty, risks and therefore the costs associated with the current long-term planning requirements
(*c*) reduce Eskom's exposure to negative environmental claims, such as CO_2 emissions and use of highveld water resources
(*d*) improve quality of supply at remote locations (e.g. Eastern Cape) without the need for new line compensation equipment
(*e*) provide an economic mitigation strategy for greenhouse gas reductions.

In terms of economics it is of note that the development cost of the PBMR is some 2·5% of the capital cost of a 4000 MW coal-fired station, or the interest on a three-month total project delay.

Investor benefits

As stated earlier, the project was started solely on the basis of the utility requirements. It is, however, clear that the economic advantages of the PBMR would not be limited to the South African grid. Unlike Eskom's other low-cost options (coal and hydro), the PBMR costs are virtually independent of location. The base load costs of about US 1·43 cents/kWh (based on 6% discount and a 40-year life and including 6% profit on construction and 20% on fuel) is extremely low compared to overseas costs (average cost in China is US 3 cents/kWh, in Japan is US 9 cents/kWh). There should therefore be an extensive export possibility, particularly as the safety standards being applied to the PBMR design are stricter than those applied to other modern Western nuclear stations.

Following the work in 1997, an analysis of the project investment returns was undertaken (assuming it captured $\pm 2\%$ of the world market for new power plant). This analysis assumed that an owner's generating cost of US 1·6 cents/kWh would be attractive and result in a base case where the internal rate of return on invested equity and loan capital over a 25-year period is over 25% real. Beyond the first module no sales to Eskom are included in this analysis and no external gearing is assumed. Analysis showed that the result was not highly sensitive to the start-up cost but was sensitive to the gearing of the construction period after the first unit.

On this basis the project can be seen to be a viable and attractive investment opportunity. It should be noted that the project can exploit specific competitive advantages that South Africa can offer

(*a*) adequate technology level in the local industry
(*b*) large enough utility to provide backing for the project
(*c*) non-prescriptive nuclear licensing regime
(*d*) cost structure for power generation that imposes a strong cost cap

(*e*) utility having good public image and credibility.

The competitive edge of the PBMR over potential suppliers of similar technology will be maintained due to the initial time lead (which must be protected) combined with an ongoing active product development and enhancement programme funded from revenue. This is included in the current financial model as 4% of total turnover.

Safety comparison

As was explained previously, the design was aimed to achieve a 'catastrophe-free' design, irrespective of probability. In support of this, a study into the risks of the PBMR compared to other nuclear activities was established. It should be noted that the risk from other industrial activities exceeds that of the highest nuclear risk by a substantial margin. Under these analyses a PBMR could, for all sensible purposes, be considered to be a normal industrial plant, in terms of siting and emergency planning. Table 3 illustrates the low risk posed by a PBMR.

Current status

Overview

The PBMR programme has achieved the following milestones

(*a*) application for nuclear licence of the design from the South African regulator (Council for Nuclear Safety (CNS))
(*b*) initiation of environmental impact assessment process to allow approval of the first site
(*c*) establishment of single programme team (of over 80 full-time staff) outside the utility head office
(*d*) finalization of concept design
(*e*) prequalification of key suppliers
(*f*) negotiations with potential joint-venture partners
(*g*) initiation of tender process for detail design of long lead items.

This is intended, when combined with public and stakeholder consultations, to enable the decision on the potential construction of the first reactor to be taken in due course. This would include full consideration of the development, construction and costs, design parameters and the site location.

Licensing

The South African licensing system is based on a fundamentally probabilistic basis, with the requirement to meet international standards. This has been the case since the beginning of the South African nuclear power programme in the early 1970s. The process requires that any nuclear activity (including the design process) shall be covered by the CNS.

In light of this, and the need to establish the design rules for the PBMR, Eskom has formally applied for a licence for the PBMR design. The application has started a programme aimed to achieve the initial 'licensability' statement on the PBMR by the first quarter of 2000. This will include the safety criteria that the plant must meet, the general and specific design criteria, the event list, the classification systems, and a review of the design basis. These activities are under way with the involvement of international consultants (both to support the CNS and Eskom).

Environmental impact assessment (EIA)

Under South African legislation there is a requirement for a full EIA for any new power plant. In light of this, Eskom has started the process and has called for potential suppliers to submit their capabilities. A formal inquiry to the qualified suppliers will be issued shortly. As part of the EIA process there will be a number of coastal sites considered, as well as the overall societal impact as to the value of the PBMR project (to allow for the 'no go' option). The decision

Table 3. Risk index

Index	Description
37*	Public† risk due to accidents, including a 'Chernobyl' at a PWR‡
35	Occupational risk due to normal operation at a PWR
6·1	Public risk due to normal operation of reprocessing plant
1·8	Public risk due to normal operation of a PWR¶
1·4	Public risk due to normal operation of an APWR‖
0·4	Public risk due to accidents, including a 'Chernobyl' at an APWR
0·2	Occupational risk due to normal operation of reprocessing plant
<0·1	Public risk due to accidents, including the most severe, at a PBMR

* For example a risk index of 37 means a dose of 0·37 manSv/TWh.
† The public risk is for members of the public living within 1000 km of the installation.
‡ The PWR chosen is one of those at Tricastin. It is typical of modern reactors.
¶ The public risk due to normal operation is the environmental risk.
‖ The advanced PWR (APWR) chosen is Sizewell B. It incorporates additional safety features compared with the PWR.

has now been taken to consider only the existing Koeberg site for the first unit.

