Introduction to Pebble Bed Reactors

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U.S. Nuclear Regulatory Commission

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(1984)

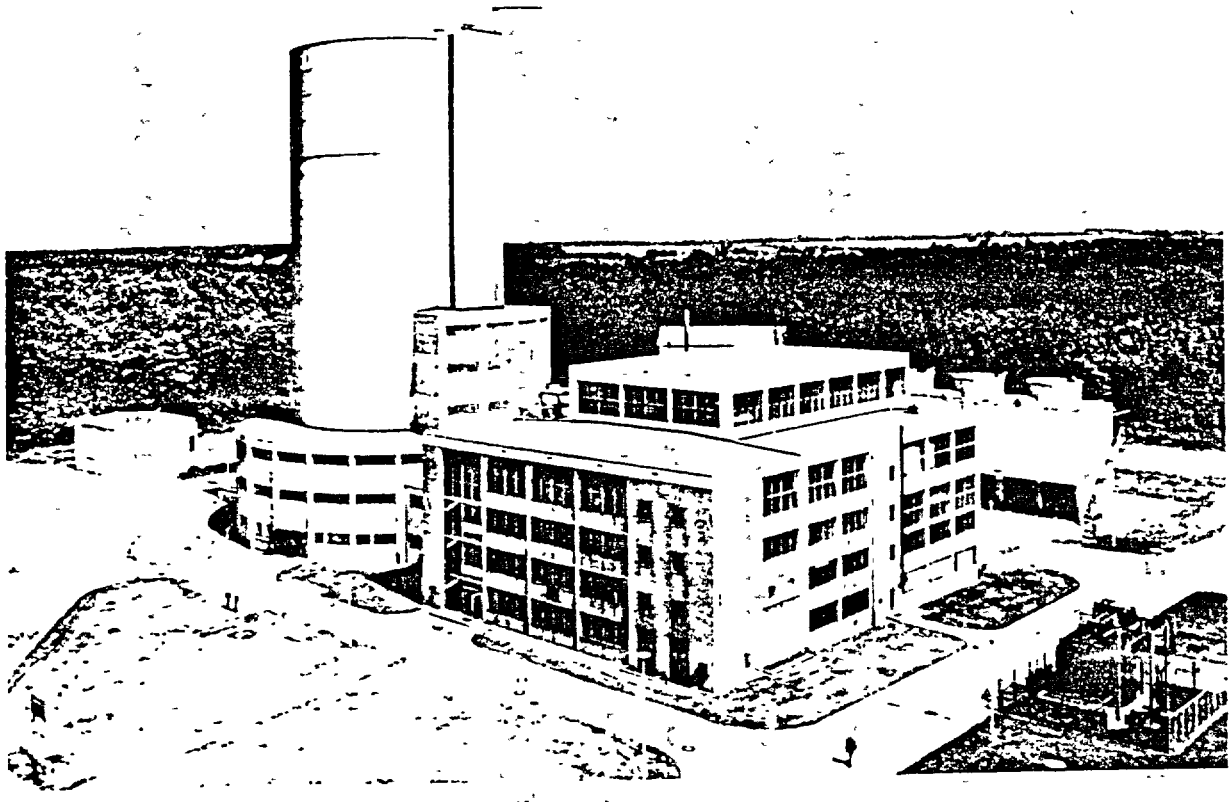


Fig. 1. View of the AVR-experimental power station with the high temperature reactor

rounded by a safety containment and a concrete shell (150 cm thick). The concrete shell is simultaneously the reactor building and the outer biological shield.

The charge and discharge of fuel elements is performed continuously during the reactor operation, so no excess reactivity is necessary to compensate the fuel

Table 1
Steam generator material

HEATING SURFACE	MATERIAL	CHEMICAL COMPOSITION IN %			
		C	Mn	Cr	Mo
ECO 1 AND 2	St 35 8	0.17	0.7	—	—
PRE - SUPERHEATER	15 Mo 3	0.15	0.6	—	0.3
SUPPORTING PIPES	13 Cr Mo 44	0.13	0.5	0.8	0.45
FINAL - SUPERHEATER	10 Cr Mo 910	0.10	0.5	2.4	1.0

Fig. 2

Types of HTGRs

- Pebble Bed Designs
  - Pebble: ~15,000 fuel particles in a 6 cm graphite ball
  - Pebble bed: Like a gum ball dispenser. Forced helium flow between ~330,000 fuel pebbles.
  - Continuous refueling: Pebbles discharged after single pass (OTTO) or multiple passes (MEDUL) thru core

- Prismatic Block Designs
  - Fuel particles incorporated with carbon binder into compacts in replaceable, stacked graphite blocks.
  - Channels for helium coolant flow.

HTR PEBBLE CROSS-SECTION CUT-AWAY COATED PARTICLE

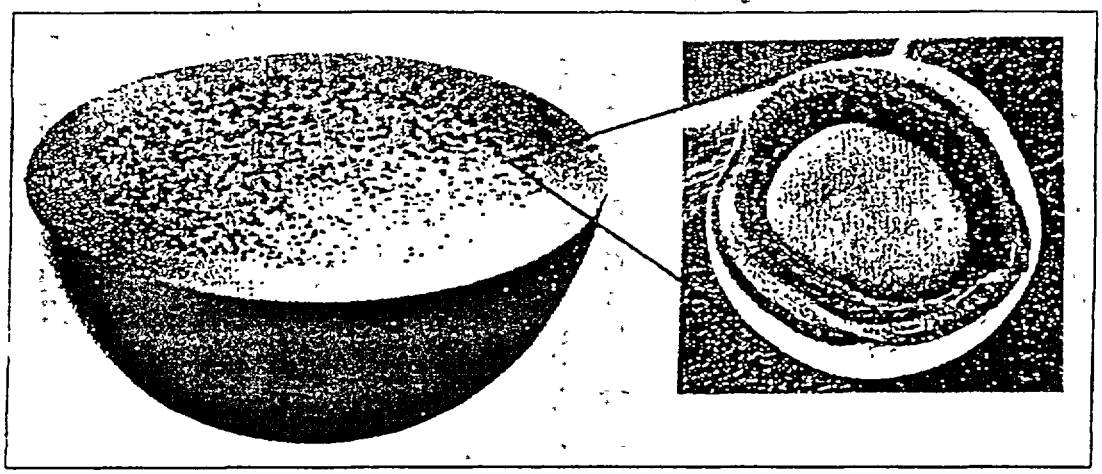


FIG 3 8. PBMR fuel particle and sphere

TABLE 3 5 PBMR FUEL CHARACTERISTICS

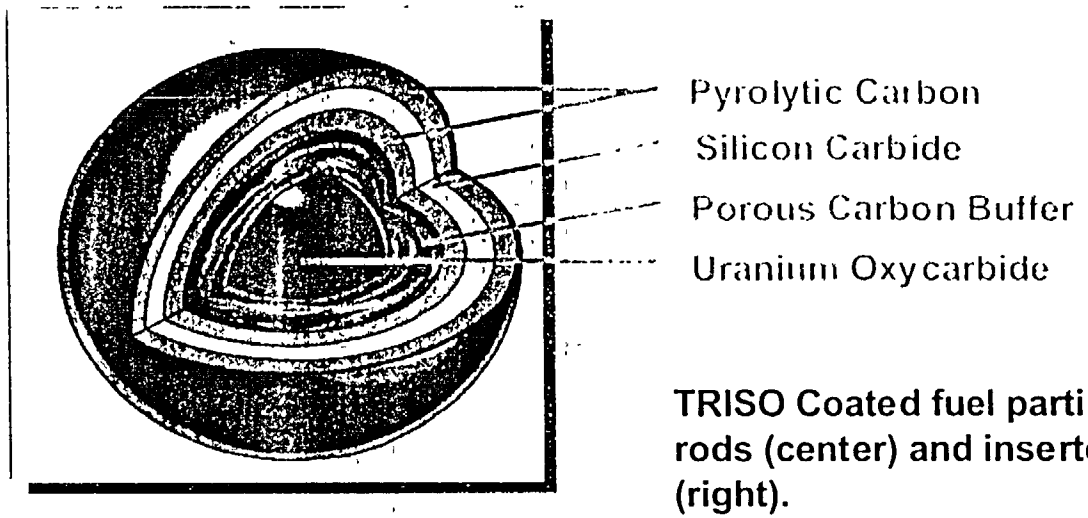
Kernels:	
Material of Kernels	UO_2
Enrichment	7.8–8.5% ($^{235}U/U$)
Density	10.4–10.75 g/cm^3
Diameter	500 μm
Coated Particles	
Outer Diameter of Coated Particles	~920 μm
Material of Coatings	C/C/SiC/C
Density	1.05/1.90/3.18/1.90 g/cm^3
Thickness	95/40/35/40 μm
Spheres	
Diameter	60 mm
Diameter of Fuel Zone	50 mm
Density	1.75 g/cm^3
Thermal Conductivity (Temperature and Irradiation Dependent)	0.17–0.39 W/cm· °C
Graphite Material (Matrix Outer Shell, and Graphite Spheres)	A3 (proposed)
Loading	
Heavy Metal Loading	9 g/sphere
No of Fuel Particles	15 000 n/sphere

contract, has started laboratory scale work including the manufacture of fuel kernels. This work is to support the external technology that is being obtained [3-2].

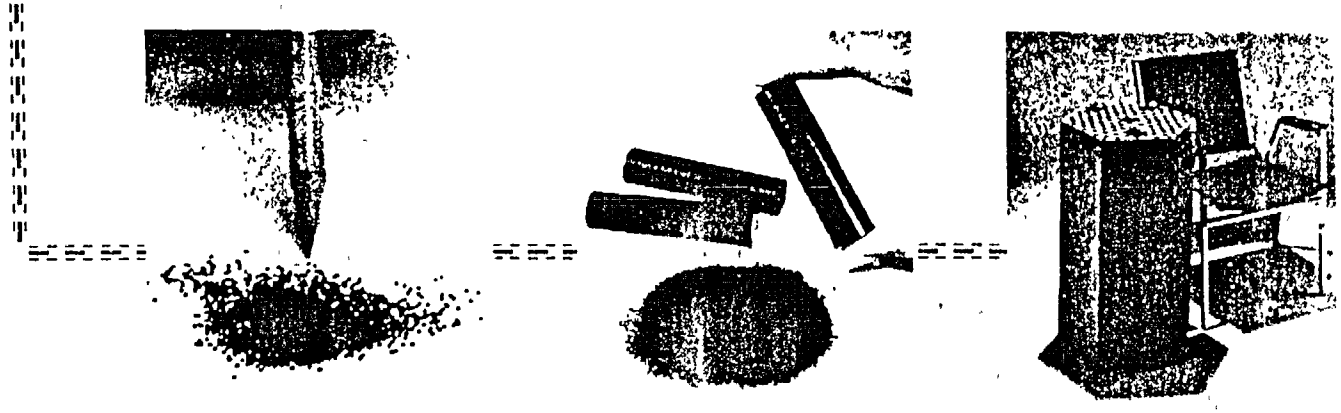
3 3 2 2. Power conversion unit [3-12]

The PCU (Fig 3.9) includes the equipment necessary to convert the heat of the hot helium from the reactor into electricity.

CERAMIC FUEL RETAINS ITS INTEGRITY UNDER SEVERE ACCIDENT CONDITIONS



TRISO Coated fuel particles (left) are formed into fuel rods (center) and inserted into graphite fuel elements (right).



PARTICLES **COMPACTS** **FUEL ELEMENTS**

GENERAL ATOMICS

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3.1. Fuel elements

Besides the demonstration of long term reliability of the reactor components, the large scale test of the different kinds of fuel elements during reactor operation is a main tasks of the AVR. At the present, there are alltogether twelve different kinds of fuel elements in the AVR, distinguished by their enrichment, ratio of uranium to thorium, chemical composition, size and quality of the fuel particles.

Due to the significance of the fuel elements testing programme, a spatial review is given in a separate article in this issue.

