

# Pebble Bed Modular Reactor High Temperature Materials Graphite

Mark A Davies  
October 2001

# Purpose of Presentation

- Highlight the safety issues related to the use of graphite technology in High Temperature Reactors
- Identify options that lead to the successful resolution of these issues

# Overview

- PBMR design
  - Functional Requirements
  - Assessment Criteria
- Graphite
  - Manufacture
  - Material Properties
  - Performance Assessment
  - Risk Mitigation
    - MTR Programme
    - Inner Reflector Replacement
- Conclusions

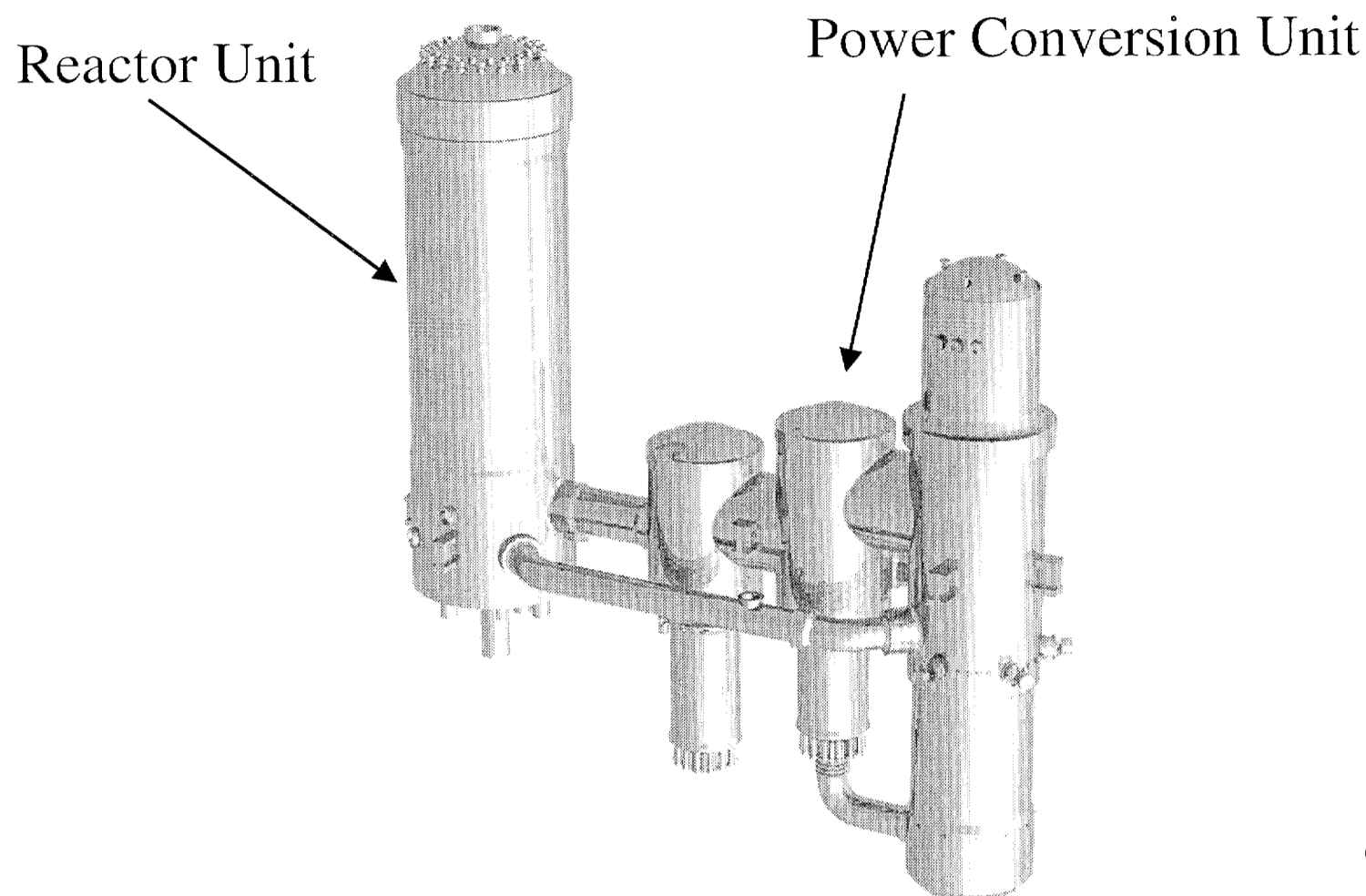
# PBMR Design

- Core Structures Safety Functionality
  - Maintain Pebble Bed (PB) Geometry
  - Maintain Adequate Cooling of the PB under normal and abnormal conditions
  - Maintain Access for the Reactivity Control and Shutdown System (RCSS)
  - Maintain the De-fuelling Path

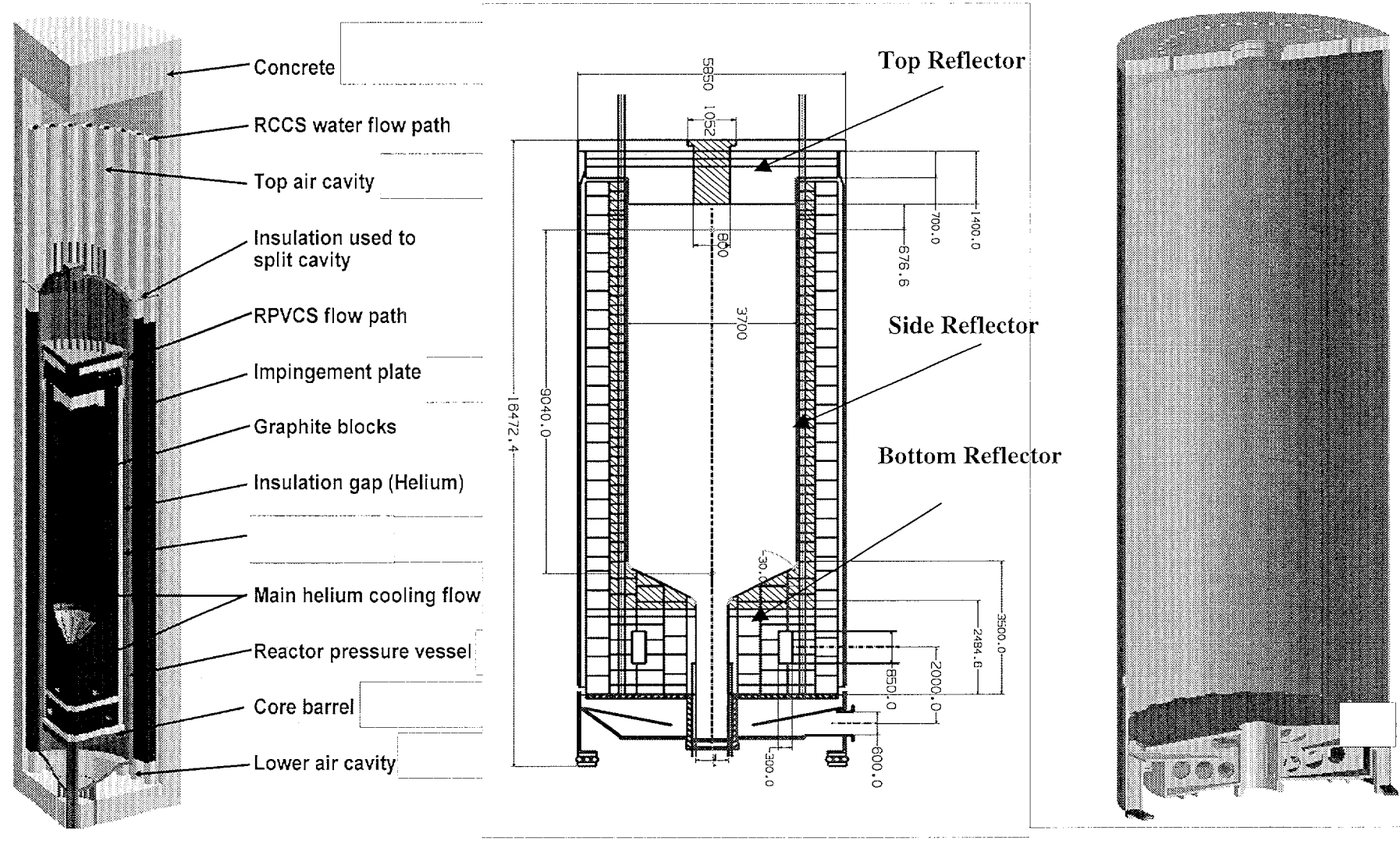
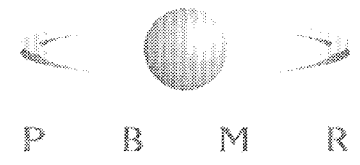
# Performance Assessment Criteria

- Structural
  - Build up of stresses exceeding strength
- Deformation
  - Excessive distortion of reflector columns
- Material exhaustion
  - $f(\gamma, T_{\text{irr}})$