Engineering

The concept design is now largely complete, with the basic design process under way. Those areas of the plant which are not covered in significant detail are those which are standard 'off-the-shelf' equipment and do not contribute significantly to the cost (e.g. the air compressor system), but even in these cases a performance specification has been generated.

The load-following capability has been a very specific feature of the work to date. There is an engineering simulator (based on a G2 platform) which has been developed to allow non-real-time dynamic analysis. There are two other system analysis tools being used in the cycle development and these are FLOWNET and a MATHCAD-based model. While these tools are sufficient to handle the concept and basic design phases, they are not seen to be appropriate for the detail design phase and a new engineering simulator is under development with a view to being in service by the middle of 1999.

Another key area has been the maintenance analysis. There has been extensive work on the maintainability of the design, and all components are classified by their life expectancy and difficulty of repair. This leads to some components having very easy access (e.g. the bypass and interrupt valves) which can be maintained without breaching the helium circuit, and some which can be changed, but only in the same manner as the changing of steam generators on a PWR (e.g. the recuperator). This analysis has allowed the maintenance cycles and removal routes to be established (along with, for example, their impact on building design and crane requirements).

In particular areas there are test rigs under construction. These are required to demonstrate very specific features which would be valuable to include in the design. These are separate from the inquiry documents now being issued for the detail design and manufacture of long lead components. (Note: the manufacture option in these contracts would only be exercised once the commitment to the first unit has been made.)

Fuel manufacture

The manufacture of fuel is a key element of the PBMR programme, and the quantities required exceed any previous high-temperature gas-cooled reactor (HTGR) project. Therefore, while there are facilities to construct HTGR fuel on a small scale currently in the world (in Japan and China, for example) the PBMR requires a new fuel manufacturing facility. The present intention is to construct it at the South African Atomic Energy Corporation (AEC) site, in the complex that built the fuel for Koeberg. This project has identified the layout for such a line (to initially manufacture 1·4 million spheres/year) and the equipment specifications. Discussions are now being held with vendors for the equipment. In terms of the actual fuel technology, Eskom has involved a number of suppliers and potential suppliers in various ways (such as partnership agreement, commercial contract, or in some cases negotiations are still under way). In the meantime the AEC, under Eskom contract, has started laboratory work. This work is to support the external technology that is being obtained.

Conclusion

The PBMR has a number of advantages over other potential power sources. In South Africa it will provide the country with a competitive option for coastal generation. Internationally, it will be highly competitive with virtually all other generation options.

SAFETY CONCEPT AND DESIGN OF A MODULAR GAS COOLED HIGH-TEMPERATURE REACTOR FROM THE VIEWPOINT OF EXTERNALLY GENERATED LOAD CASES

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Abstract:

The HTR-MODUL is a system which incorporates the advantageous safety characteristics of the HTR reactor design. A comparison with the conventional HTR concept with regard to shut down, decay-heat removal and safety barrier systems shows that they become much less important if not totally unnecessary for the HTR-MODUL design, from the viewpoint of safety. This higher level of inherent safety is due mainly to the reduced reactor size and power density as well as a special geometric design. The result is a maximum possible accident temperature of 1600° C under which the integrity of the fission-product barrier remains.

Externally generated load cases - the two main load cases to be considered in Germany, earthquake and aircraft impact - already constitute a significant proportion of the safety risk for conventional reactor designs. The greatly reduced internally generated safety risks associated with the HTR-MODUL result in these externally generated load cases becoming dominant and very important with regard to any overall risk reduction.

The system's integrity with regard to safety under any eventuality is totally assured so long as the geometry which ensures passive heat removal is maintained. A single failure in the outer confinement does not in itself constitute a safety risk. Only several further failures of a specific nature, involving the pressurized primary system, could cause safety related problems. The main aims of any design appertaining to earthquake or aircraft-impact loads must be therefore to maintain a geometry which allows passive cooling and ensures the integrity of the pressurized primary system. These aims are most effectively reached by a special reactor building design which attempts to isolate the reactor components from the externally generated loads.

A design concept for the building will be analysed. This design maintains a high degree of reactor component isolation, which protects the components from aircraft induced vibrations, while still retaining an overall stiffness high enough to negate magnification of the important low frequency seismic loads.

1. Introduction

Within the last years reactor development has tended to the increase of power output per reactor unit.

As far as HT-reactors are concerned the advantages of the high temperature level was not sufficient to promote the construction of a high power plant.

One of the major reasons is that during the development technical difficulties became apparent: with growing power output the inherent physical properties of HTRs become less and less effective with respect to controlling accidents so that active safety systems have to be introduced.

In a situation characterized by growing political opposition to the establishment of nuclear plants it could be attractive to reach high power output by connecting smaller units in series - that means modular technology. We may assume that by preserving the inherent safety characteristics of the low power HTRs simple construction techniques become possible and as a consequence uncomplicated licensing and calculable erection time will be regained.

2. The Modular Unit

To reduce the costs for development the operating AVR-pebble bed reactor is used as a basis.

The principal point with regard to safety is the criterion that under all circumstances with meaningful occurrence rates the core temperature will not exceed 1600° C. So the fission product barrier within the pebbles will be retained.