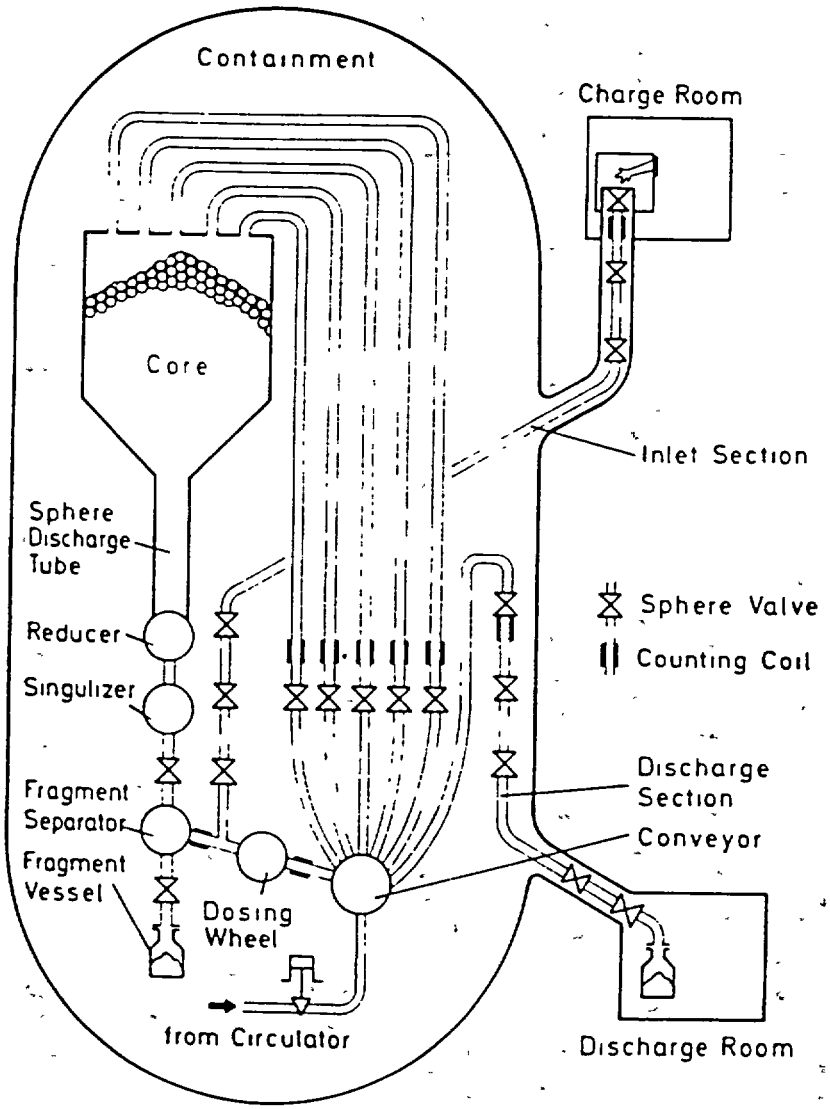

Fig. 3. Schematic diagram of the fuel handling equipment.

E. Ziermann
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production of power is neve
purposes. The AVR fuel cha
of its kind, so no knowledg
phite spheres in helium gas
nents was available prior to i
Naturally, this device has bee
has nevertheless been demon
cations can be removed with
to the staff, and without a
production. But since this eq
containment (in a low radia
has to be limited to minimiz
staff which received rigorous

3.3. Graphite and carbon block

It is not yet possible to ins
or carbon block installation
devices, so to this date not st
tion can be made. It may be
samples from the reflector b
vices which pass through t
tubes. The operation of the p
changes in the ceramics insta
investigation of all these con
after plant decommissioning.

3.4. Shutdown rods

The shutdown rods are ins
the core by the weight of a co
of removal and insertion is co
None of the rod mechani
pairs so far. Only as a preca
transmission, and siphon glanc
ment) were replaced; the gear
indicators were improved. Th
the rods themselves are in perf

3.5. Steam generator

By June 1983 the total ope
generator (fig. 6) was 93000

High Temperature Gas Cooled Reactors - Defining Features

- Fuel
  - Ceramic coated particles of oxide or carbide fuel
    - BISO coated particles
    - TRISO coated particles since early 1980s
  - TRISO particles retain fission products up to 1600°C
- Moderator, Core Structure, Reflectors
  - Graphite - large heat capacity, sublimes at 3800°C
- Coolant
  - Helium - chemically and neutronically inert
  - Core average outlet temperatures up to 950°C

History of HTGR Operation (1)

- Dragon - United Kingdom, 1966-75
  - Block type, 20 MWt, 750°C Outlet
- AVR - West Germany, 1967-88
  - Pebble bed, 46 MWt (15 MWe), 950°C Outlet
- Peach Bottom 1 - United States, 1967-74
  - Block type, 115 MWt (40 MWe), 725°C Outlet
- Fort St. Vrain - United States, 1976-89
  - Block type, 840 MWt (330 MWe), 785°C Outlet
- THTR - West Germany, 1985-89
  - Pebble bed, 750 MWt (300 MWe), 750°C Outlet

History of HTGR Operation (2)

- HTTR - Japan, 1998-present
  - Block type, 30 MWt, 850-950°C Outlet
- HTR-10 - China, 2000-present
  - Pebble bed, 10 MWt, 700°C Outlet

To build up the specific configuration of the initial core, a special loading facility was developed. Using this facility, the spherical elements were loaded by gravity into the core from the 48-m platform above the prestressed concrete reactor vessel. After passing through headers of loading pipes for the different types of spherical elements, the elements passed through release valves, pneumatic decelerators, and a telescopic pipe into the carousel (Figs. 1 and 2), which distributed them to the 15 individual loading pipes (3 for the inner core zone).

In the first step of the nuclear commissioning program, the core was filled with elements to such a level as to reach criticality with no control rods inserted. During the loading process, the inner and outer core

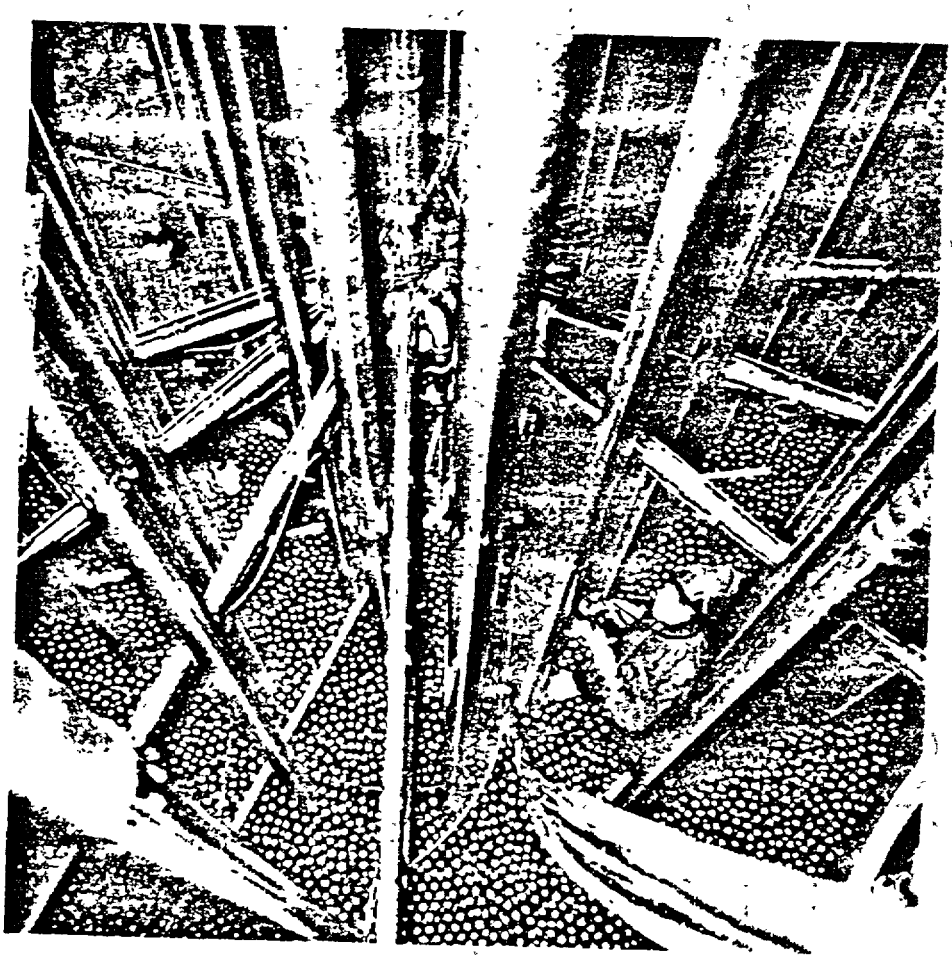


Fig. 1. View of THTR pebble bed during loading, showing core rods, carousel switch, and loading tubes.

of 0.62 was obtained. This s

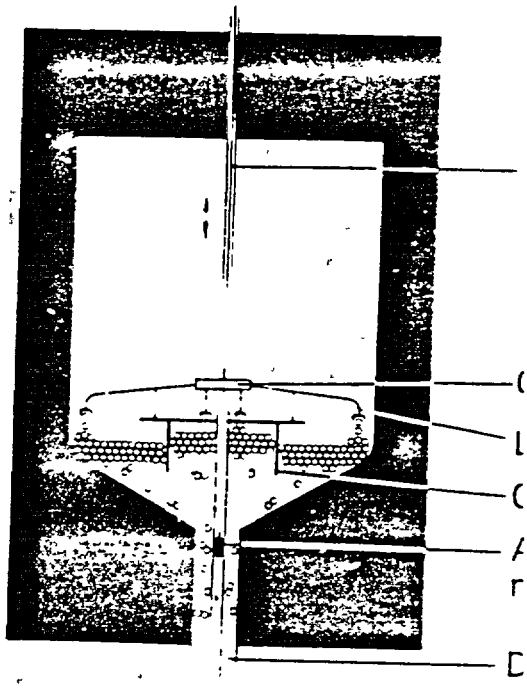


Fig. 2. Schematic vie

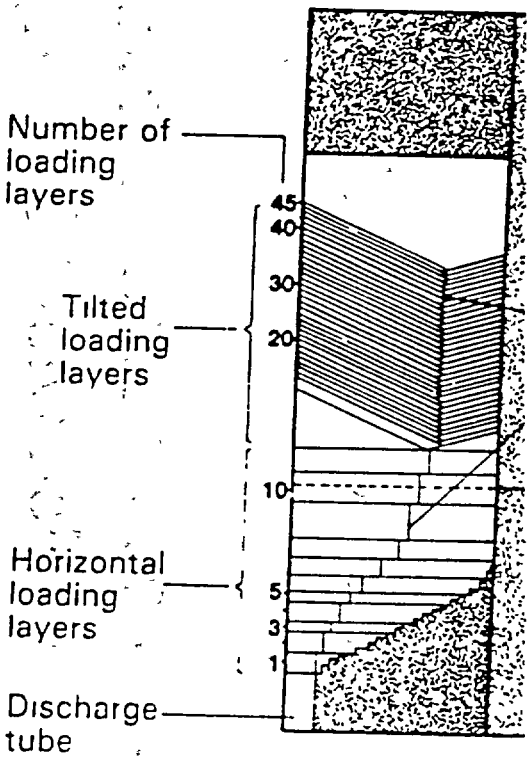


Fig. 3. Loading of THT

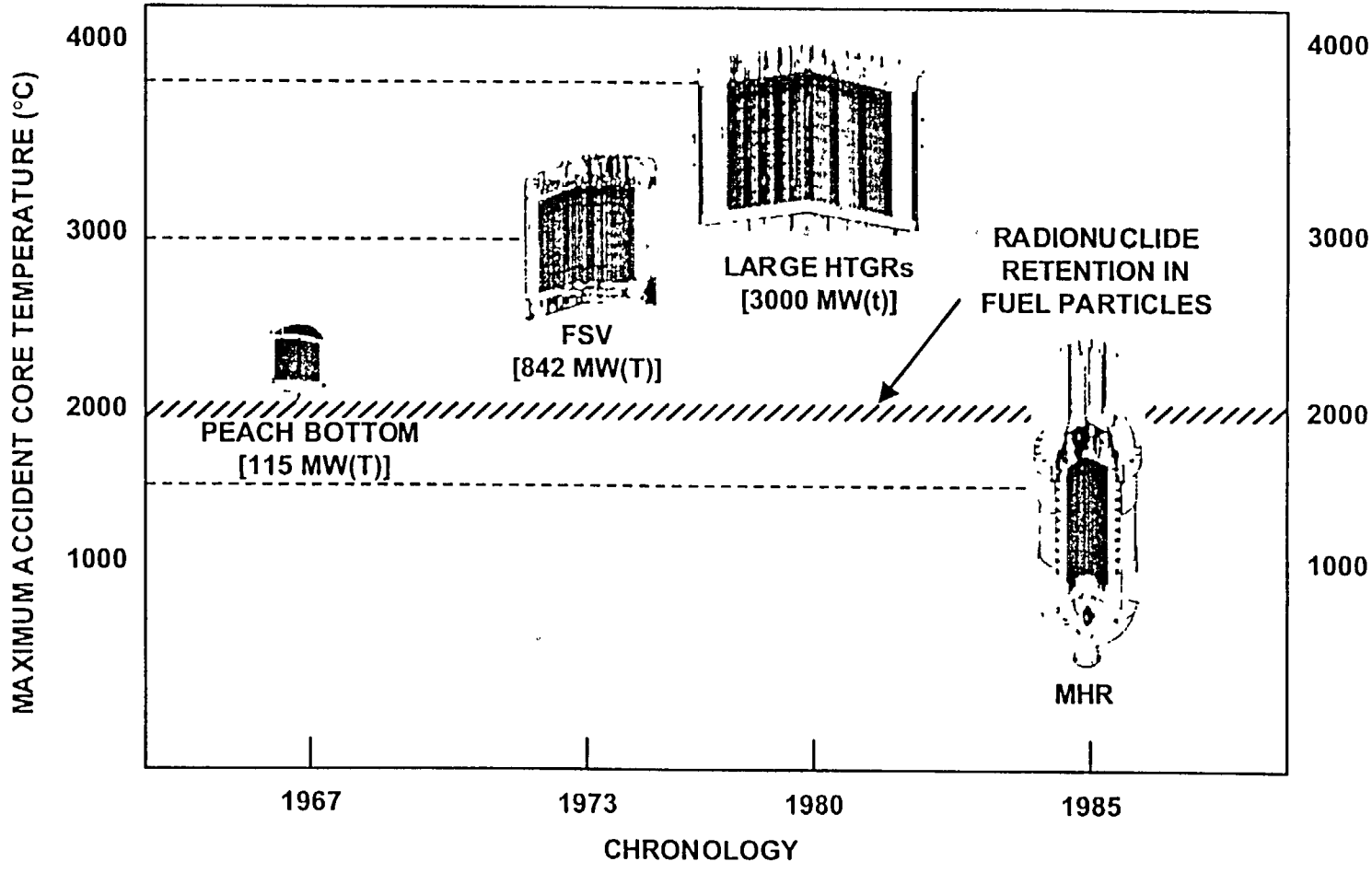
New U.S. Interest in Modular HTGRs

- Pebble Bed Modular Reactors
  - Exelon with ESKOM, IDC, and BNFL: PBMR
    - Demonstration module planned in South Africa
    - NRC pre-application review started
  - MIT & INEEL - MPBR
    - Proposing a "license by test" reactor in Idaho
- Gas Turbine - Modular Helium Reactor
  - DOE/GA: Russian Pu-burning GT-MHR
  - General Atomics: LEU GT-MHR
    - Request for NRC pre-application review