# PBMR Design – Main Power System

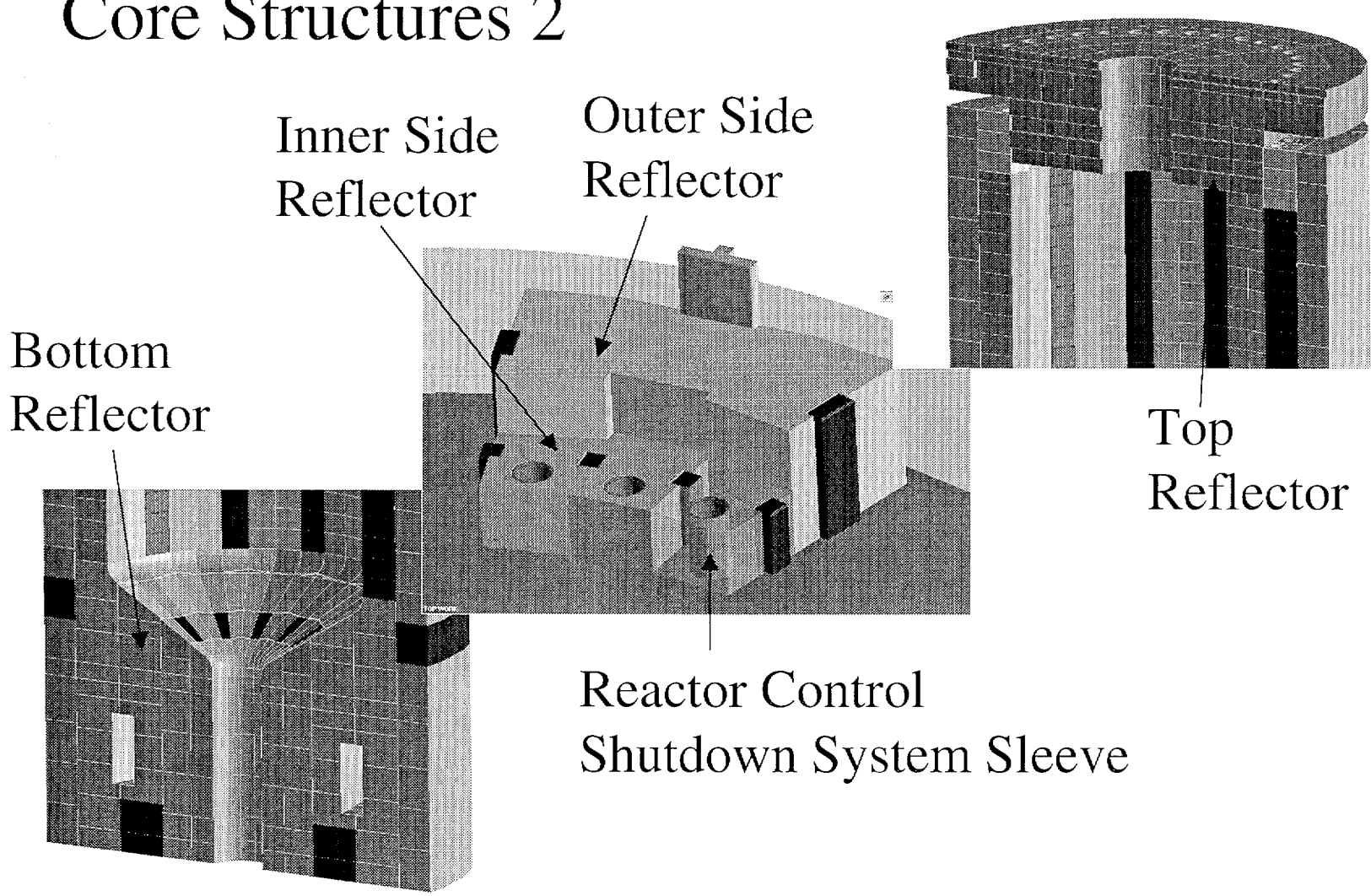


# PBMR Design – Core Structures 1



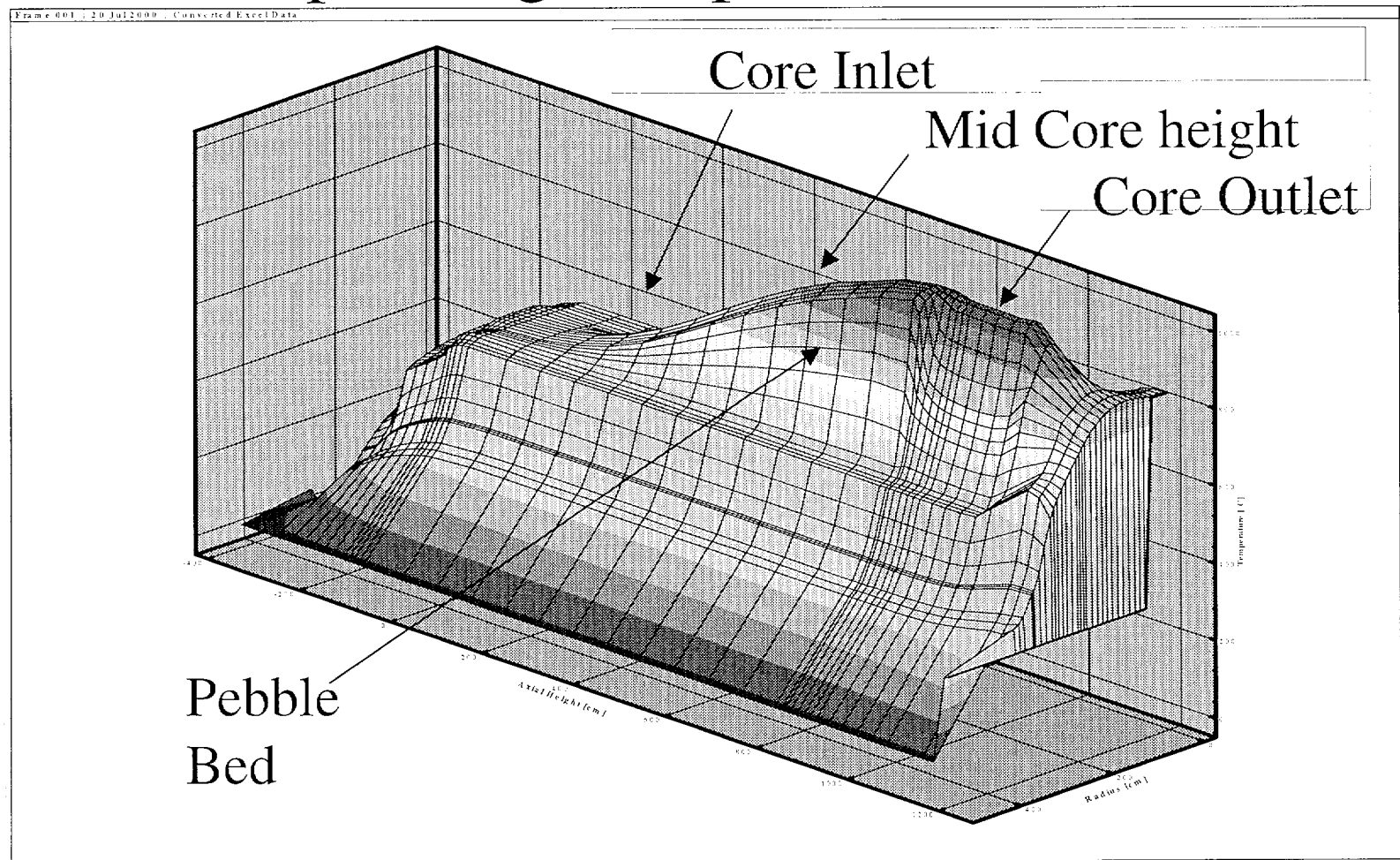
# PBMR Design –

## Core Structures 2



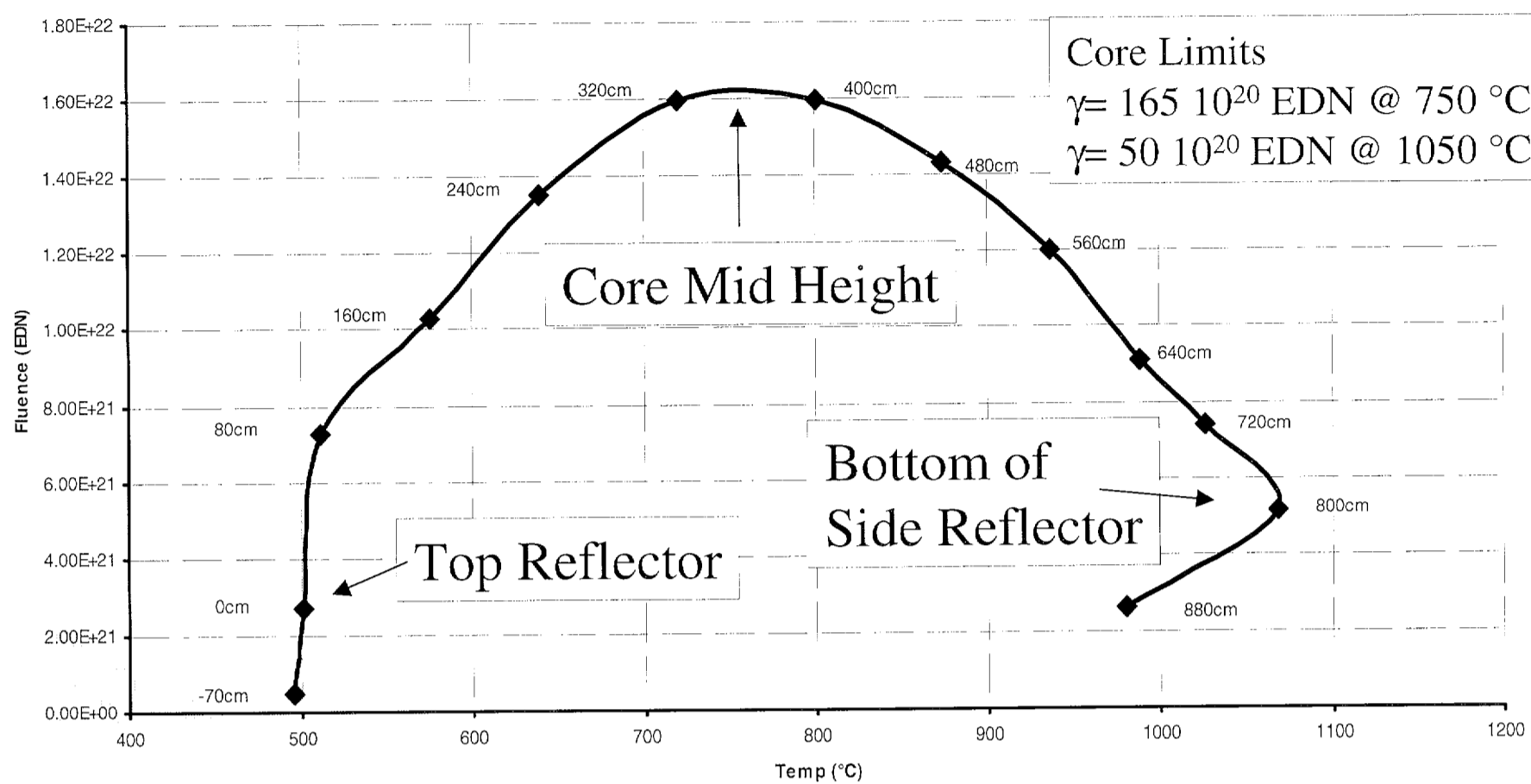
PBMR Reactor Data –

Normal Operating Temperature



# PBMR Reactor Data –

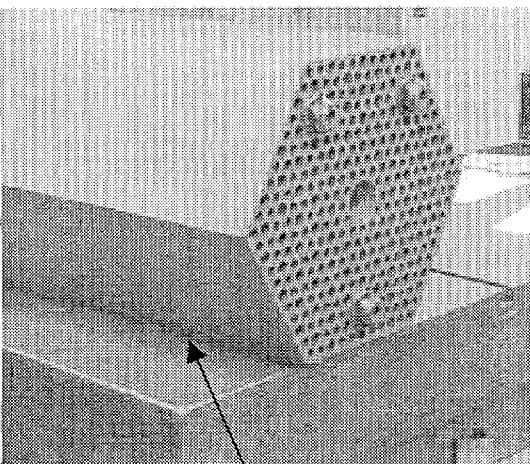
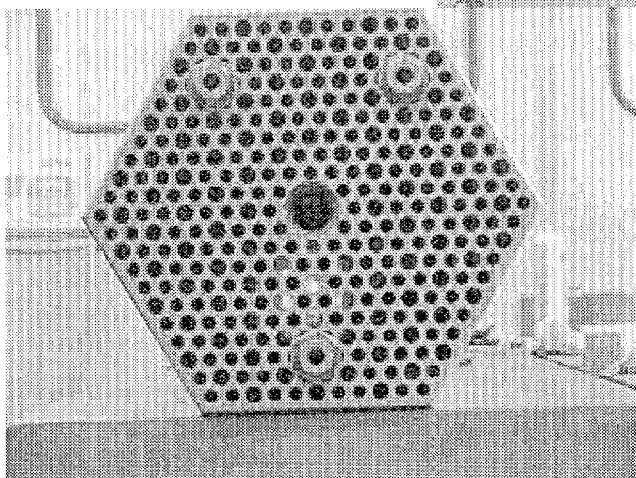
Fluence EDN ( $7.62 \cdot 10^{20} \text{ n/cm}^2 \text{ EDN} \equiv 1 \text{ dpa}$ )