For keeping under this temperature limit neither an active heat sink within the primary system nor an active shut down device should be necessary.

The resulting modular unit is shown in figure 1:

The reactor and primary heat sink have a joint steel pressure vessel. The gas exchange takes place beneath the pebble bed. In the case of break-down of the primary heat sink, radiation and convection is sufficient for a surface cooler to facilitate the necessary heat transfer, even if the shut down device fails or a depressurisation accident occurs. The dimensions of the core are controlled with respect to the diameter and power density by the 1600° C - temperature limit and with respect to the height by pressure loss and power density distribution. So the power of a modular unit is limited to approximately 170 - 200 MW depending on the use (e.g. steam generation, process heat production).

From the safety point of view, severe problems can only be caused by extreme damage of the part of the pressure vessel that belongs to the reactor or by the failure of the surface cooler.

For a detailed description of the HTR-Modul concept you are kindly asked to look for REUTLER / LOHNERT : "The Modular HTR ", to be published in NUCLEAR TECHNOLOGY.

3. Some Aspects of the Design Against External Events (EE)

As mentioned above it is necessary to ensure the integrity of the following components in the case of EE:

- integrity and stability of the reactor pressure vessel;
- the operational ability of the surface cooler.

Such calculations are standard practice. It is not necessary to ensure that the ceramic structures remain in a good condition because neither gas conduction nor shut down ability is asked for after or during extreme external events.

On the other hand it is a result of the high level of safe operation of a modular plant that EE gain importance with regard to any overall risk reduction. It is worthwhile to think about the minimization of EE-risk.

This could be important for the present with regard to plant location and also for the future with respect to plant numbers.

Under the assumption that the site is known the starting point for a EE-risk minimization is obviously the design of the reactor building. It can be taken for a maxim that it is better to reduce the component loads due to external events by a special building design than to design against higher loads in case of conventional building.

4. Proposal for a HTR-Modul Reactor Building

Nuclear plants that are to be built within the FRG must be designed for the external events earthquake and aircraft impact. The typical German site is characterized by:

- low to middle values of soil shear modulus G : $9 \cdot 10^7 \leq G \leq 2{,}5 \cdot 10^8$ N/m^2
- maximum ground acceleration b_o : $b_o \leq 3$ m/sec^2
- impulse I of the aircraft: $I = 4 \cdot 10^6$ N . sec

In addition we assume a plant with four modular units. The fundamental configuration can be seen from figure 2. Due to functional reasons this configuration should not be altered.

The reactor building consists of an outer region that is the aircraft impact penetration barrier and an inner region which contains the components, in particular the primary cells. The reactor hall remains unprotected against aircraft impact. The roof of the penetration barrier contains openings above the reactors and primary heat sinks which are closed by concrete plugs during normal operation (see fig. 3).

If soft to middle soils are assumed a rigid building shows the best effects with regard to component loads due to:

- high damping values due to energy radiation;
- there is no magnification of the floor accelerations for frequencies higher than 8 Hz.

Independent of soil properties the aircraft impact induces significant accelerations in the higher frequency range if rigid couplings exist between impact point and component location.

Figure 4 shows the typical situation the component designer has to take into account in the case of a "conventional" reactor building. So what is needed is a building that reacts monolithically with respect to the earthquake and nevertheless weakens vibrations of higher frequencies decisively on their way to the component locations.

The building concept resulting from these considerations is shown schematically on figure 5: the penetration barrier is constructed as a box within which is contained the free standing inner region. The roof of the penetration barrier is partly supported by springs on the walls of the primary cells. This leads to a reduction of the roof thickness from unfavourable 6 m to approximately 3 m without loss of decoupling in the higher frequency range. For the roof support approximately 70 spring packs with spring constants of 2×10^7 N/m would be required. With a sufficiently rigid penetration barrier and inner region the whole building reacts monolithically during an earthquake as the assumed soil acceleration of 3 m/sec^2 does not cause lifting of the inner region.