Other HTGR Design Efforts

- Japan's Helium Turbine HTGR Designs
  - 300 MWt Annular Core Pebble Bed HTGR
    - Electricity, steam co-generation, water desalination
  - 600 MWt Annular Core Prismatic Fuel HTGR
- Netherlands' ACACIA Plant
  - 40 MWt Pebble Bed HTGR with Helium Turbine
    - Electricity, steam co-generation

MODULAR HELIUM REACTOR REPRESENTS A FUNDAMENTAL CHANGE IN REACTOR DESIGN AND SAFETY PHILOSOPHY



...SIZED AND CONFIGURED TO TOLERATE EVEN A SEVERE ACCIDENT

GENERAL ATOMICS

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Modular HTGR Design Concept

- Smaller HTGRs designed to withstand depressurized loss of forced cooling (DLOFC)
  - Passive "Conduction Cooldown" through vessel wall
  - Maximum fuel temperature < 1600°C
- Key Modular HTGR Design Features
  - Lower power density, high thermal capacity
  - Tall, slender core and vessel
  - Annular core in some designs
- Typical Module Output
  - Prismatic block: 600 MWt (250 MWe)
  - Pebble bed: 200-270 MWt (85-110 MWe)

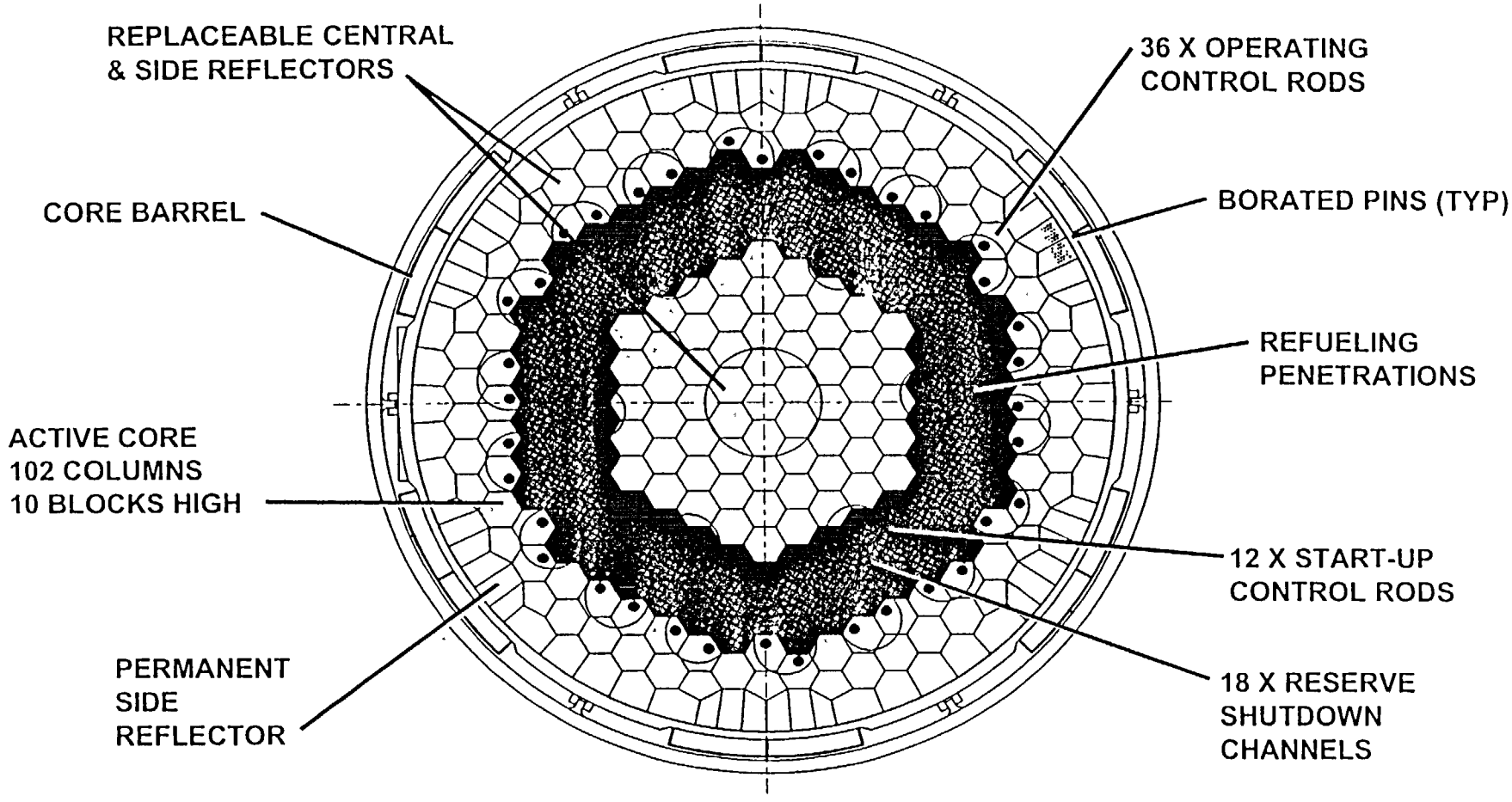
Key MHTGR Safety Goals/Concepts

- Safety does not depend on presence of coolant
  - Passive heat removal prevents fuel damage in depressurized LOFC events
- Early insertion of control or shutdown rods not required in accident scenarios
  - Passive shutdown thru negative temperature feedback
- No inherent mechanisms for runaway reactivity excursions or rapid power transients
- Severe air ingress leading to "graphite fire" effectively precluded by design

Past Modular HTGR Design Efforts

- HTR-Module - West Germany, 1979-90
  - Pebble bed, 85 MWe, Steam cycle, Process heat
- DOE/General Atomics MHTGR, 1985-95
  - Block type, 250 MWe, Steam cycle
    - NUREG-1338, 1989 (Jerry Wilson, Tom King, Pete Williams)
    - Draft update to NUREG-1338, 1995 (Jack Donohew, NRR)
  - 1994 Changed to Helium turbine, 300 MWe GT-MHR
- DOE/CEGA New Production MHTGR, 1989-92
  - HEU with tritium producing Li targets, plus containment.

ANNULAR REACTOR CORE LIMITS FUEL TEMPERATURE DURING ACCIDENTS



. . . ANNULAR CORE USES EXISTING TECHNOLOGY

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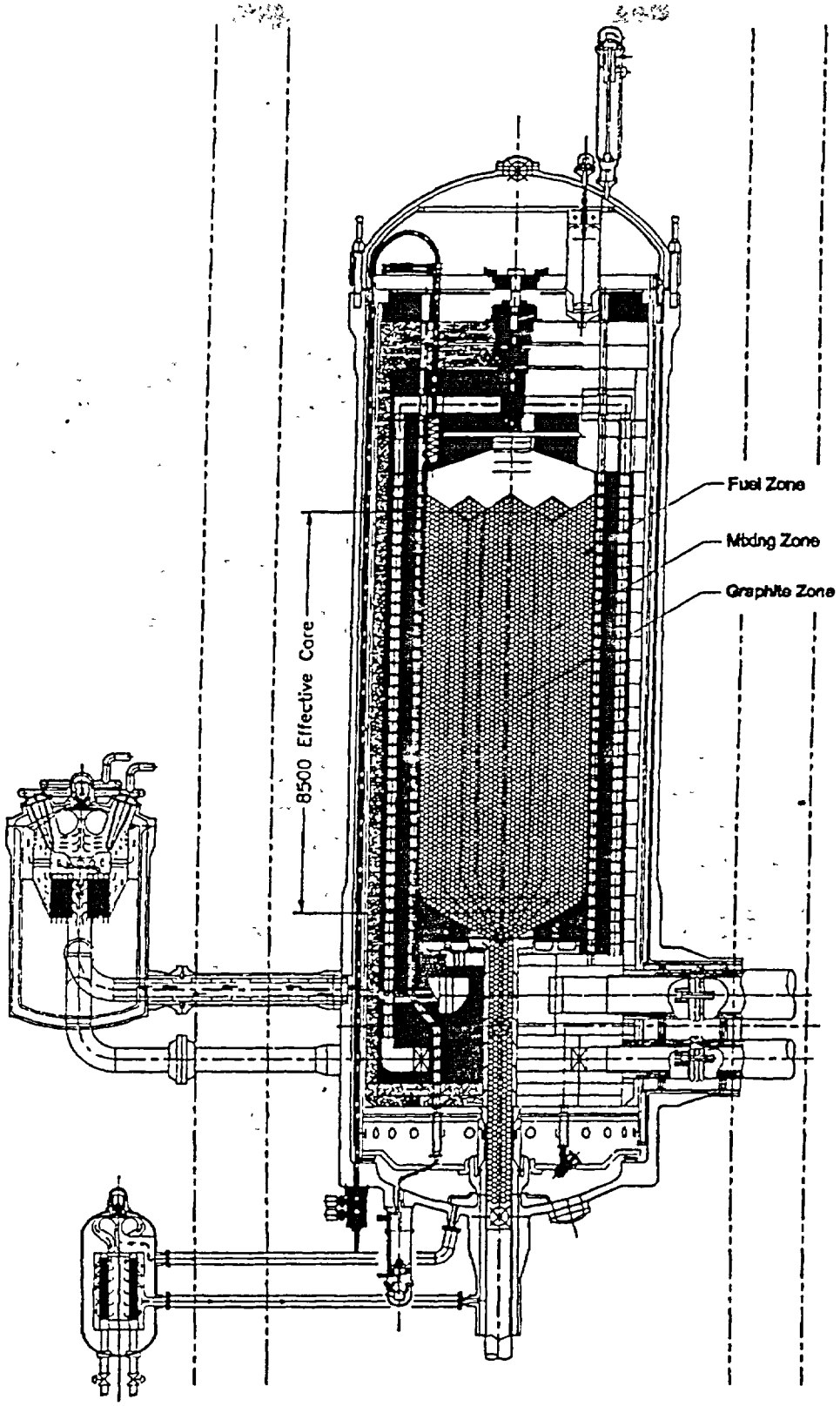
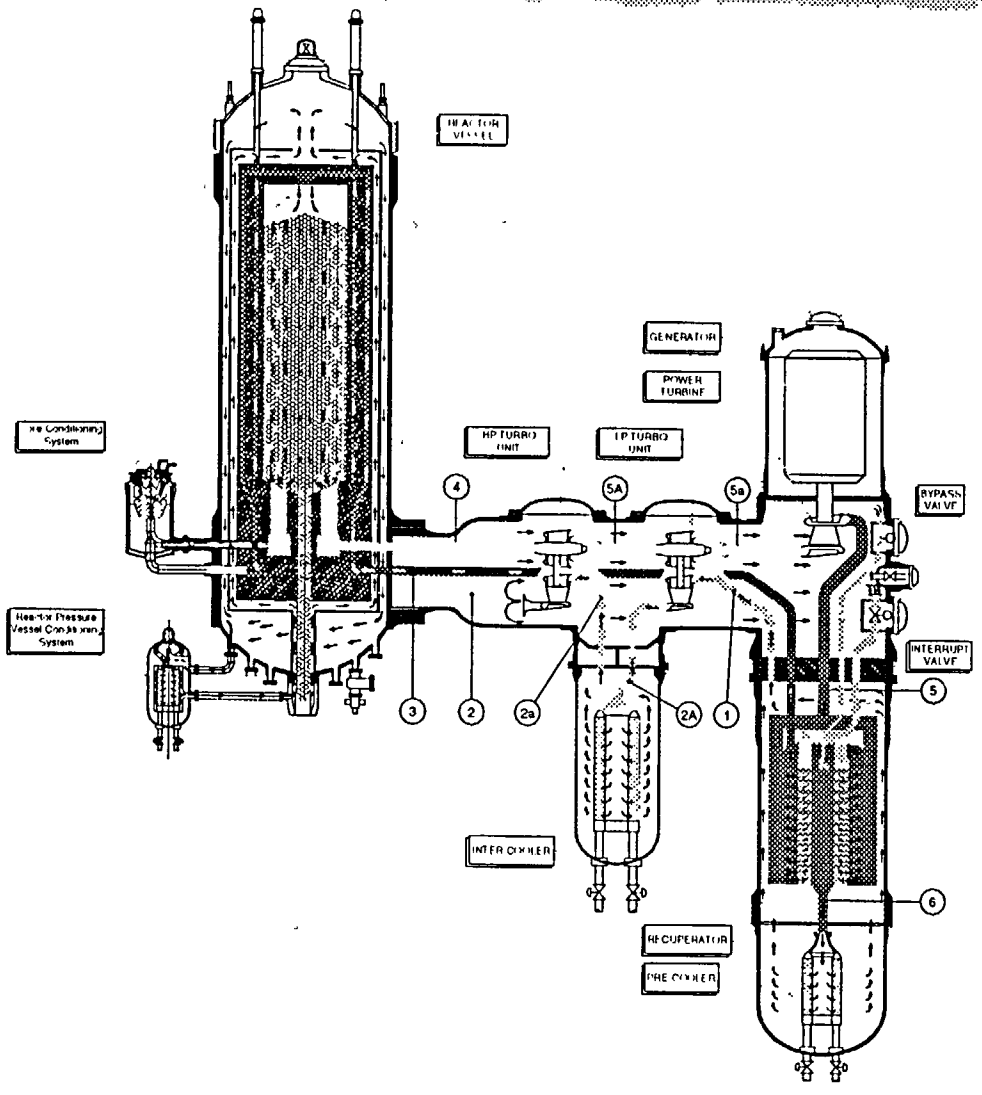


FIG. 3.6 Reactor layout.