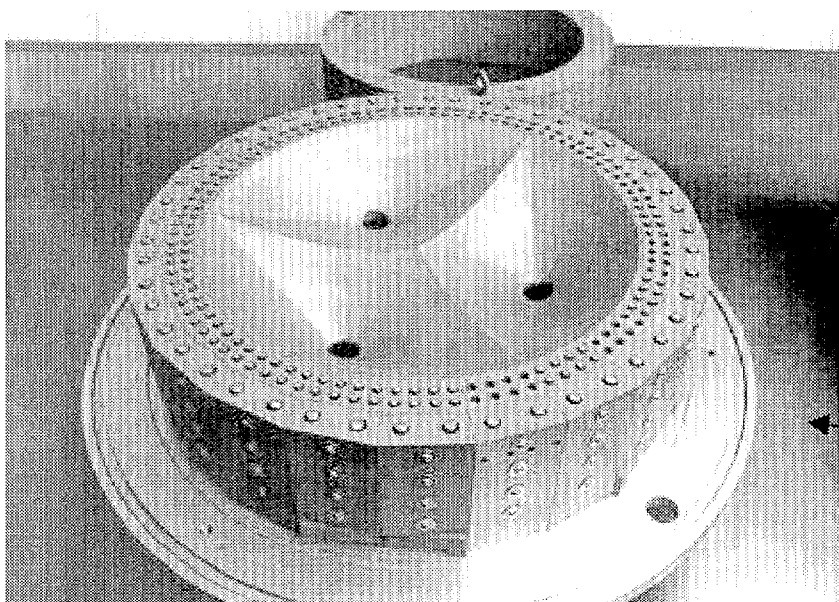
# Nuclear Graphite Manufacture



UK AGR  
Fuel Sleeve



US GA  
Prismatic  
Fuel Design

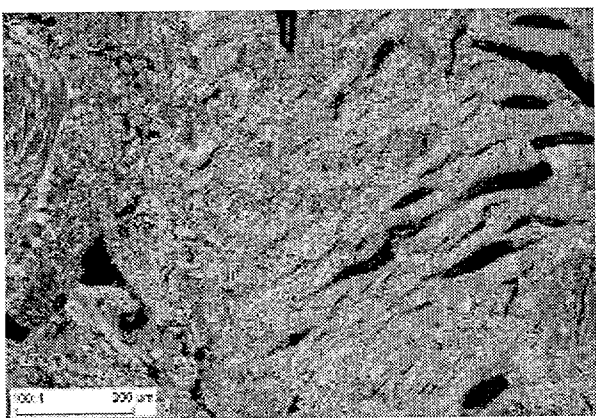


German  
HTR 500  
1/5 Scale

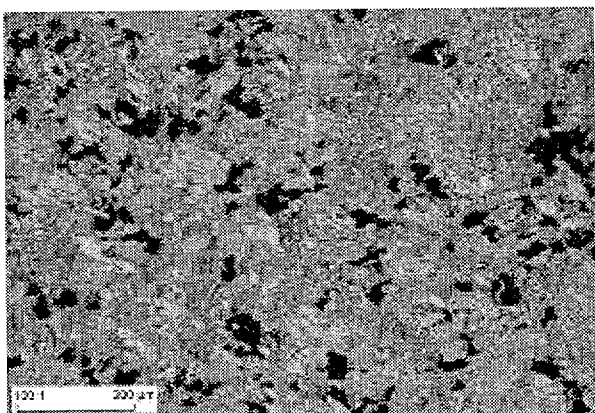
# Nuclear Graphite Manufacture

## Particle size of different materials

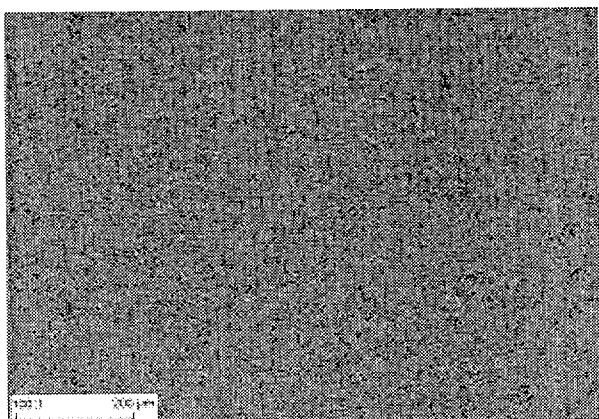
Coarse grain



Medium grain



Fine grain

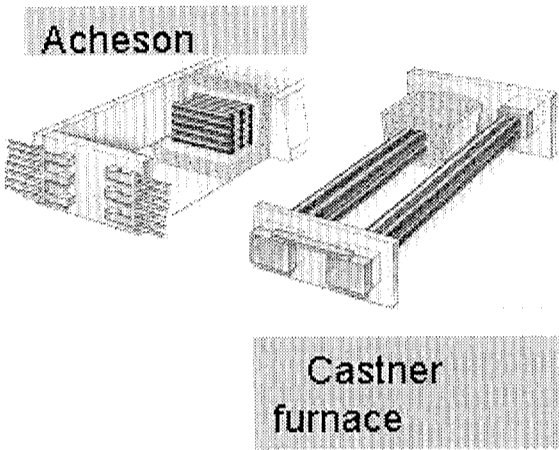
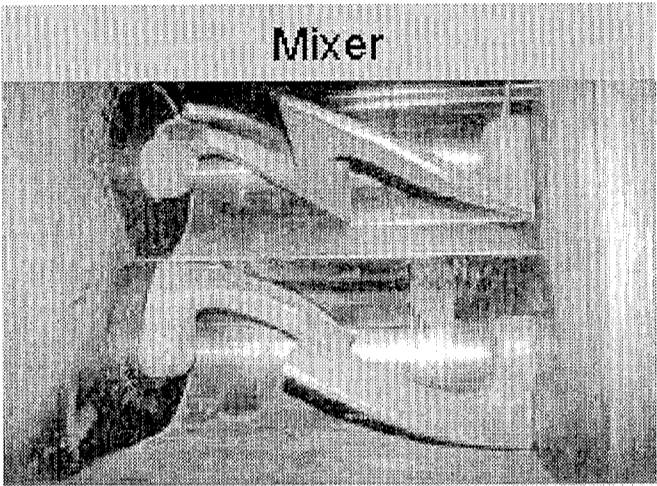
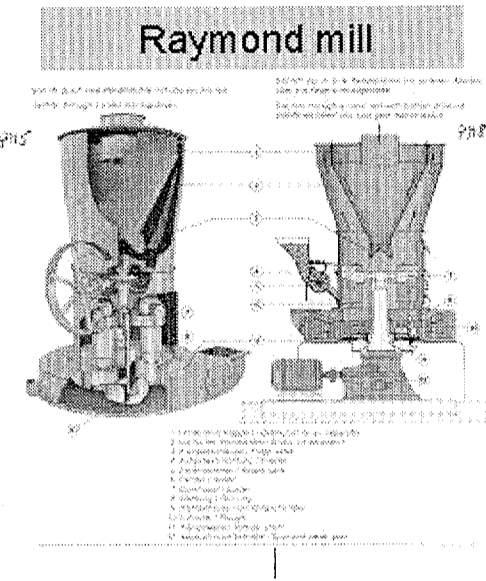
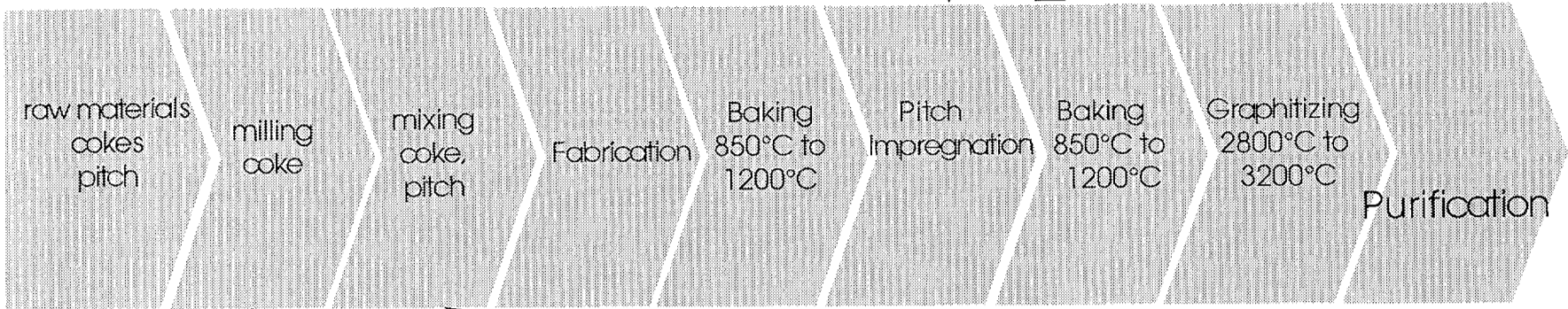