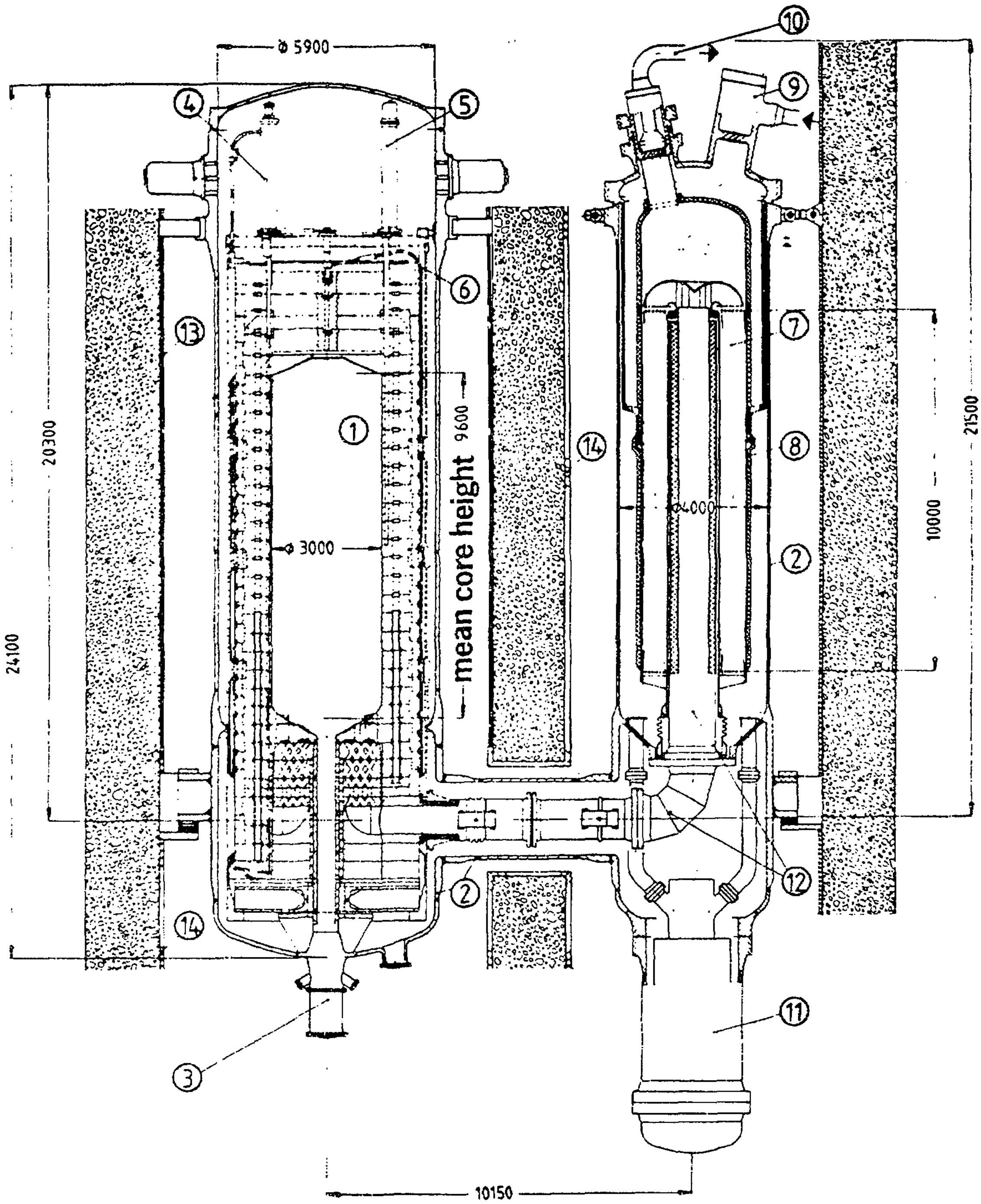
It remains to show the effectiveness of decoupling in the case of aircraft impact. The results of two typical computations are presented:

- vertical impact on the middle of the penetration barrier roof: figure 6 shows the roof displacement which reaches a maximum value of approximately $4 \cdot 10^{-2}$ m. The displacement of the inner region attains a value of about $2 \cdot 10^{-3}$ m (see fig.7)

- vertical impact on the middle of the longer wall of penetration barrier: figure 8 shows the floor response spectrum for a point on the penetration barrier, figure 9 that of an inner region point. The high frequencies are significantly filtered out.

5. Conclusion

The high operational safety of a HTR-Modul can be supplemented with respect to external events by an adapted reactor building. The design of the building results in a decoupling of components from aircraft impact induced vibrations while still providing the advantage of monolithical behaviour in the case of an earthquake.



1 Pebbel bed
2 Pressure vessel
3 Fuel discharge
4 Boronated spheres
5 Reflector rod
6 Fuel loading
7 Pipe assembly
8 Outer shroud
9 Feed line
10 Live steam line
11 Blower
12 Hot gas duct
13 Surface cooler
14 Insolation