Fig. 2 Main Power System



Moderator: Graphite versus Water

- Average Collisions to Thermaliztion
  - Water: 18
  - Graphite: 114
- Slowing Down Power (/cm)
  - Water: 1.5
  - Graphite: 0.06
- Migration Length (cm)
  - Water: 5.8
  - Graphite: 62
- HTGR is "neutronically" much smaller than LWR of similar dimensions

As nuclear data library the 43 group MUPO-Library 5 [3] has been used in DITO [4] with updates from ORIGEN [5] arriving now at a total of 77 isotopes. As an alternative it is planned to integrate the WIMS-D Library [6] into the system. First calculations indicate that predictions with MUPO and WIMS-Libraries show a different sensitivity to spectrum and leakage effects during buckling iterations.

One important point in the HTR model describes the movement of the fuel elements in nine channels of near cylindrical shell shape (Fig. 5). This mesh net is generated in FLIMO and transformed in MAGI to r-z-phi geometry. It contains 139 burnup regions where the fuel reshuffling and recycling has to take place. The burn up module has to be fed with one group fluxes obtained from the four group flux solution from the CITATION code [7]. CITATION solves the 3 dimensional model diffusion equation. Fig. 6 and 7 show axial neutron fluxes and radial temperatures for two different reloading stages. A schematic diagram of the two main blocks of HTR-2000 are shown in Fig. 8 with the buckling iteration between spectrum and diffusion calculation.

To follow selected fuel elements within the reactor during its burn up and movement in the core as realistically as possible a further development beyond the 77 isotopes of HTR-2000 has been undertaken starting from the Isotope Generation and Depletion Code ORIGEN [5]. This code has been named HTROGEN. It can solve the burn up equations for 1156 nuclides including such transmutations as

(n, α)-, (n, γ)-, (n, p)-, (n, 2n)- and (n, 3n)-processes.

The code works with three groups of neutron energies and has to be fitted to give the same reaction rates as the four group picture in HTR-2000 (Fig. 9).

For licensing purposes the KFA is obliged to use HTROGEN under safeguard aspects to calculate all fuel inventories and other radioactive materials.

In an earlier test HTR-2000 and HTROGEN were used sucessfully to follow the reactor operation since September 1982 up to the shutdown in 1988 [8]. It provides results for gamma-activities, shut down heat and the build-up of heavy metal isotopes beyond plutonium.

TOMKU

For the determination of the effective potential scatter cross sections in HTRs with spherical fuel elements of different types the program TOMKU was written. According to the reactor model of the AVR 9 different fuel types in 10 burn up groups are catered for covering 25 spectral zones. Separate calculations are performed for the Thorium 232 and Uranium 238 resonances.

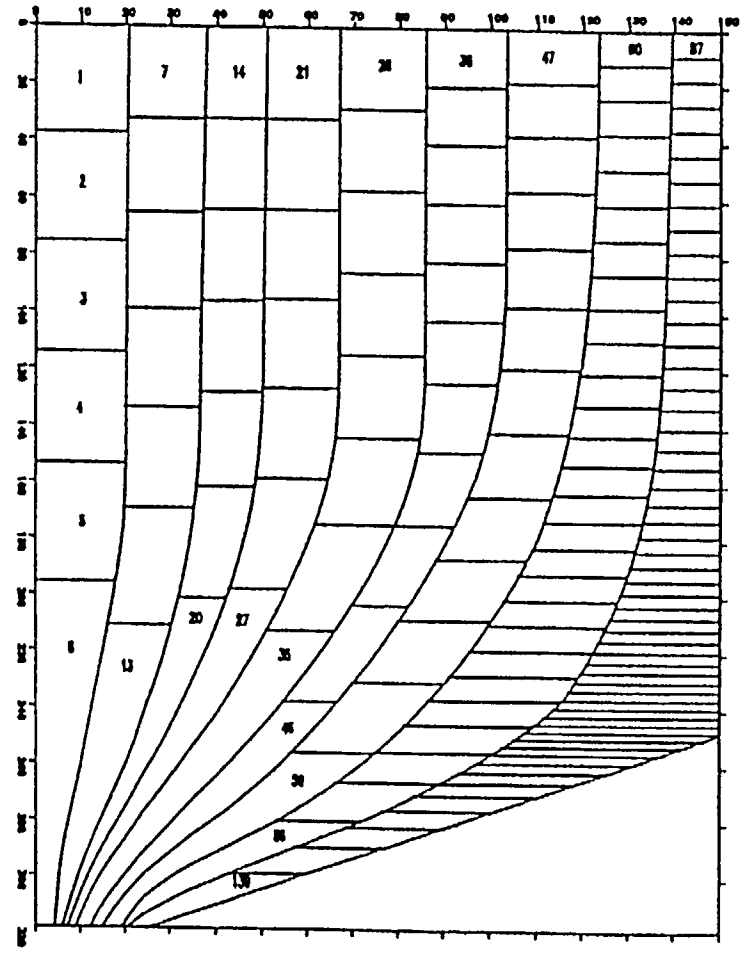
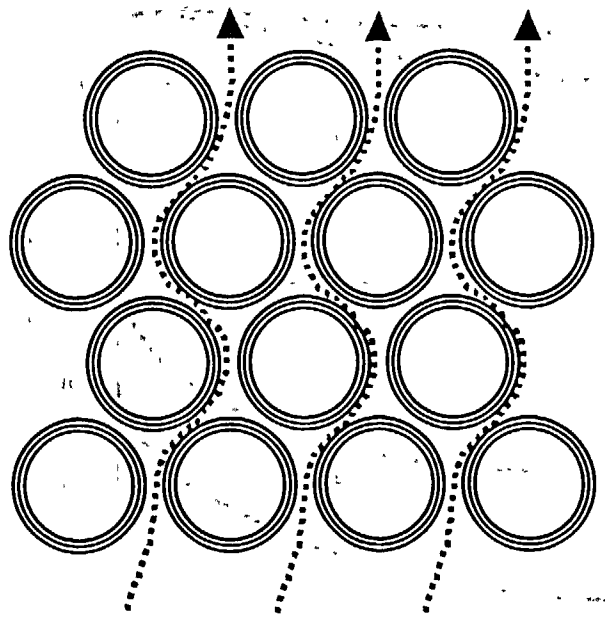
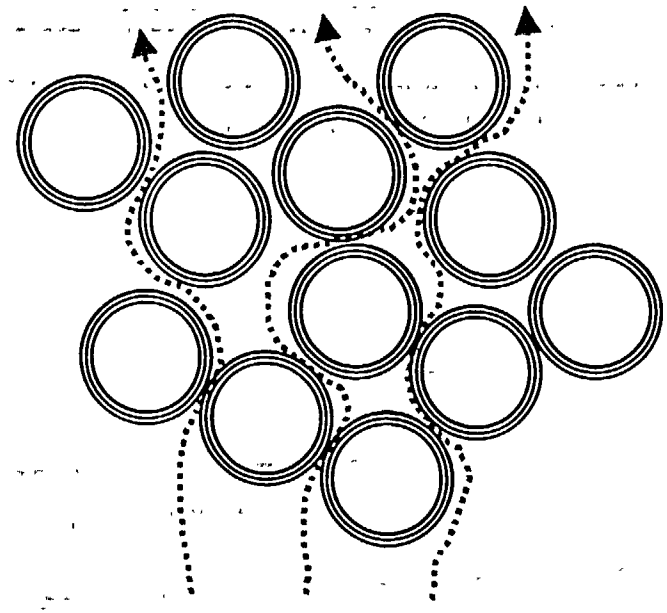


FIG 5.