Electrodes

Most Nuclear Applications, e.g. MAGNOX, AGR, AVR, THTR

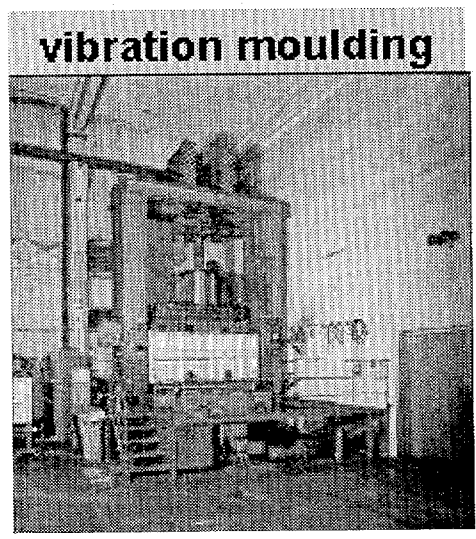
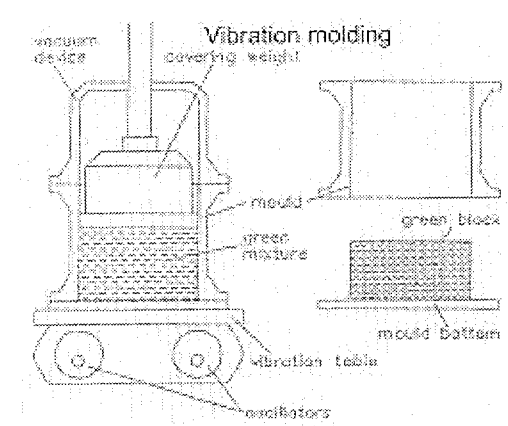
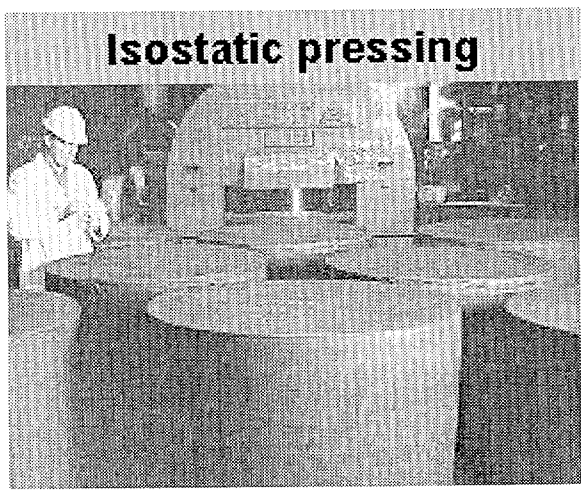
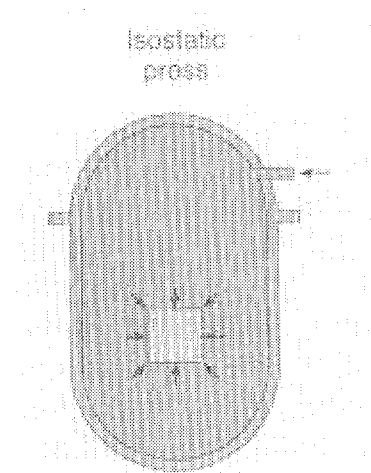
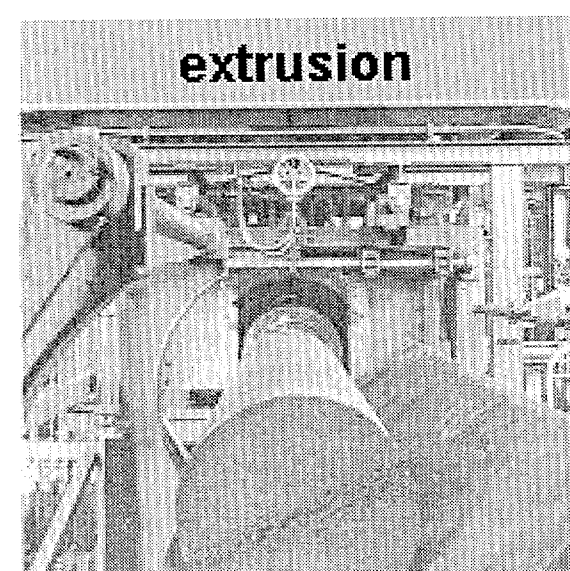
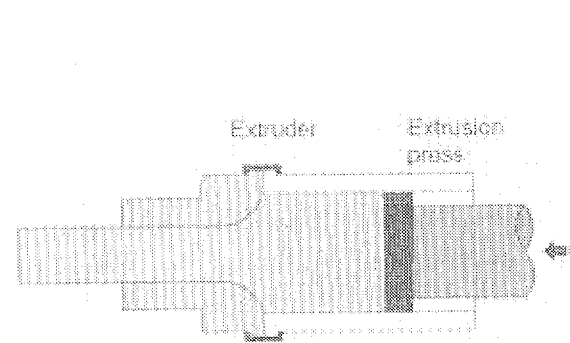
Recent Nuclear Applications, e.g. HTTR, HTR-10

# Nuclear Graphite Manufacture



# Nuclear Graphite Manufacture

## Fabrication techniques



# Manufacturing Summary

- The properties of Nuclear Grade Graphites can be determined by suitable choice of
  - Raw Materials
  - Grain size
  - Manufacturing Route
    - Fabrication
    - Impregnation
    - Purification

# PBMR - SGL Unirradiated

## Material Properties

Property	Sleeve	Grade 1	Grade 2	Iso	Sasol
Density (10 <sup>3</sup> Kg/m <sup>3</sup> )	1.79	1.75	1.79	1.82	1.79
Thermal Conductivity @ Room Temperature (W/m.K)	Not known	130	130	133	130
CTE 20-200 °C (10 <sup>-6</sup> K)	4.35	4.2	4.4	4.1	4.7
Neutron Absorbancy (mBarns)	4.62	4.5	5.4	4.0	12
Compressive Strength (MPa)	72	55	65	100	70
Modulus of Elasticity (GPa)	8.9	9	10	10	10
Poisson's Ratio	0.21	0.21	0.21	0.21	0.21
Anisotropy Ratio (Par/Per)	1.1	1.1	1.1	1.1	1.1

# PBMR – SGL

## Material Impurities ppb

	Element	Li	Be	B	Na	Mg	Al	K	Ca
Typical		2.1	380	250	310	45	680	79	2.6

	Element	Sc	Ti	V	Cr	Mn	Fe	Co	Ni
Typical		15	3.8	100	16	75	500	3.4	37

	Element	Cu	Zn	Ga	Ge	Se	Rb	Sr	Y
Typical		43	40	3.2	24	21	2.7	2.5	5.7

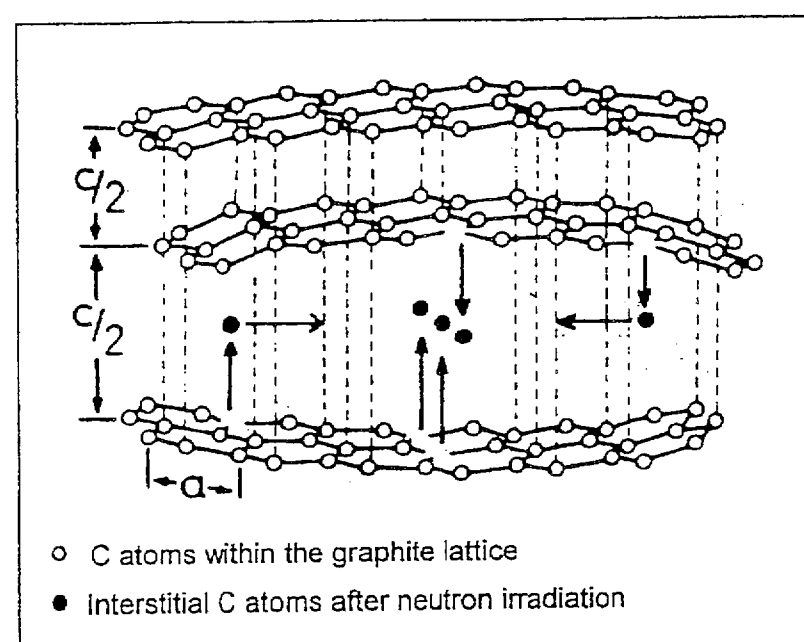
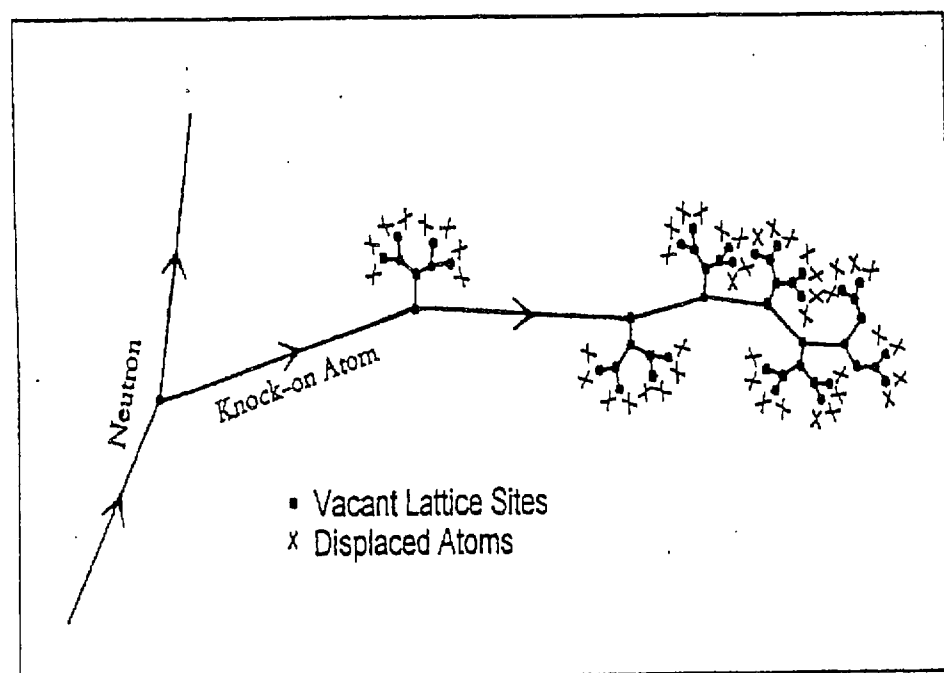
	Element	Zr	Nb	Mo	Ru	Rh	Ag	Cd	In
Typical		3.7	1.9	12	6	1.9	3.7	15	150