FIG. 1 - CROSS SECTION OF A MODULAR UNIT FOR STEAM GENERATION

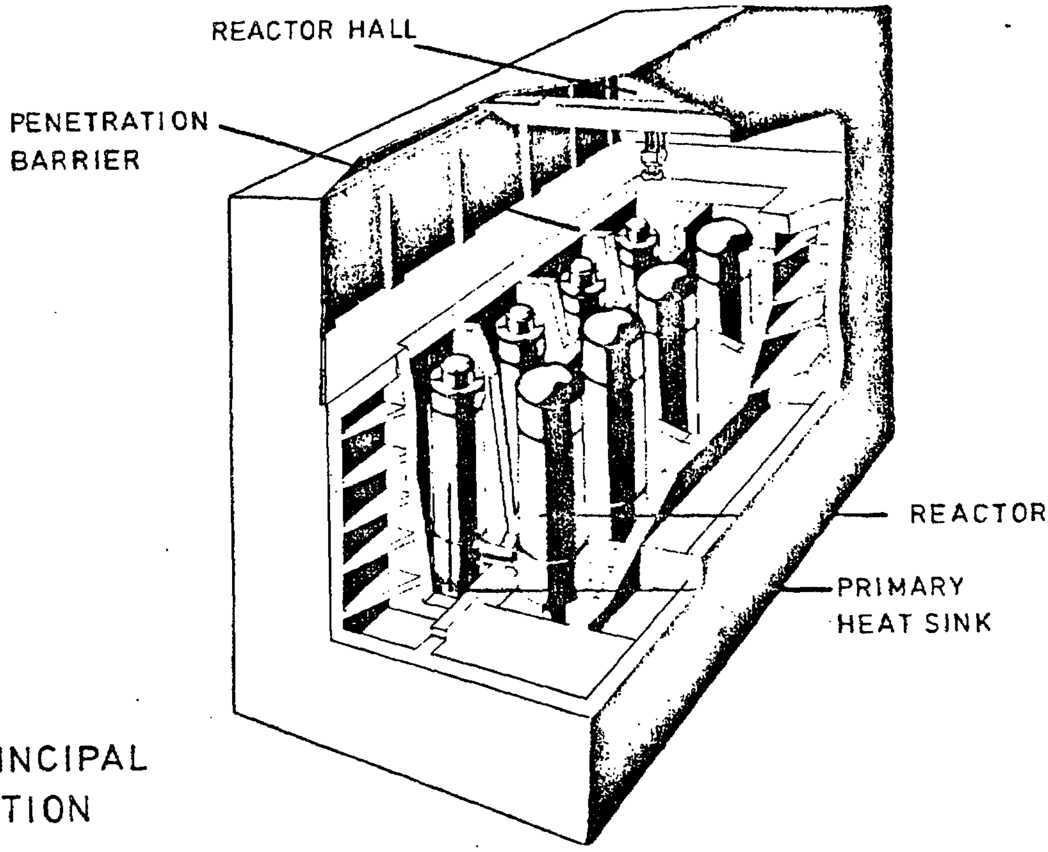


Fig. 2

VIEW OF PRINCIPAL CONFIGURATION

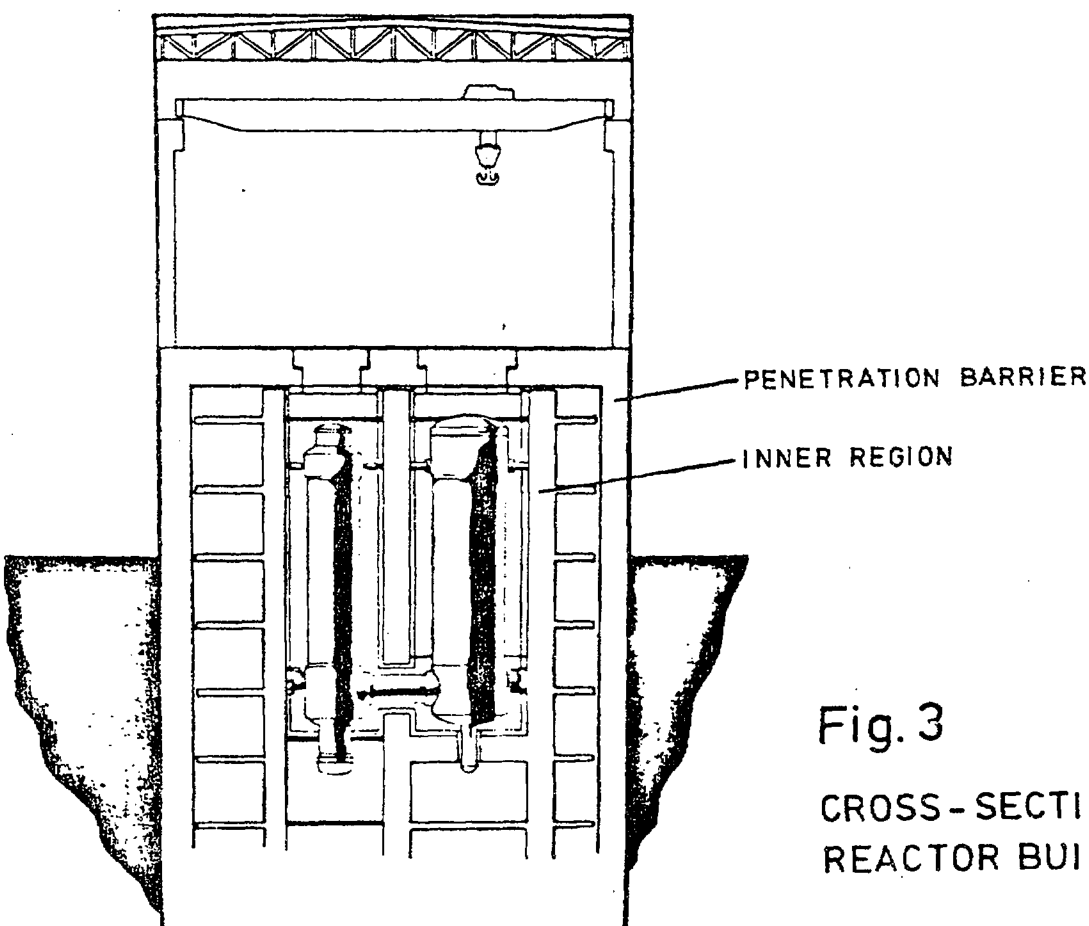