Vessel Model Development and Integration



Uniform bed configuration and flow distribution

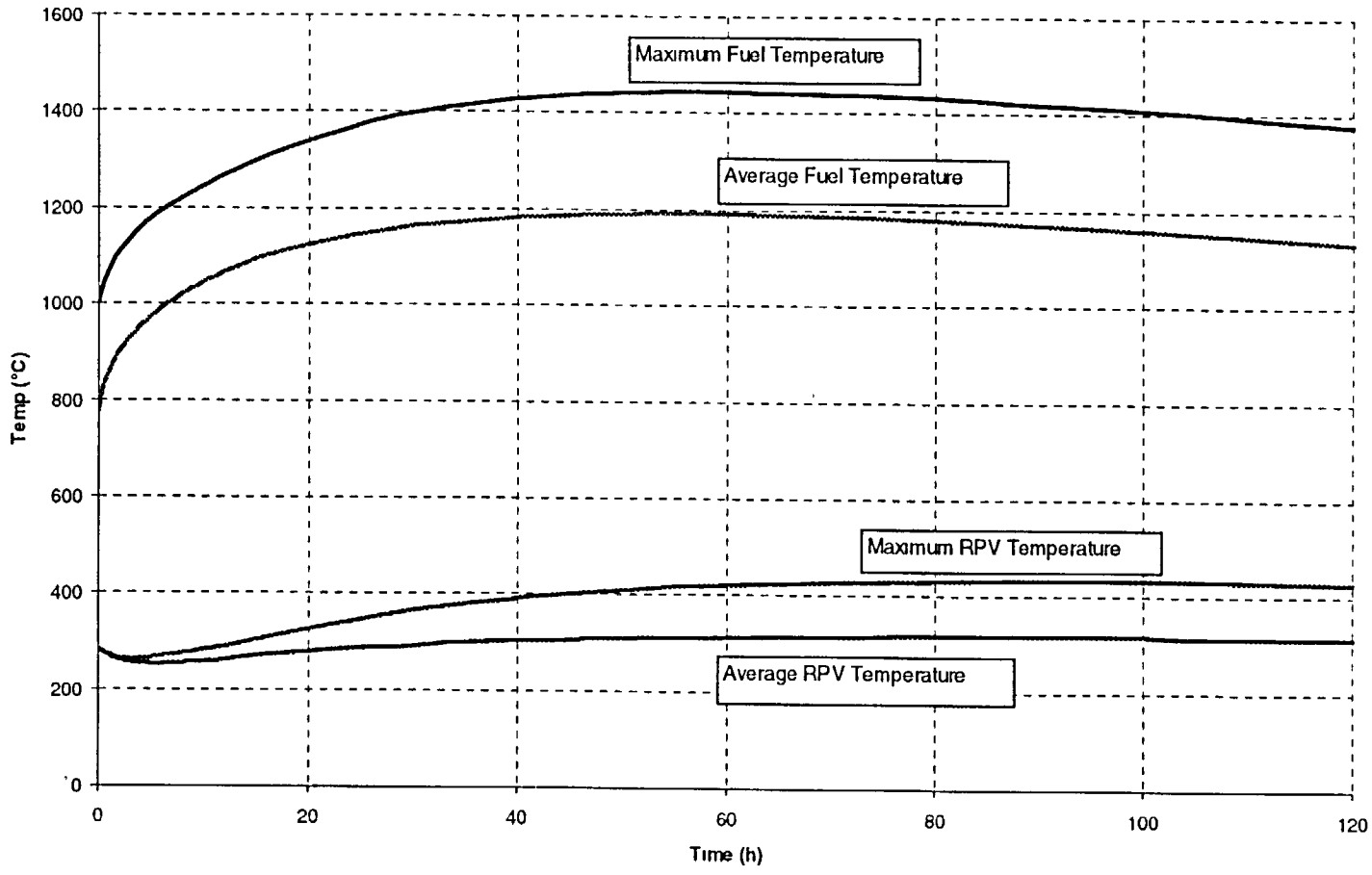


Non-uniform bed configuration and flow distribution

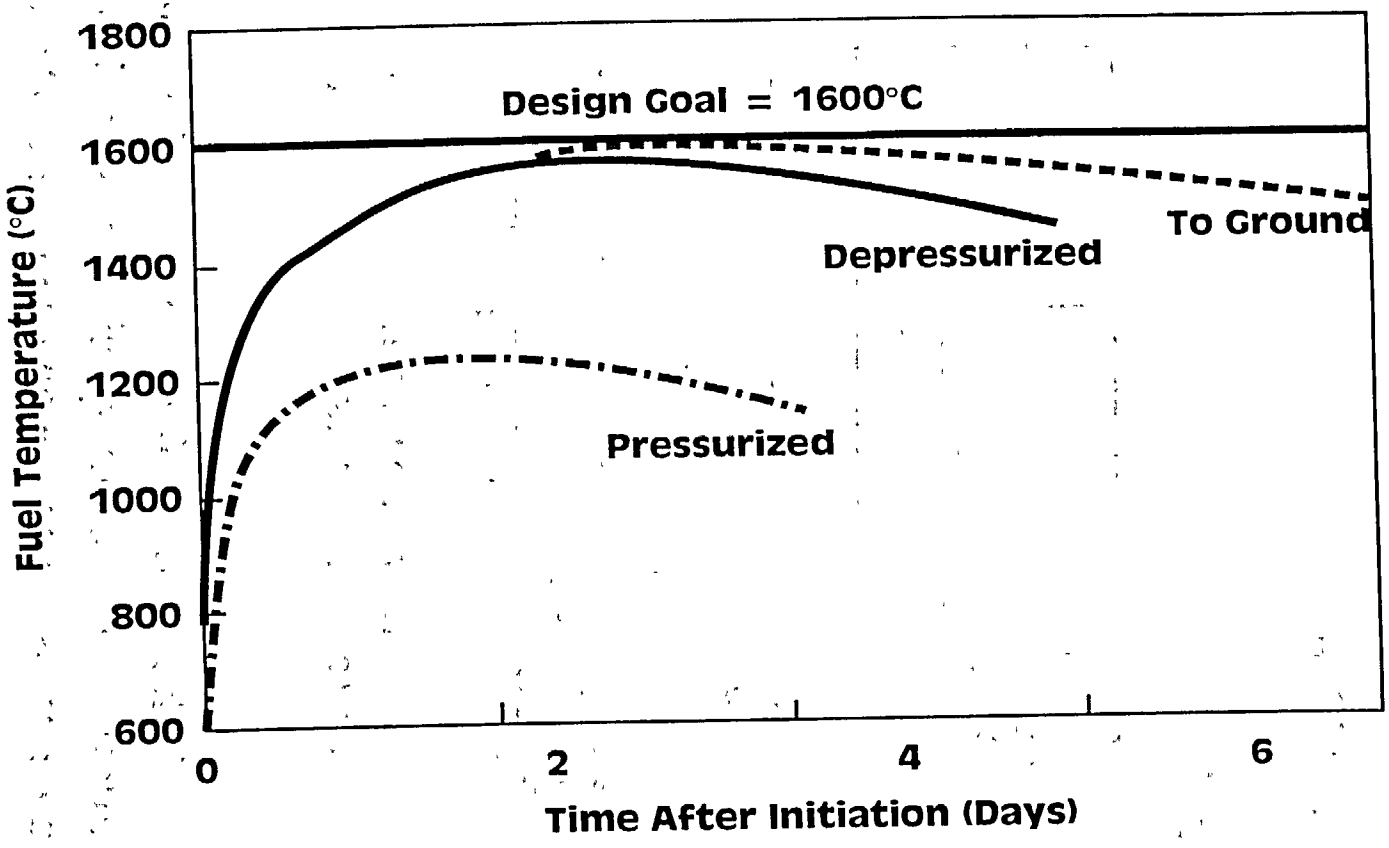
- Real flow is 3-dimensional. With eddies for separation
- For steady state simplification may be possible.
- Use of quasi-steady closure relationships for accidents and non-uniform configurations has not been verified

Fig. 6 Loss of Coolant Event

265 MW PBMR Ref. Core: Temperature Distribution during a DLOFC



FUEL TEMPERATURES REMAIN BELOW DESIGN LIMITS DURING LOSS OF COOLING EVENTS



. . . PASSIVE DESIGN FEATURES ENSURE FUEL REMAINS BELOW 1600°C

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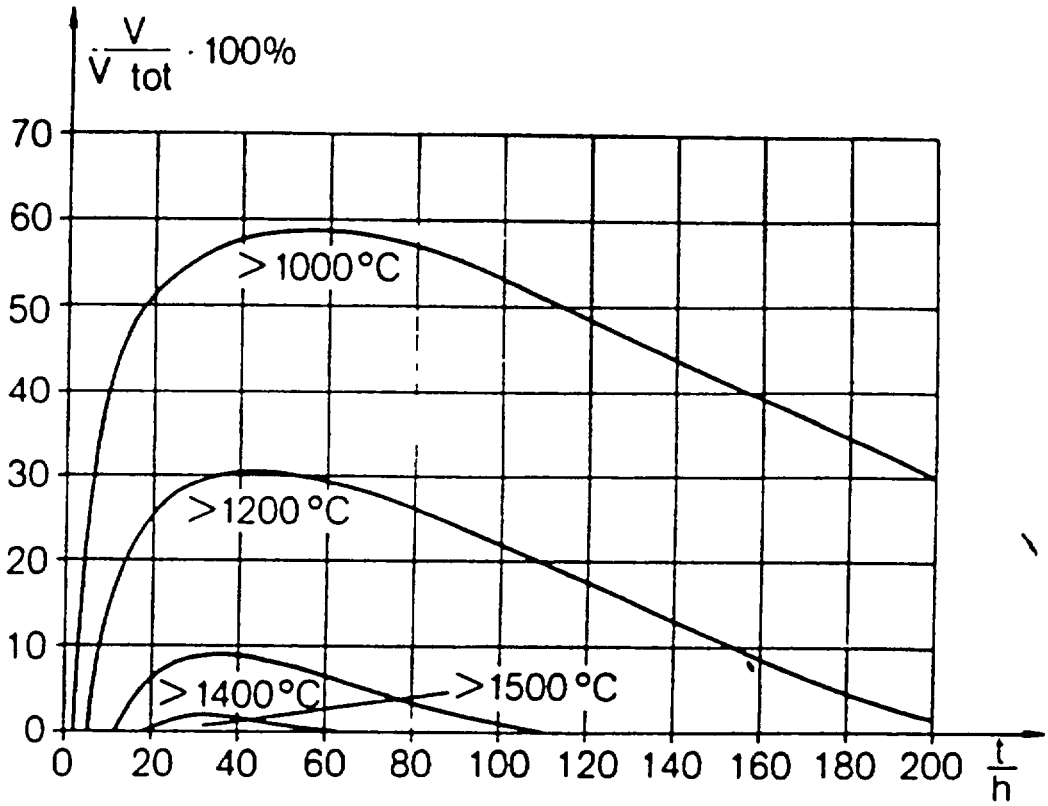
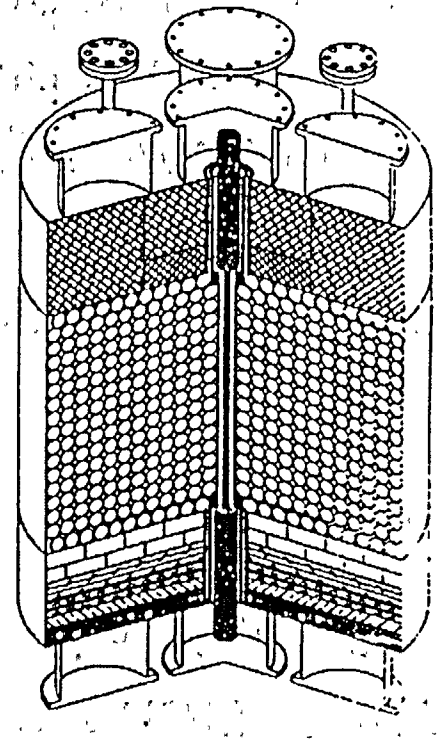


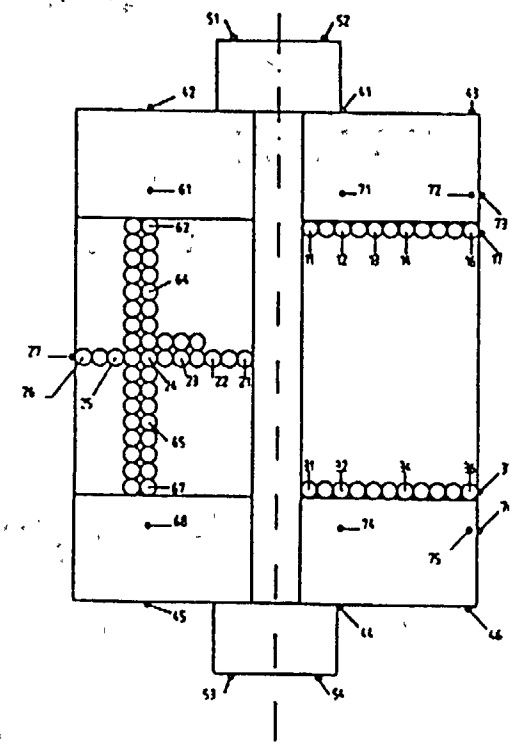
Fig. 3. Time-dependent fraction of fuel element for various temperatures (depressurization with core heat-up).

Rupture of the line is followed by the complete depressurization of the primary circuit until it has reached the same pressure as the environment after several minutes. In order to prevent an inadmissibly high internal pressure in the reactor building, the primary coolant is discharged to the environment via pressure relief openings with flaps in the reactor building.

The environmental exposure caused by the radioac-

TRAC Bench Marking: SANA Test Data





Fuel is Key to MHTGR Safety Case

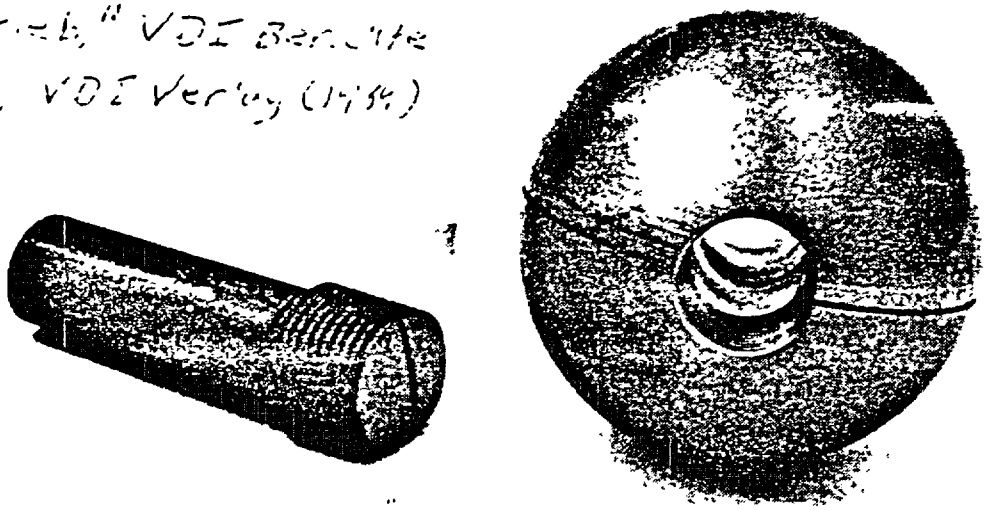
- Fuel Fabrication Quality
  - Low fraction of initially defective particle coatings
  - Low U contamination outside coatings

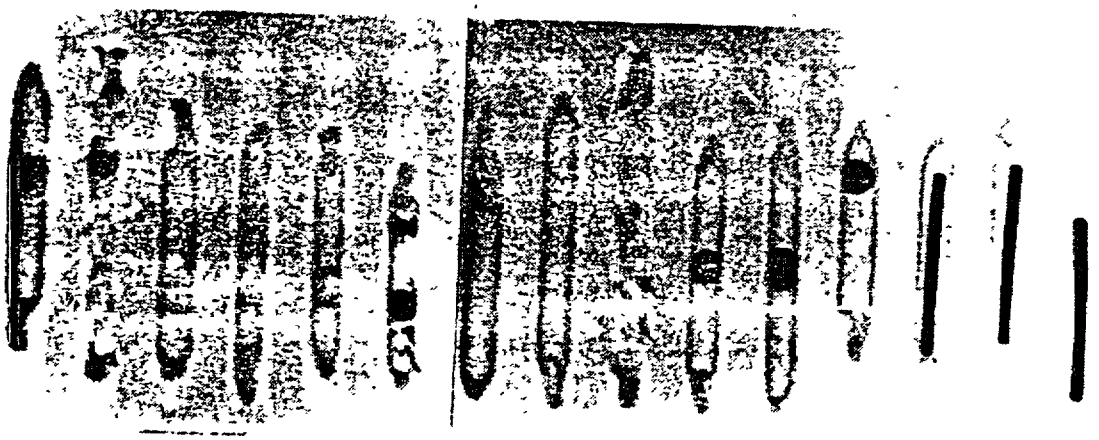
- Fuel Performance
  - Fission product retention in intact coated particles
  - Very low coating failure and release rates
    - During irradiation
    - During maximum heatup accidents

TRISO Coated Fuel Particles

- Uranium oxide kernel: 0.500 mm dia
- Porous carbon buffer layer: 0.095 mm
  - Serves as fission gas plenum
  - Isolates kernel from gas-tight coatings
- Inner high density pyrocarbon layer: 0.040 mm
  - Provides foundation for silicon carbide
- Silicon carbide layer: 0.035 mm
  - Contains fuel and fission products
- Outer high density pyrocarbon layer: 0.035 mm
  - Protects silicon carbide

Aus "AVR-Anlage Betrieb," VDI Berichte 721, VDI Verlag (1984)





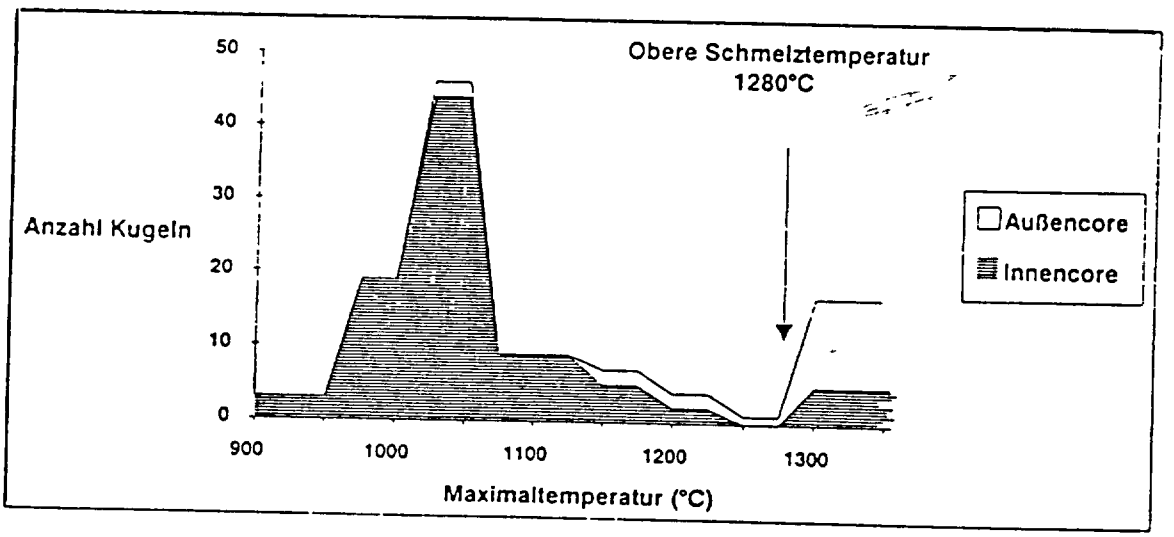


Abb. 6: Temperaturmeßkugeln (oben) mit 20 Schmelzdrähten (Aussc aus Röntgenaufnahme, Mitte) zeigten teilweise unerwarte hohe Temperaturen (unten).