	Element	Sn	Sb	Te	Cs	Ba	La	Ce	Pr
Typical		6	3.4	27	1	2	1.9	3.8	7.6

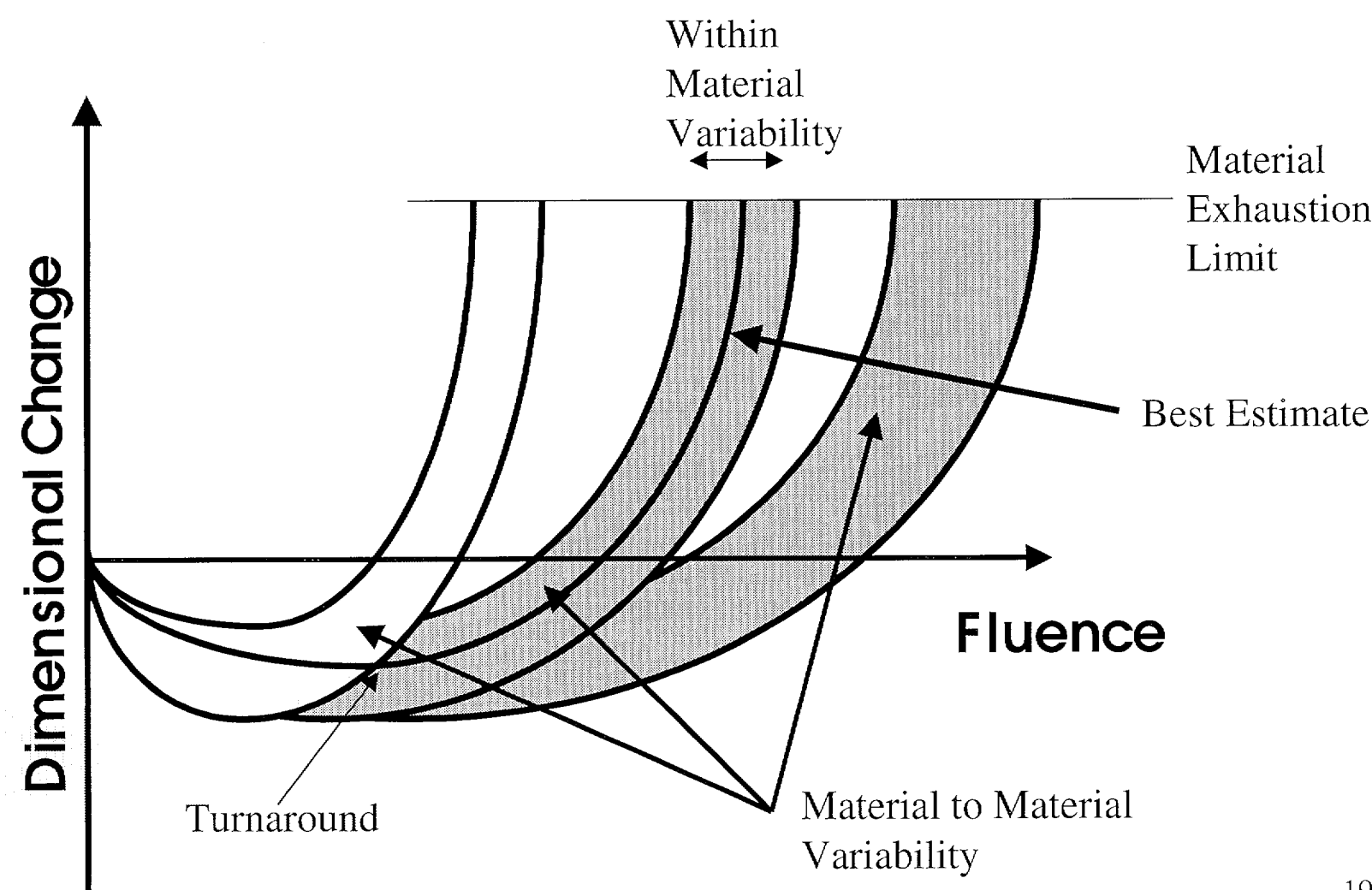
	Element	Nd	Sm	Eu	Gd	Hf	Ta	W	Re
Typical		11	13	4	12	7.1	4.1	7.3	5.2

	Element	Au	Hg	Tl	Pb	Bi	Th	U	
Typical		1.3	4.2	1.8	3.1	1	1	2.4	

# Graphite Irradiated Behaviour



# Material Property Variation



# Dimensional Change

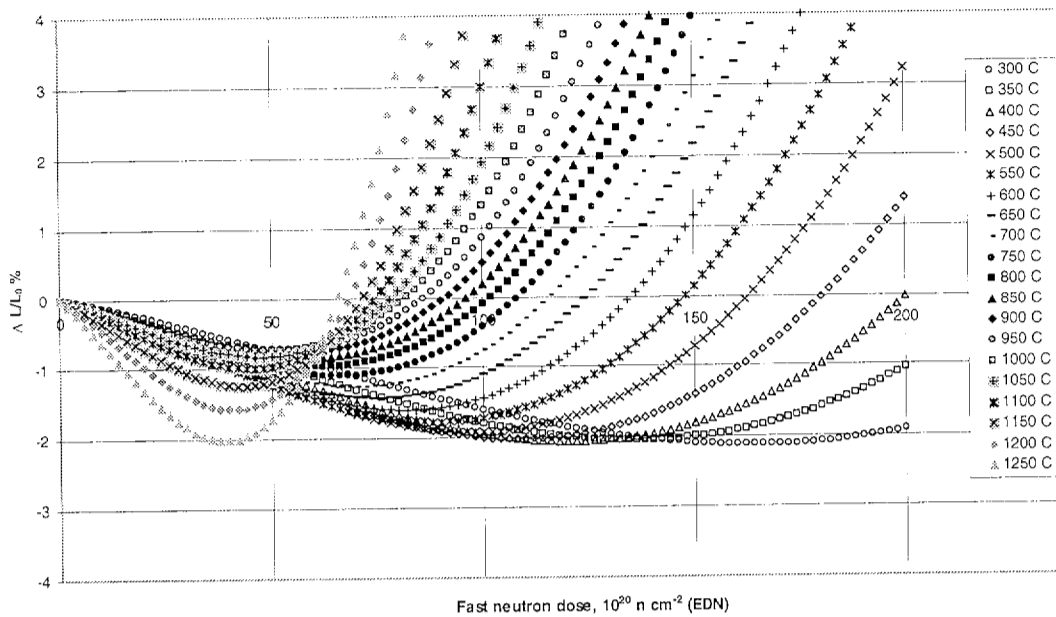


Figure 18. Perpendicular dimensional changes for generated PBMR graphite, for different values of EDT

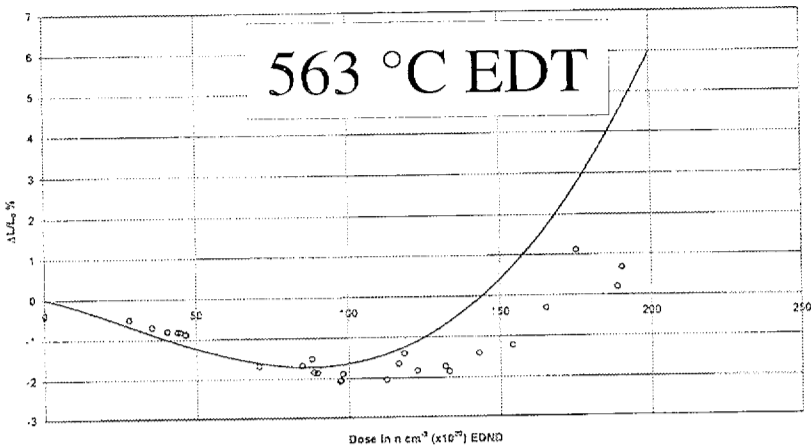


Figure 14. Comparison of predicted dimensional change curves with experimental data (perpendicular direction)

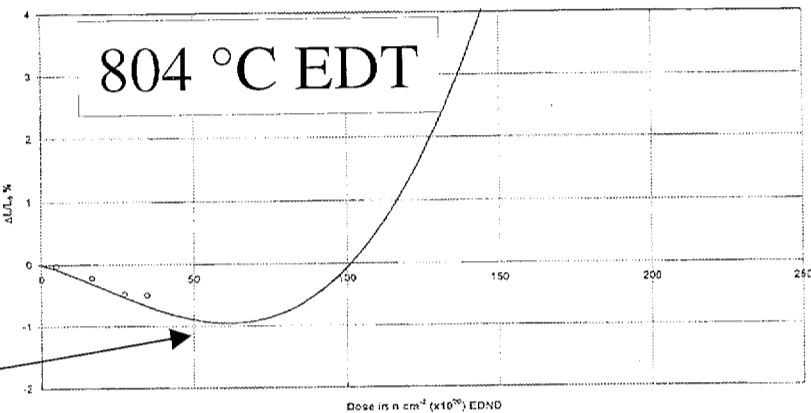
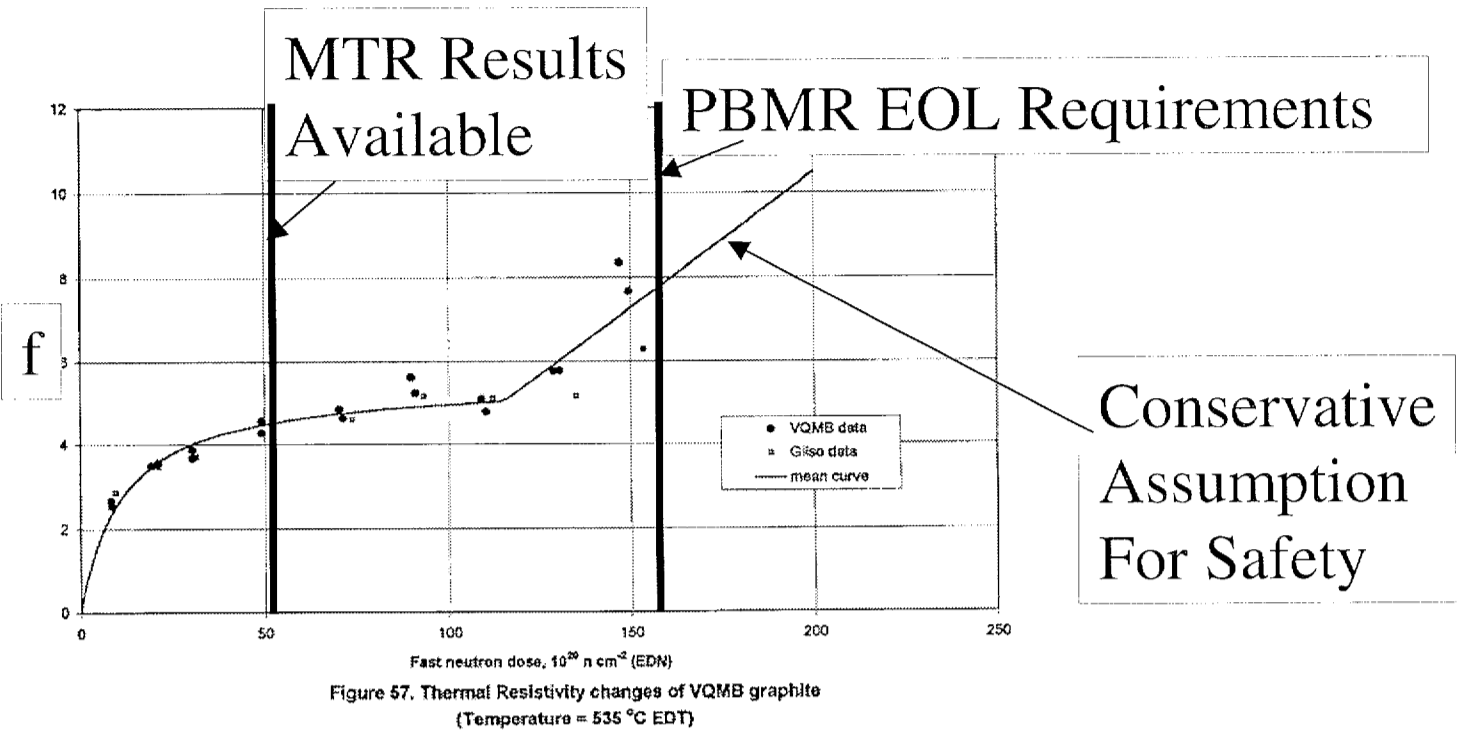