Fig. 3

CROSS-SECTION OF THE REACTOR BUILDING

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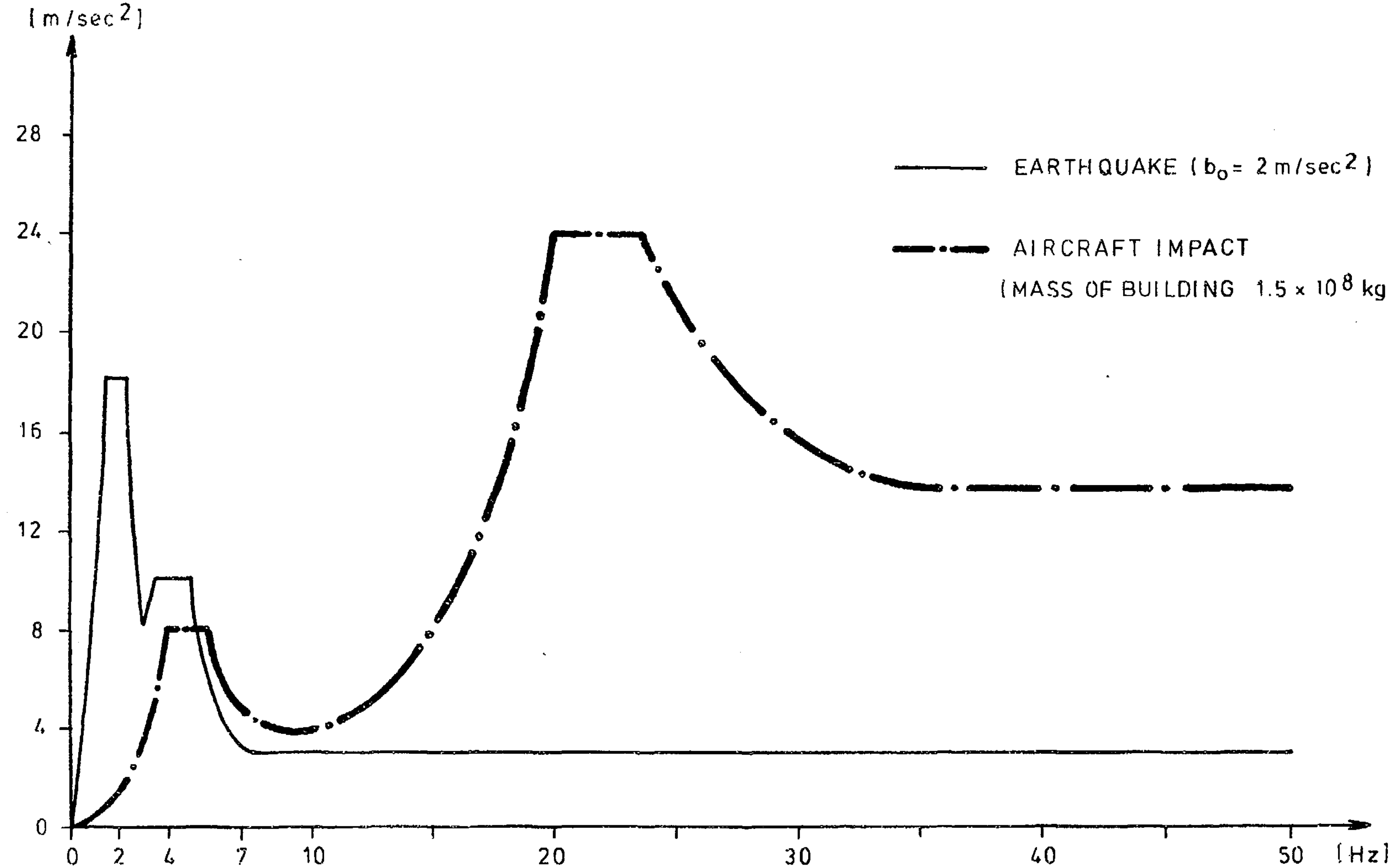
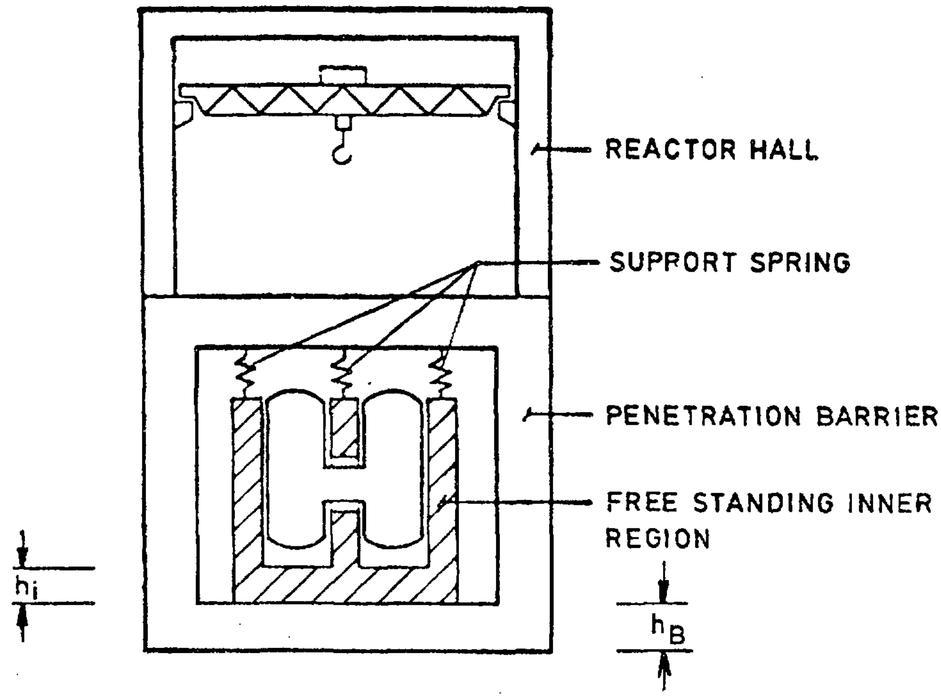