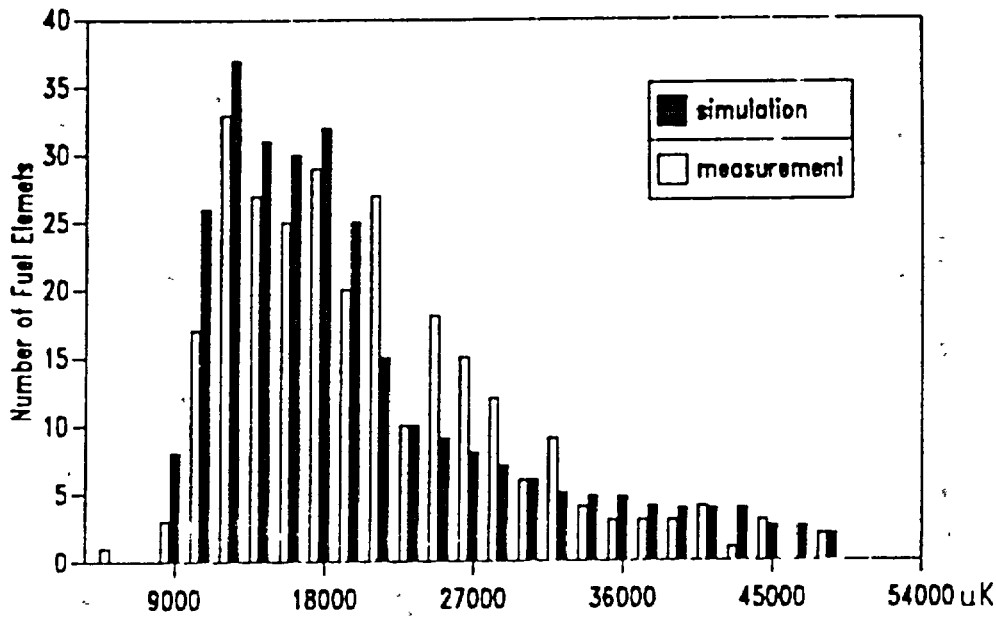


Fig. 2. Typical AVR inner core residence spectrum.

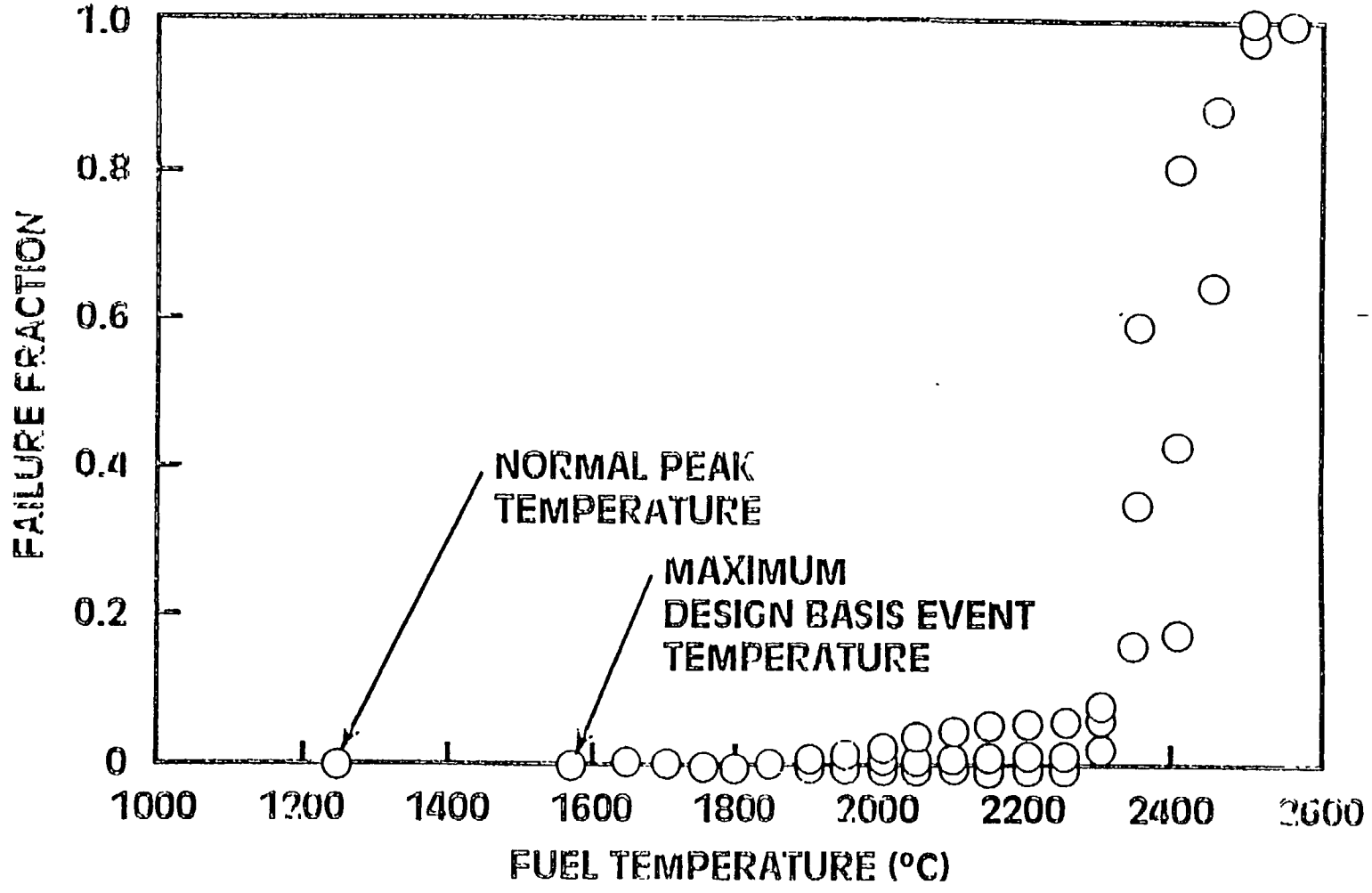
Other approved code systems may be linked to the MOCCA system by using adaptive pre- and post-processor modules.

The MOCCA system was extensively used in the design calculations for the HTR-100, HTR-500 and GHR.

2.2. System experience and verification

The quality of the theoretical models and the numerical algorithms as they are incorporated into the reactor modelling systems has to be tested in different ways. For nuclear data and basic neutronics, cold critical experiments are useful. However, most of the uncertainties, especially concerning safety-related questions,

COATED PARTICLES STABLE TO BEYOND MAXIMUM ACCIDENT TEMPERATURES



GENERAL ATOMICS

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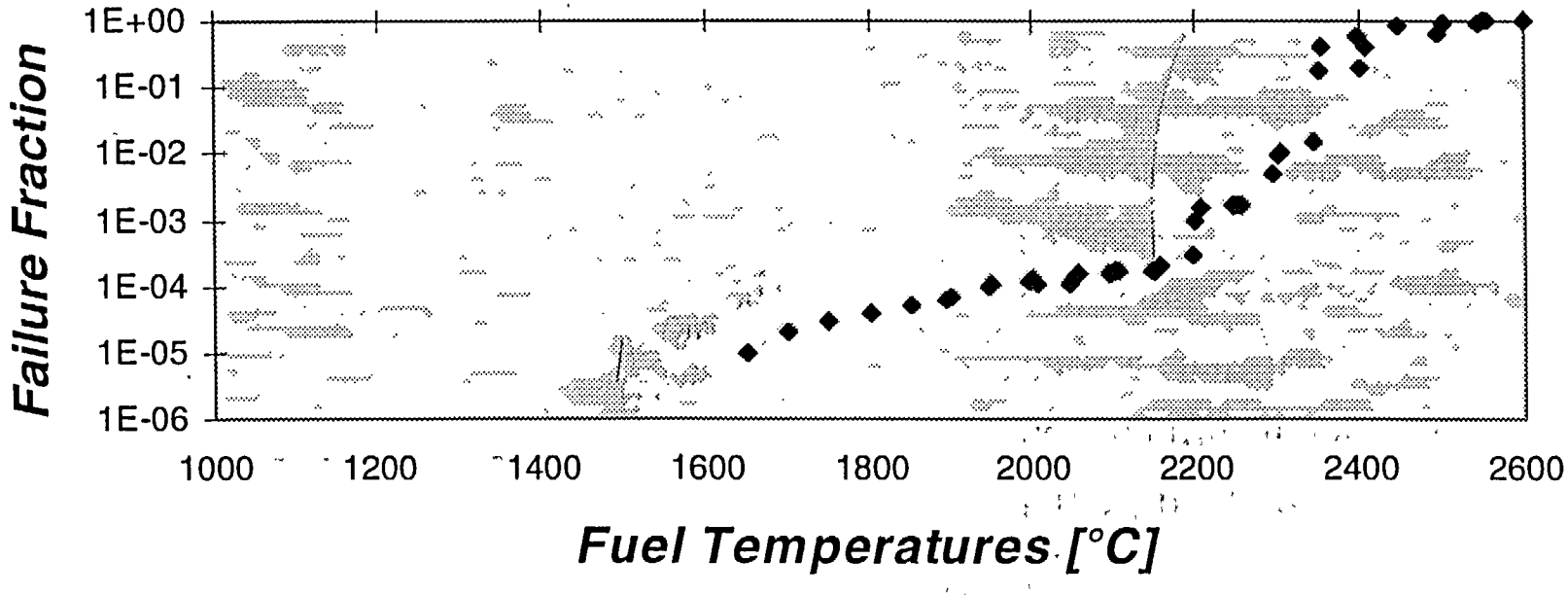
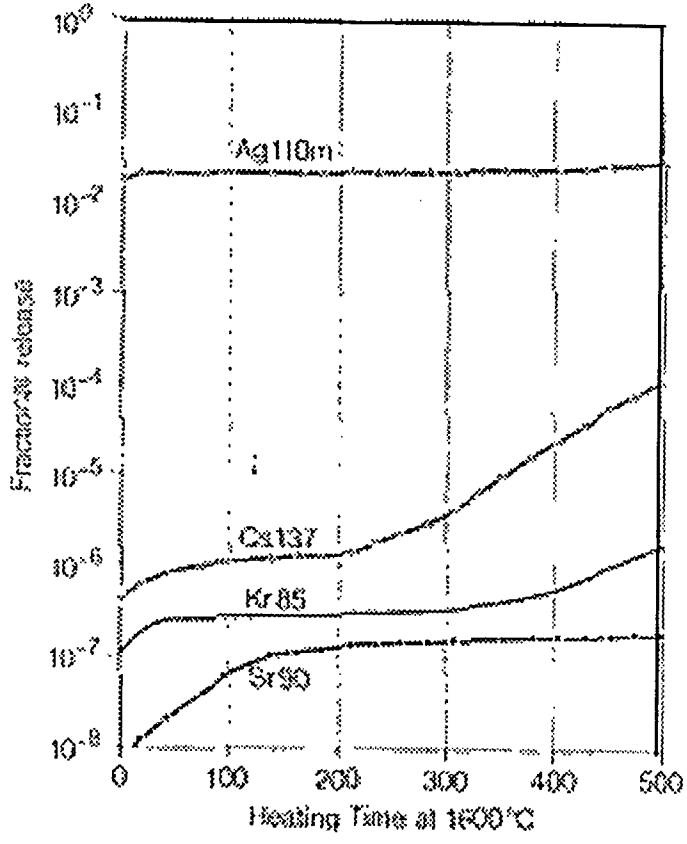
Fig. 5 Fuel Performance