Figure 15. Comparison of predicted dimensional change curves with experimental data (perpendicular direction)

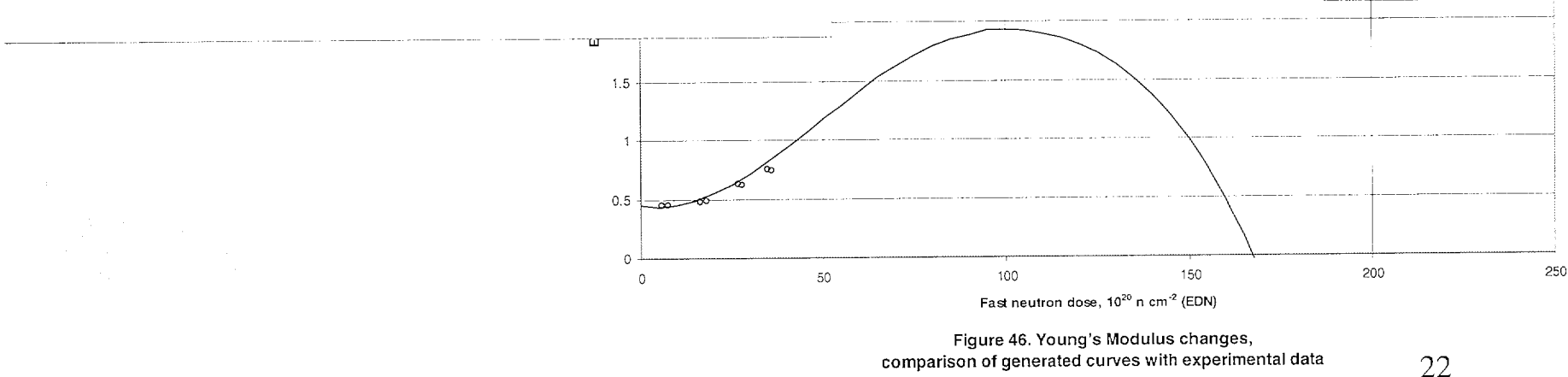
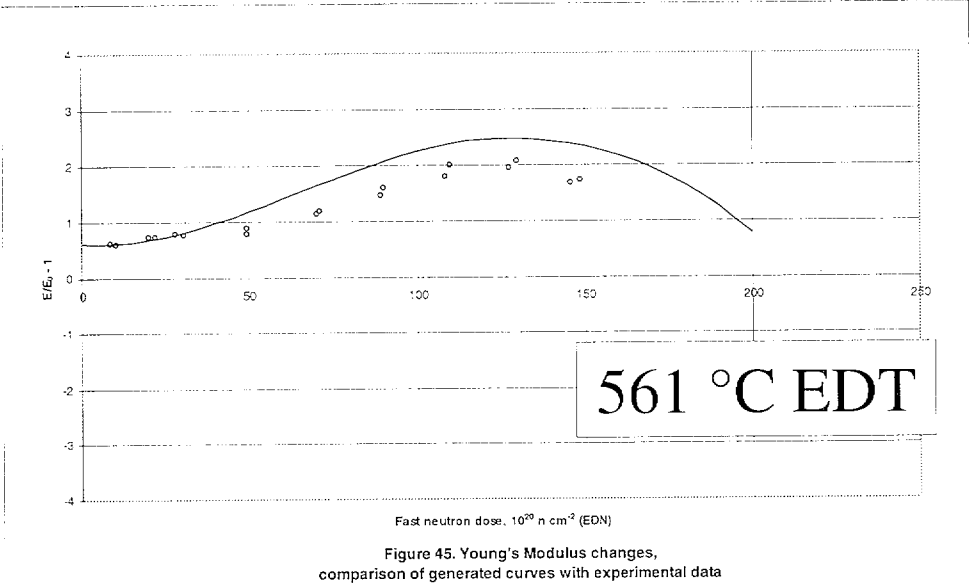
Turnaround

# Thermal Resistivity



$$\frac{1}{K_y(\gamma,T)} = S_k(\gamma) \left[ \frac{1}{K_y(0,T)} + \delta(T) \frac{f}{K(0,30)} \right] \left[ \frac{K_o}{K} \right]_{ox}$$

# Young's Modulus



# Irradiation Induced Creep

## (Beneficial)

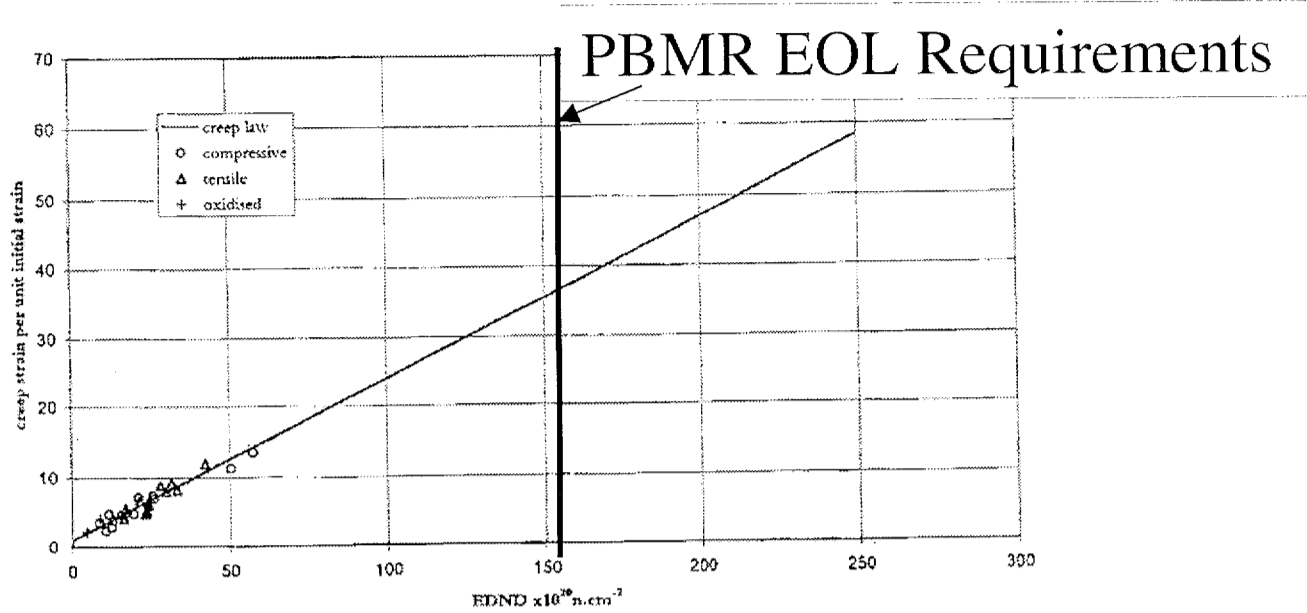


Figure 52 Constant Stress Irradiation Creep Data (Isotropic Graphite)

$$\varepsilon_c = \alpha(T_i) \frac{\sigma}{E_c} \left[ 1 - e^{-\frac{\gamma}{\gamma_o}} \right] + \frac{K}{E_c} \beta(T_i) \sigma \gamma$$

# Strength

- Weibull ‘weakest link’ theory
  - Typical Nuclear graphite Weibull Modulus = 10
- German Performance Assessment Model
  - Probability of Failure =  $10^{-4}$
  - Safety Factor = 2.4
- Irradiation behaviour is correlated with Young’s Modulus
  - Pre turnaround (Y.M irradiation induced change)<sup>1/2</sup>
  - Post turnaround (Y.M irradiation induced change)<sup>1</sup>

# Material Summary

- There is considerable graphite data, empirical relationships and operational experience available to PBMR from previous gas-cooled, graphite moderated reactor programmes, e.g. MAGNOX, AGR, AVR & THTR.

- Pre-Turnaround Graphite Behaviour

There is a high degree of confidence that the existing graphite database is sufficient to describe the behaviour of graphites currently available to PBMR up to the point of turnaround – approximately 15 years of PBMR operation at the peak flux position (inner surface of inner side reflector at mid-core height).