Fig. 4 FLOOR RESPONSE SPECTRA FOR CONVENTIONAL REACTOR BUILDINGS



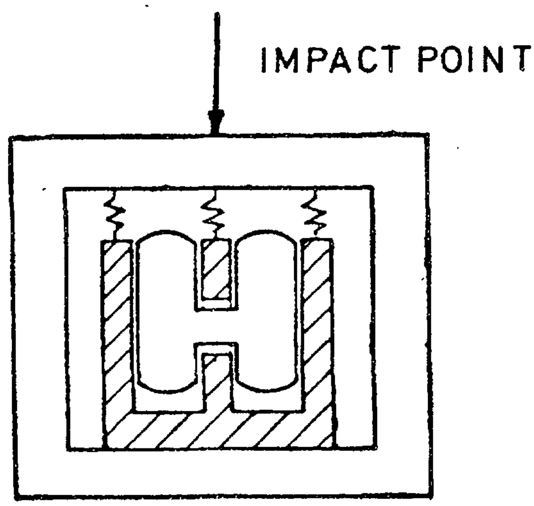
ROOF SUPPORT :

75 SPRING PACKS , SPRING CONSTANT 2×10^7 N/m

FOUNDATIONS :

EXPECTED THICKNESS $h_i \leq 3$ m, $h_B \leq 3$ m

Fig. 5 Schematic Drawing of EE-protected Building



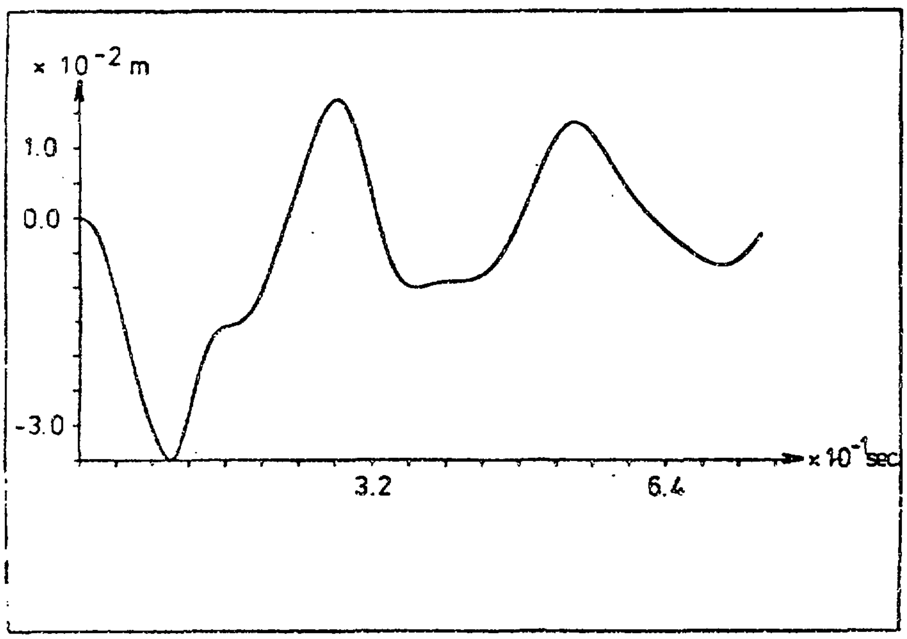


Fig. 6

ROOF DISPLACEMENT

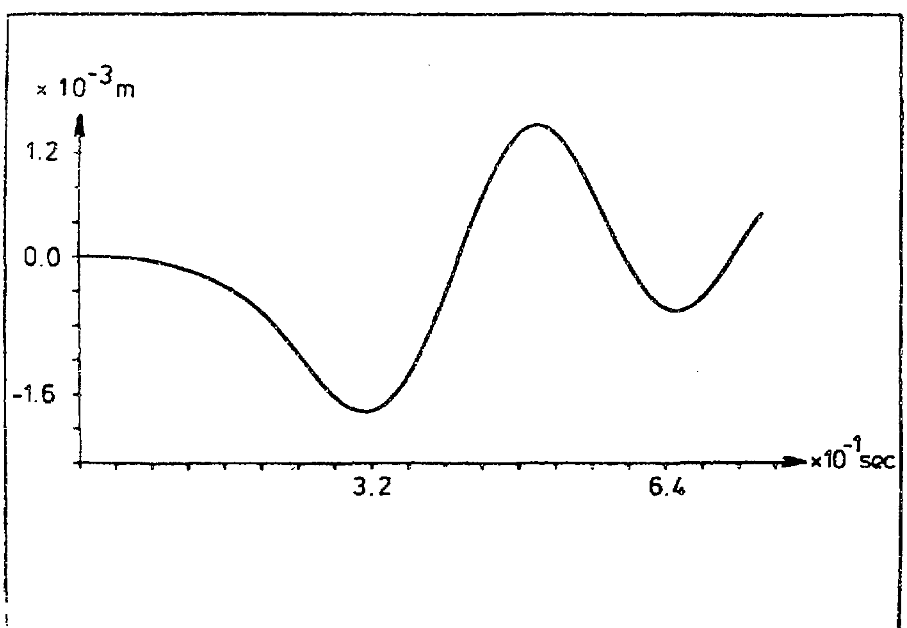
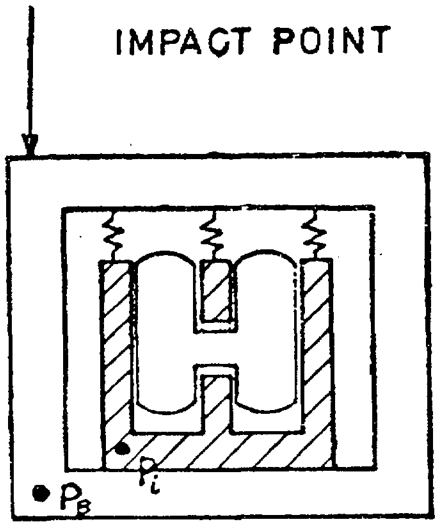