A. Kadak (1998)

Fractional Fission Product Release



- ^{110m}Ag
  - largest fractional release
  - heating time independent
  - 250 day half-life
  - plates out on colder graphite surface
  - greatest migration through silicon carbide layer
- ^{137}Cs
  - time/temperature dependent
  - fractional release significant after breakdown of silicon carbide layer
- Maximum additional fractional release due to core heat-up: ~ 10^{-5}

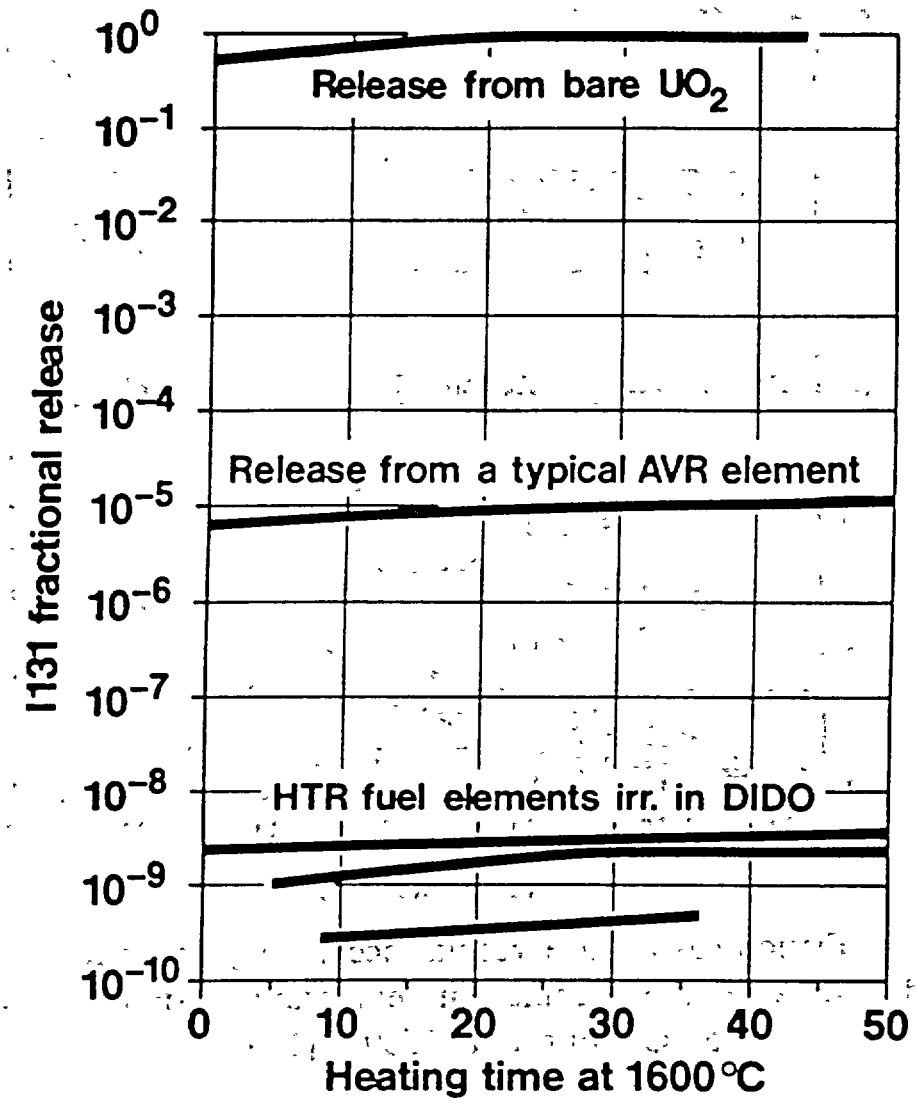


Fig. 9. Iodine release during heating at 1600 °C. Bare UO_2, as is used in most reactor systems, releases all iodine in the fuel after 15 h. The coated particles used in the HTR retain iodine completely up to 1600 °C. The heating tests of AVR-irradiated fuel elements show the level of cross-contamination from old fuel element types with high levels of heavy metal contamination.

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1600 °C
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exposec
AVR a
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which
^{131}I was
shown
10 h a
35 °C p
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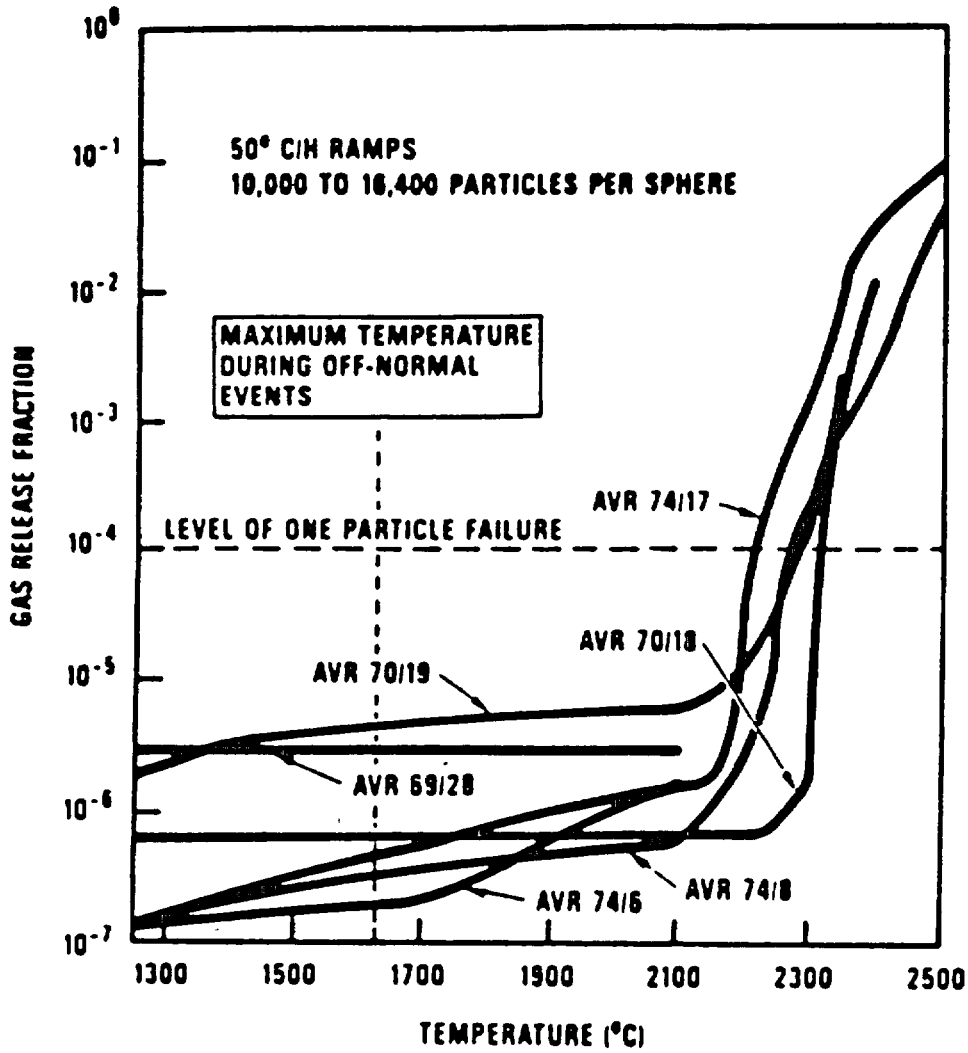
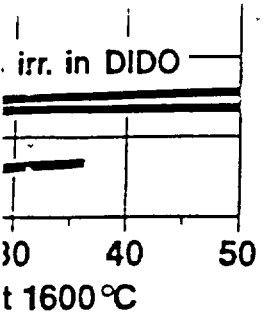


Fig. 2. Fission-gas-release results accompanying heat-up of irradiated AVR fuel pebbles in temperature-ramp tests of 50°C/hour up to 2500°C [1,2].

fuel temperature, up to 2500°C. Radiologically, the most significant nuclide is iodine, and it is released as a gas from MHTGRs. It has been shown experimentally that krypton release from fuel particles is a conservative measure of iodine release; since krypton release is easier to measure, it is measured to indicate iodine release. FRG measured krypton fission-gas releases



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demonstrates the near-identical xenon, krypton and iodine release [18].

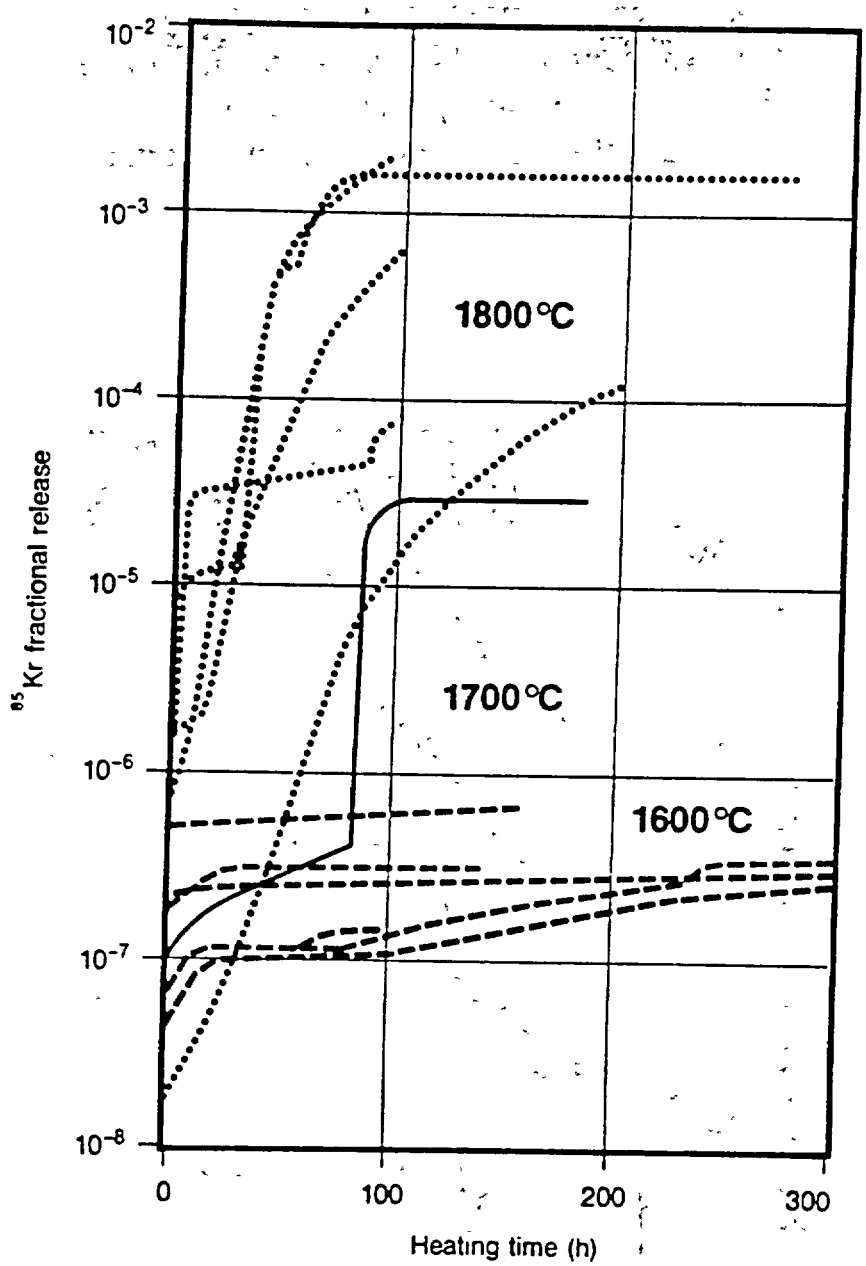

Fig. 10. Krypton release during heating of spherical fuel elements with up to 12% FIMA burnup at 1600–1800 °C. This diagram demonstrates that the ample fission product retention margin is available up to 1600 °C with fuels irradiated under MODUL-like irradiation conditions.

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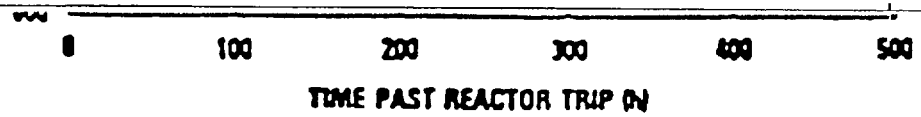


Fig. 4. Typical temperature transient following a loss-of-coolant accident in a high-performance MHTGR, with the passive heat removal system operational (temperatures includes the maximum expected uncertainties) [4].