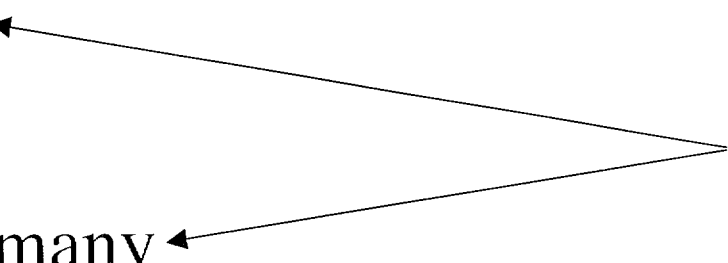
# Summary (Continued)

- Post-Turnaround Graphite Behaviour

Beyond the point of turnaround there is uncertainty in the PBMR graphite database for performance assessment due to the following:

- lack of knowledge of behaviour of graphite at high fluence (beyond return to initial volume)
- lack of actual data for PBMR graphite
- Validation of reactor parameters

# Performance Assessment

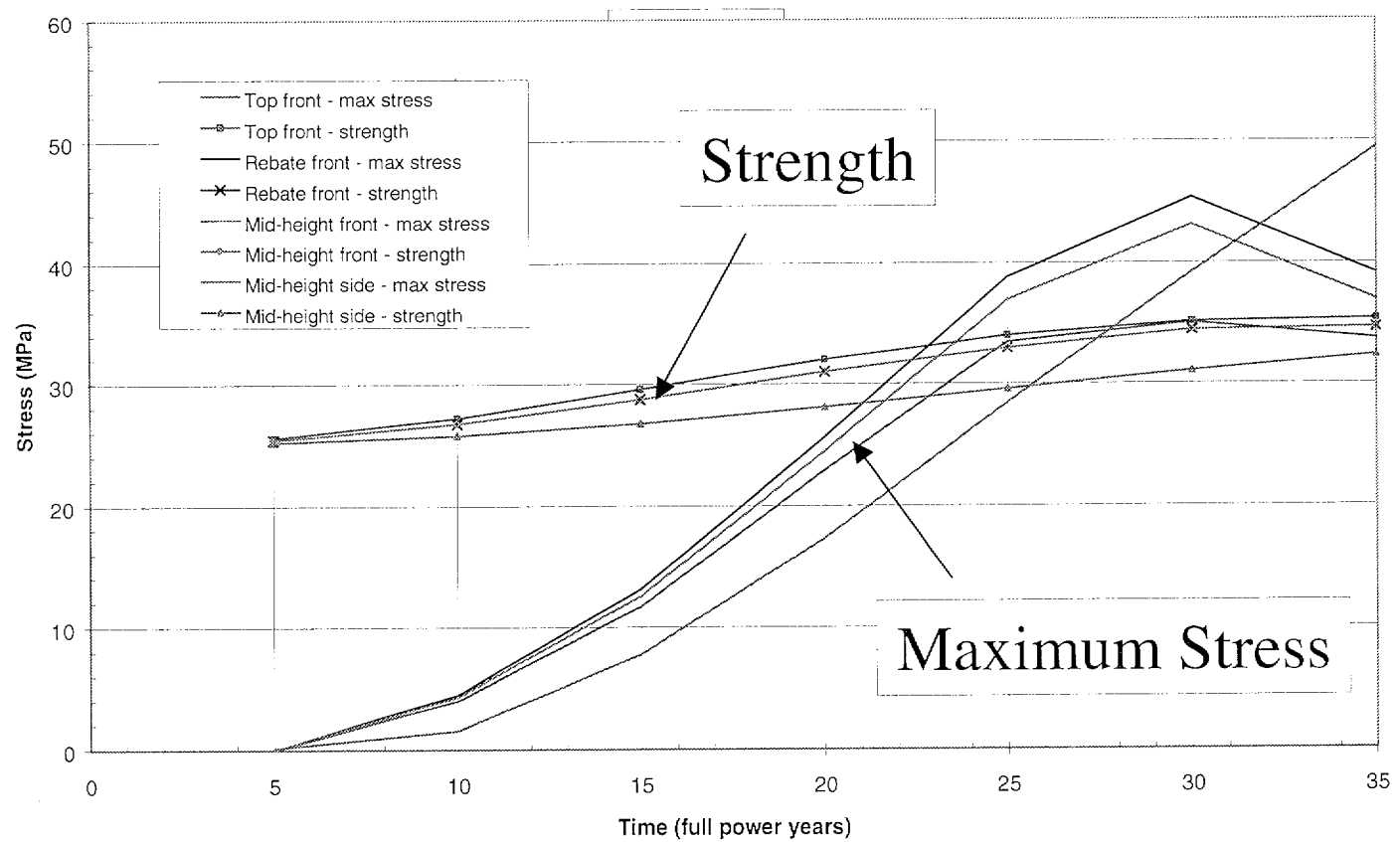
- There are numerous different approaches to assessment of structural performance of graphite moderated cores
    - UK
    - US
    - Germany
    - Japan
- (PBMR adopted)
- 

# Structural performance

- Primary or External
  - The reflector is subject to several external loads pebble bed hydrostatic pressure, coolant differential pressure, deadweight and pebble bed 'breathing' pressure.
- Secondary or Internal
  - Under irradiation graphite exhibits significant dimensional change and material property change. These property changes can set up significant shrinkage and thermal stresses. Additionally when graphite is subjected to load it exhibits creep.

# Stress Time History

## Deterministic Best Estimate



# Deformation

- Local deformation is caused within the block due to shrinkage strain, thermal strain and creep strain.
- Large global deformation could occur towards end of life due to the accumulation of local deformation, i.e. within a column. This may lead to
  - control rod articulation limits being exceeded
  - unacceptably high leakage/bypass flows and unacceptable peak fuel temperatures

# Material Exhaustion

- As discussed earlier at high fluence, graphite exhibits significant swelling and associated reduction in Modulus and Strength. This is a material limit and is determined by the irradiation fluence and temperature
- The limit for PBMR graphite has been nominally set to 5% swelling and some parts of the side reflector exceed this value for the assumed material data.

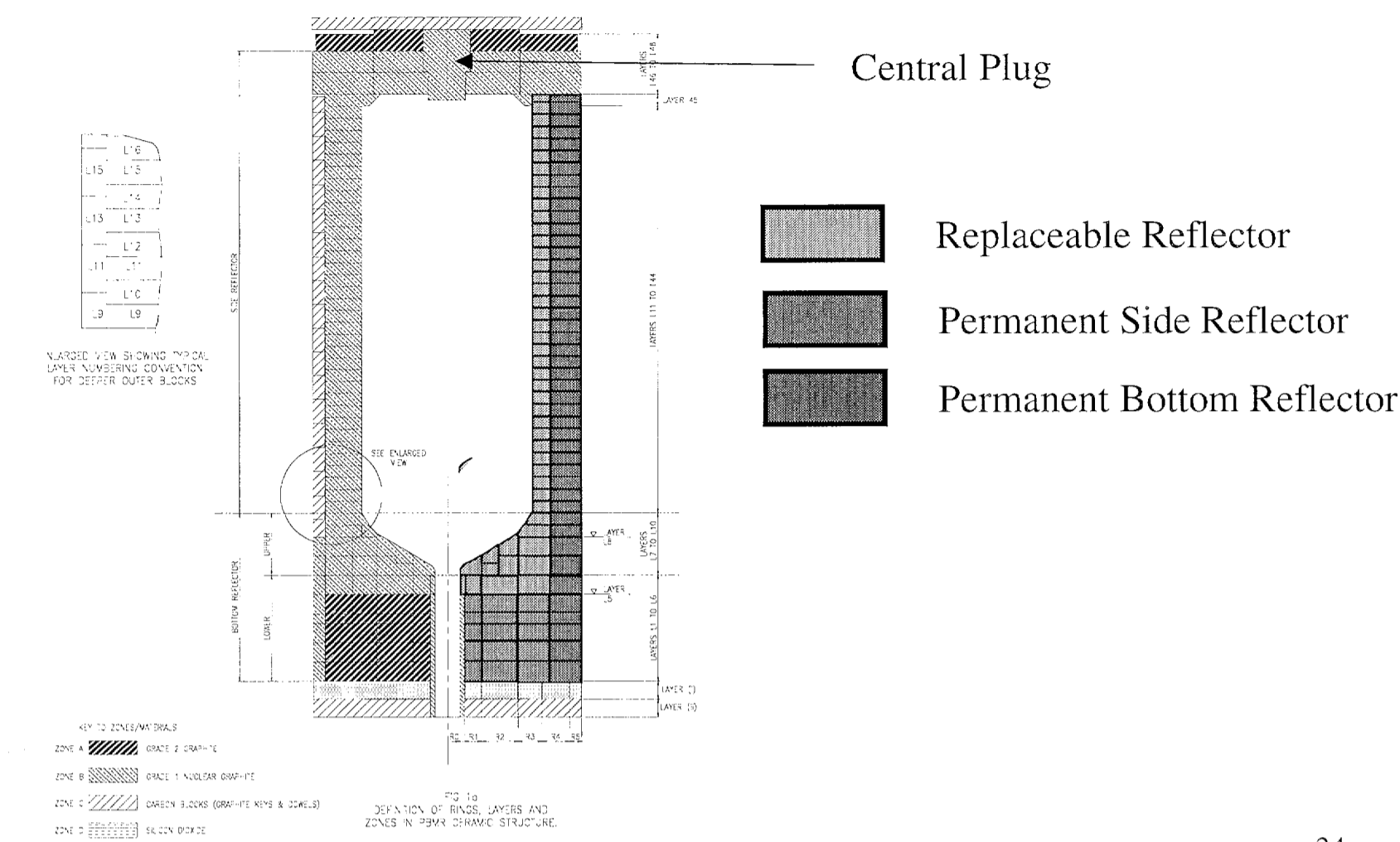
# Performance Summary

- PBMR has adopted a combination of German and UK approaches to graphite component performance assessment
- Up to turnaround, assuming the current graphite database, the performance assessment criteria, structural, deformation and material exhaustion, are met for the current design of the PBMR.
- At EOL, assuming the current graphite database, the performance assessment criteria, structural, deformation and material exhaustion, are not met for the current design of the PBMR.