Fig. 7

DISPLACEMENT OF INNER REGION



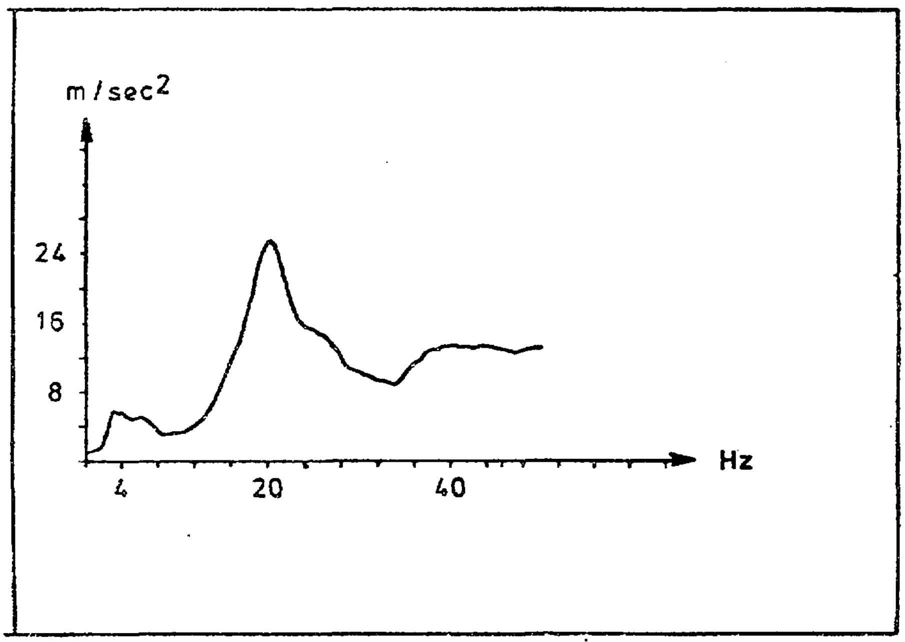


Fig. 8

FLOOR RESPONSE SPECTRUM AT POINT P_B

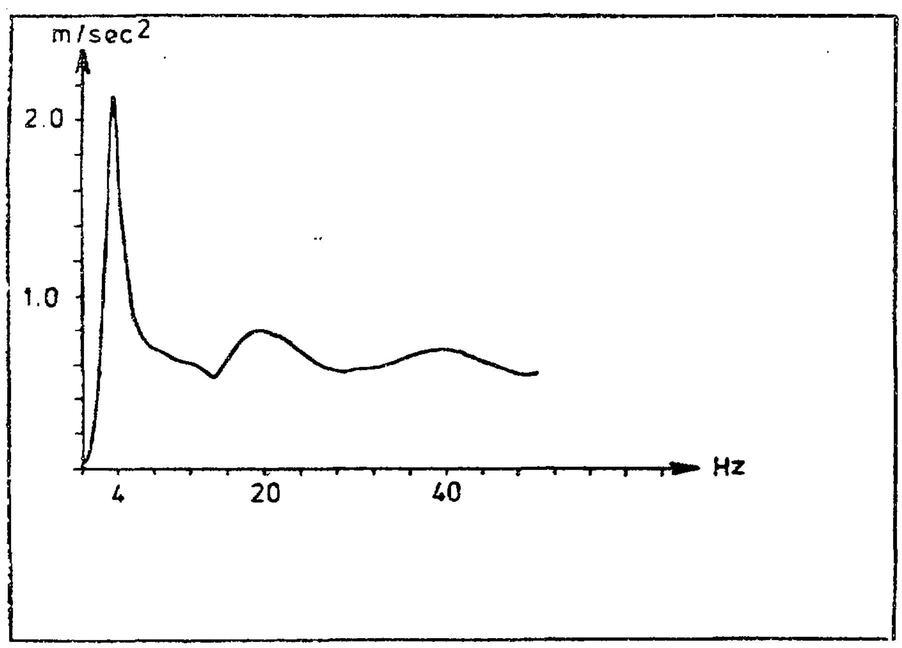


Fig. 9

FLOOR RESPONSE SPECTRUM AT POINT P_i