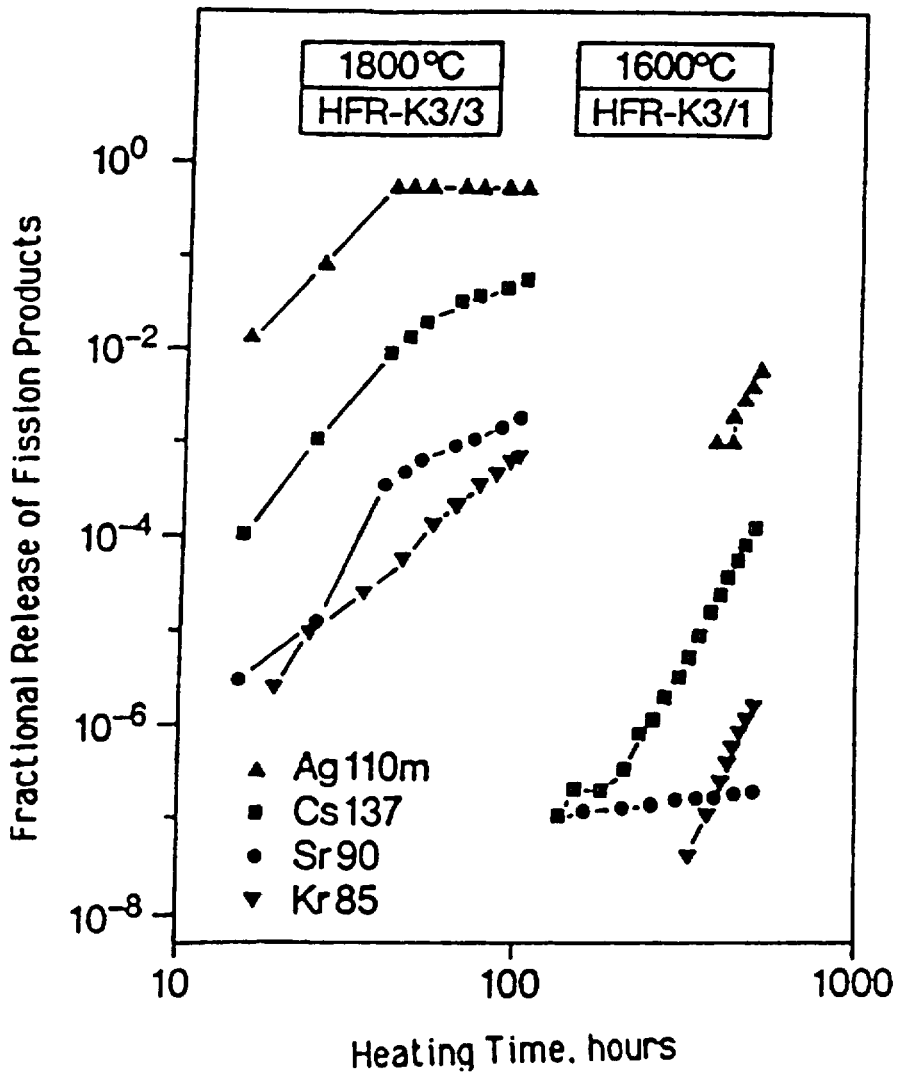


Fig. 5. Fission product release from typical irradiated HTGR fuel when exposed to constant temperatures for various times at 1600 and 1800°C, respectively [1,3].

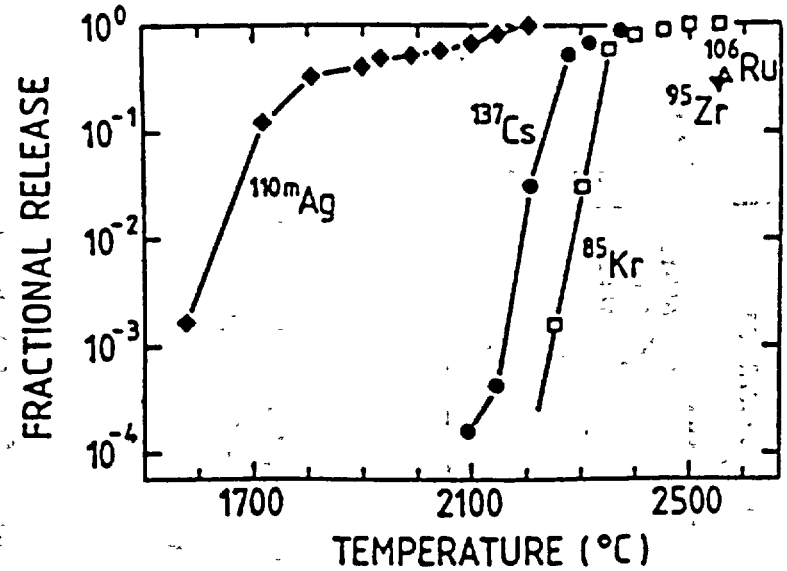
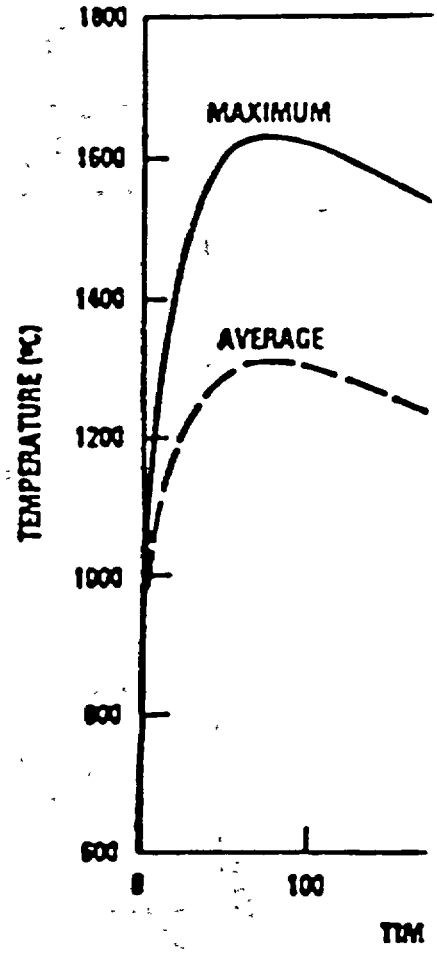


Fig. 3. Fission product release during temperature-ramp heating of TRISO-coated particles [1].

fission gases are being retained within the coated-fuel particles to very high temperatures (~ 2200°C). Figure 3 also shows that silver is not retained well at high temperatures, and diffuses relatively rapidly through SiC and pyrocarbon at temperatures above ~ 1600°C. However, silver has a very low fission yield, and also condenses readily on the "cold" surfaces outside the reactor core; consequently, it does not constitute a radiological hazard to the public. Cesium is much more a radiological hazard, but is essentially retained by fuel coatings up to about 2100°C for tens of hours based on the ramp tests of fig. 3 (these tests were carried out over a period of ~ 25 hours). The krypton release results are typical of fission gas release; the Sr and Ru release values are extremely low except at very high temperatures such as 2500°C.

The maximum fuel temperatures (including uncertainty estimates)in MHTGRs under a loss of coolant accident is illustrated in fig. 4. As shown, the peak temperature of about 1600°C occurs for a period of



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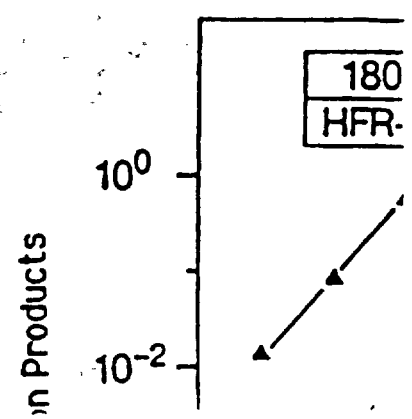

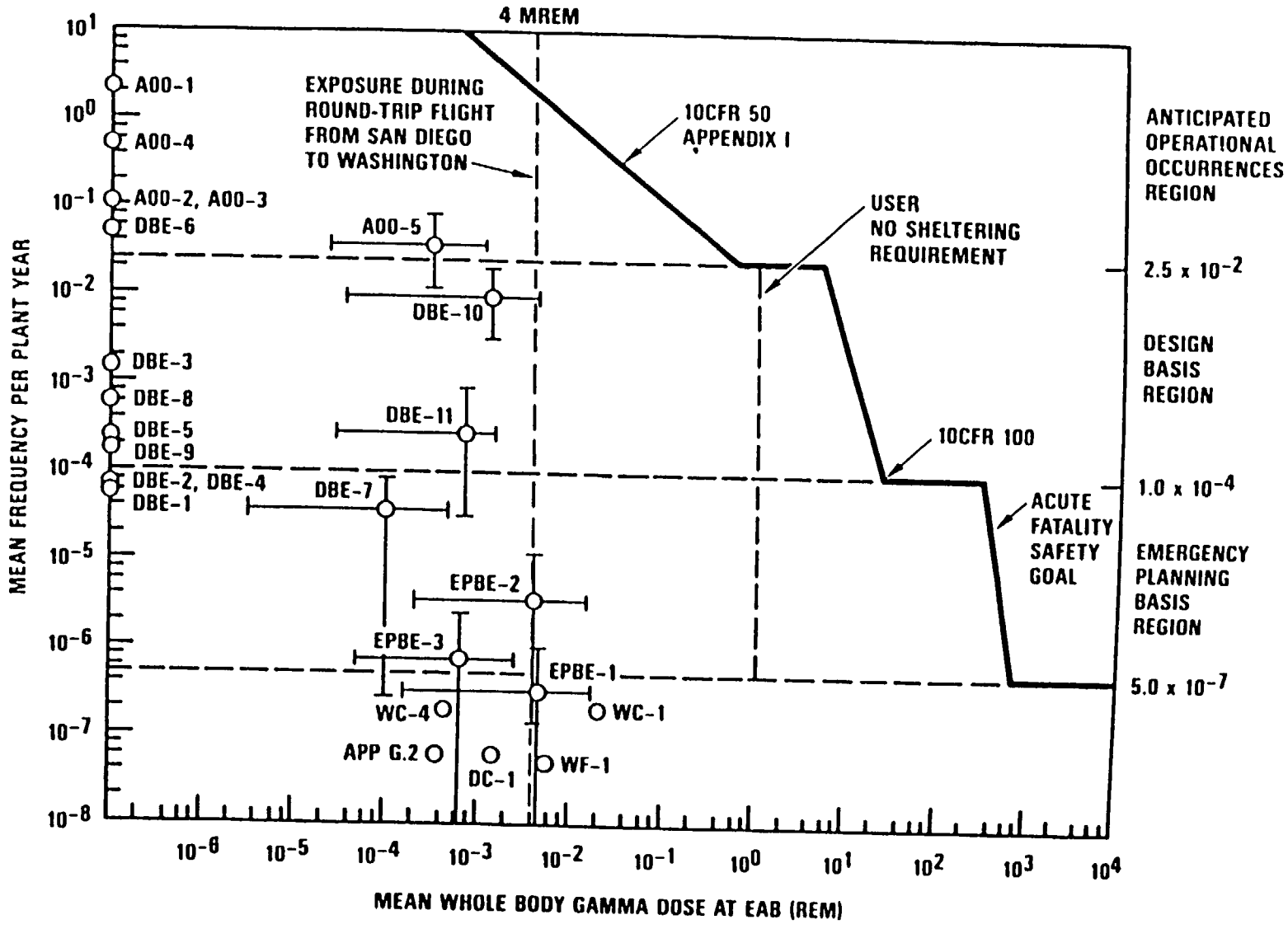

Figure 15.1 Assignment of top-level regulatory criteria and results of safety analysis
Source: DOE, 1986-3

4. LICENSABILITY ISSUES

4.1 Introduction

The staff performs preapplication reviews of an advanced reactor design in part to identify issues that may impede licensing the design. These licensability issues are where the design departs significantly from what NRC has accepted in the past or where changes to the design to resolve a staff concern may fundamentally alter the proposed design. These issues need to be identified at an early stage, so that the designers can address the issues in an application to NRC for design approval: the preliminary design approval (PDA), final design approval (FDA), or standard plan design certification under 10 CFR Part 52.

In this chapter, the staff discusses the licensability issues for the MHTGR design. The identification of these issues was discussed in the previous chapter (Chapter 3). The references in this chapter to the evolutionary light-water reactors (LWRs) and passive advanced LWRs are references to the plants listed in Section 5.1 of this report which have gone through or are going through design approval reviews by the staff.

4.2 MHTGR Licensability Issues

The nine licensability issues for the MHTGR design are as follows:

- Fuel Performance (Section 4.2.1)
- Fission Product Transport Computer Codes (Section 4.2.2)
- Source Term (Section 4.2.3)
- Unconventional Containment (Section 4.2.4)
- Safety Classification and Regulatory Treatment of Non-Safety-Grade Systems (Section 4.2.5)
- Completely Passive System for Ultimate Heat Sink (Section 4.2.6)
- Reactor Vessel Neutron Fluence Embrittlement (Section 4.2.7)
- Reactor Vessel Elevated Temperature Service (Section 4.2.8)
- Applied Technology Designation (Section 4.2.9)

4.2.1 Fuel Performance

The proposed fuel for the MHTGR is the TRISO multicoated microspheres which are discussed in Section 4.2 of the Preliminary Safety Information Document (PSID) ([DOE]-HTGR-86-024). It is essentially the same fuel as that approved for Fort St. Vrain, although the Department of Energy (DOE) has considered additional seal coats on the TRISO structure for the MHTGR. A picture of the fuel, from the DOE presentation of June 4-6, 1991, listed in Section 1.3 of this report, is shown in Figure 4.1. The fuel particles are formed into small, cylindrical compacts in the manufacturing process and the compacts are in large prismatic graphite blocks as shown in Figure 1.2 of this report. Fueled blocks and unfueled, or reflector, prismatic blocks will make up the core inside the reactor pressure vessel.

The PBMR Project in South Africa

- Eskom
  - began investigating in 1993
  - Pre-Feasibility Study in 1995
  - Techo-Economic Study in 1997
  - Engineering Design phase by 1998
  - Construction Activities by mid 2001
  - With Industrial Development Corporation have 50 % of shareholding in the project
- Two International Companies Have Invested
  - PECO Energy = Exelon
  - British Nuclear Fuel (BNF)