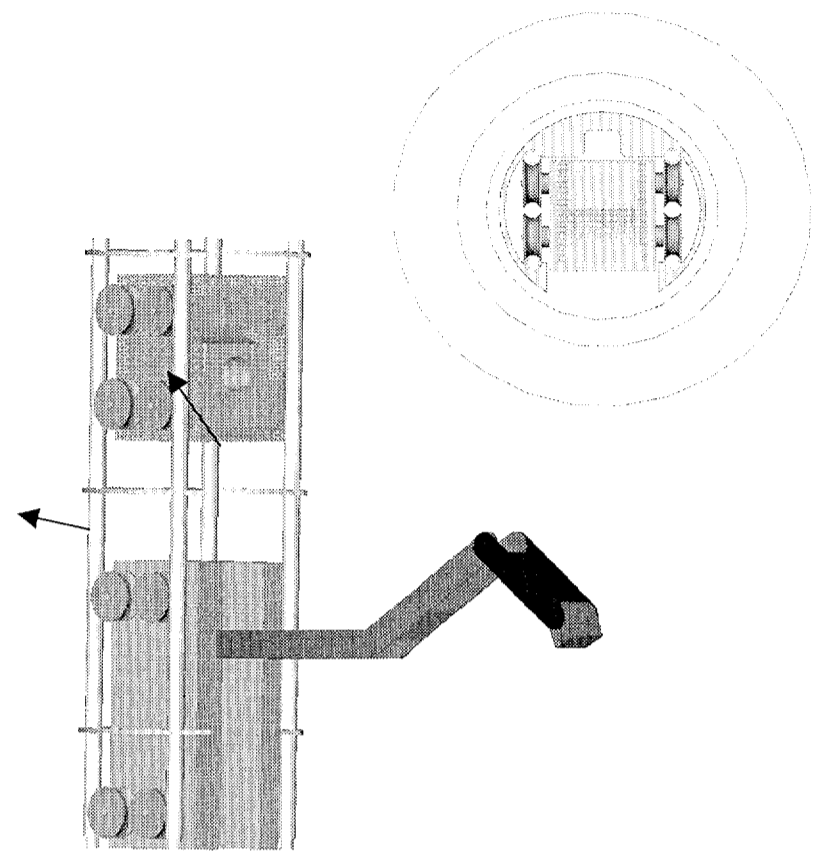
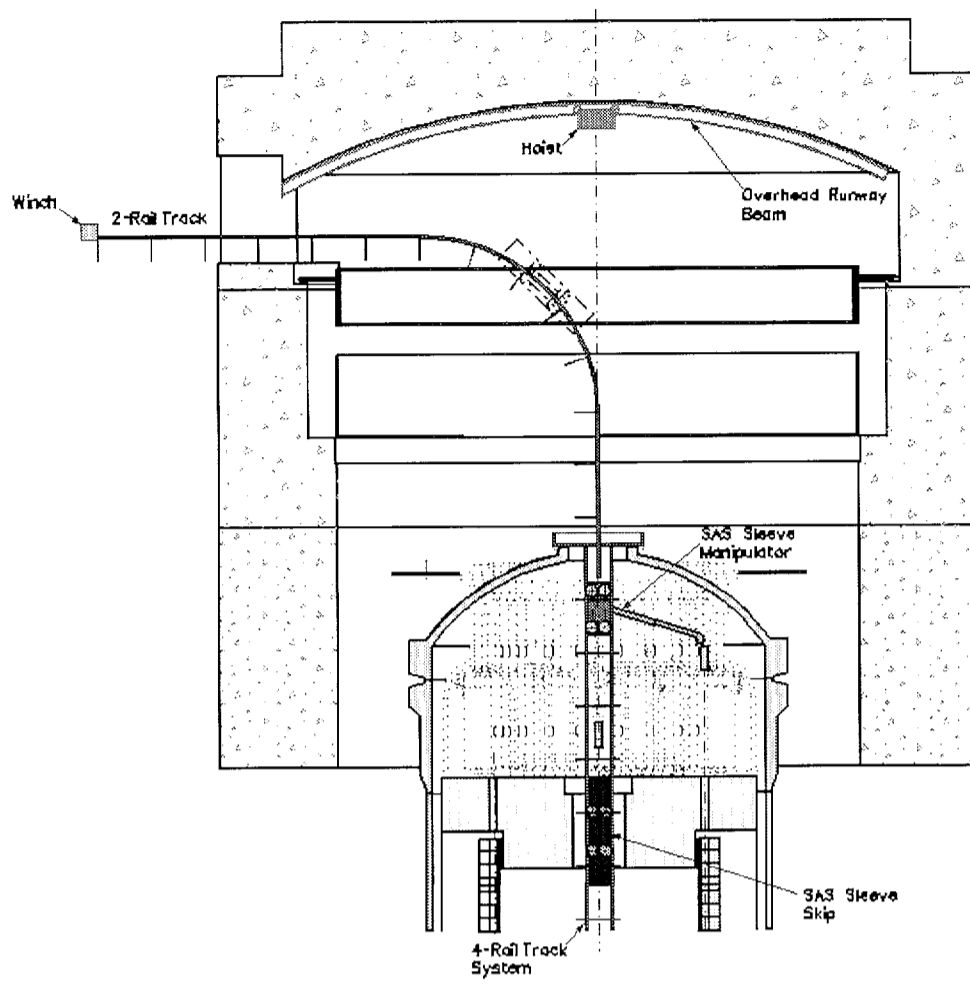
# Risk Mitigation

- MTR Programme to determine irradiation behaviour beyond turnaround
  - To achieve PBMR fluence will require
    - 8-10 year programme at a low flux facility
    - 14 months at a high flux facility
- Inner Reflector Replacement
  - via the central Plug

# Inner Reflector Replacement



# Inner Reflector Replacement System



# Inner Reflector Replacement

- Replacement of the Inner Reflector will:
  - Mitigate risk from lifetime issues
    - Structural performance
    - Distortion
    - Material Exhaustion
  - Increase margin to peak fuel temperature limit (lower thermal resistivity @ EOL)
  - Allow continued operation w/o replacement subject to satisfactory MTR programme results

# Conclusions 1

- There is substantial experience in operating gas-cooled, graphite moderated reactors around the world, e.g. UK.
- Suitable Nuclear Grade Graphites can be determined by appropriate choice of manufacturing process parameters

## Conclusions 2

- Sufficient information is available to justify PBMR operation up to the point of turnaround, approximately 15 full power years at the peak flux position.
- Graphite Technology is still mainly empirical, especially at high fluence and temperature and uncertainty exists in the material database assumed for PBMR beyond turnaround

## Conclusions 3

- A MTR programme is required to characterise PBMR graphite and to remove uncertainty associated with performance assessment of PBMR graphite components beyond turnaround
- The risks associated with performance of graphite components in PBMR can be mitigated by replacement of the inner reflector

# PBMR Systems Design Approach and Status

October 25, 2001

Vijay Nilekani  
Exelon Generation Co.

# Overview of Presentation

- Overall PBMR Design Phases & Status
- Systems Design Approach
  - PBMR System Categories
  - Vertical slice of systems design process, using Fuel Handling & Storage System as an example

# Overall Design Phases

- Conceptual Design Phase: To provide proof of concept.
- Basic Design Phase: To provide technical & commercial feasibility data to investors and other stakeholders.
- Detailed Design Phase: To provide a detailed, constructible design.

The project is currently approaching the end of the Basic Design Phase.

Conceptual Design	Basic Design Phase	Detailed Design Phase
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↑  
Current Status 10/25/2001

# Systems Design Approach

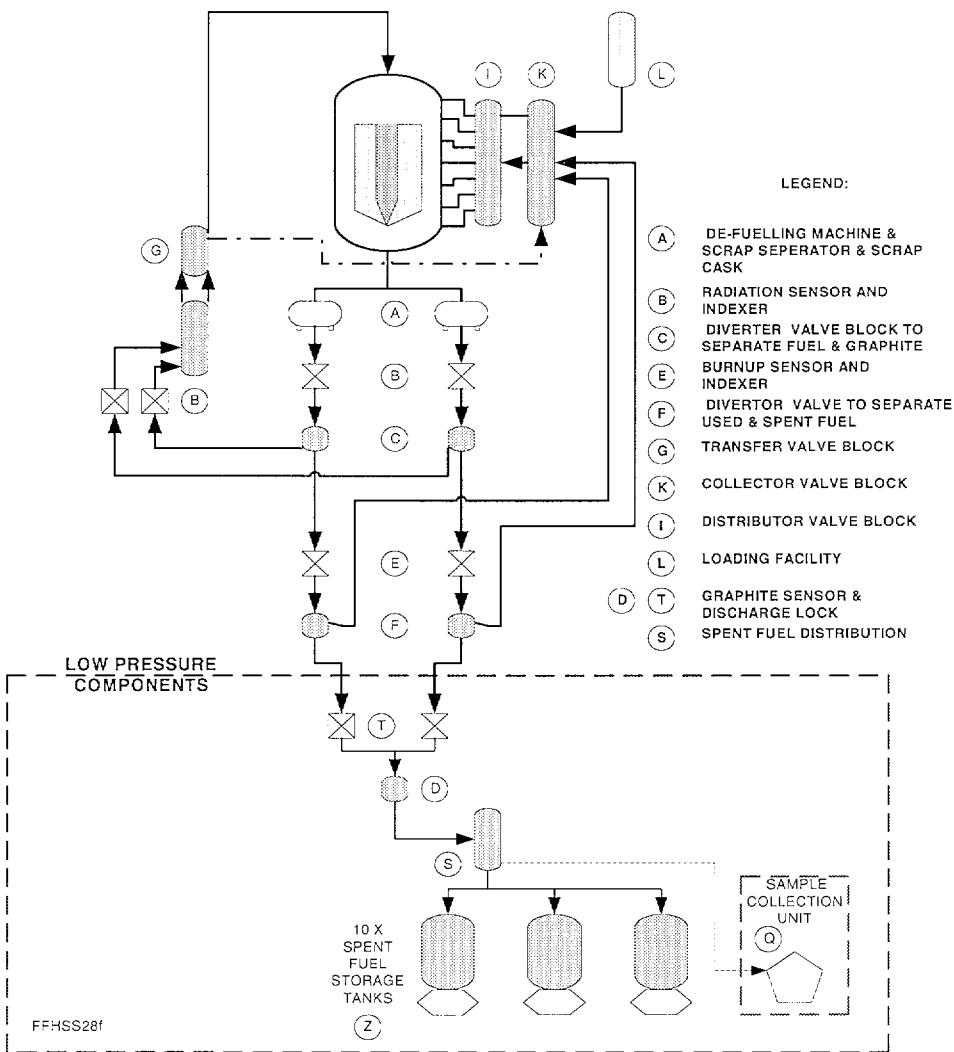
- The plant systems are designed using the process described in the “Engineering Management Plan,” Document No. 000370-120.
- This is illustrated by a vertical slice of the systems design process using the Fuel Handling & Storage System as an example.
- PBMR System categories as follows.

# PBMR System Categories

- Reactor Pressure Vessel
- Automation
- Electrical/I&C
- Helium Gas
- Plant Services
- Civil / Buildings / Infrastructure
- Water Cooling
- Main Power
- Radiological Waste
- Fuel Handling & Storage
  - **Fuel Handling and Storage System**
- Plant Support

FUEL HANDLING SYSTEM  
SPHERE FLOW DURING NORMAL  
OPERATION

HIGH PRESSURE OPERATION



# Fuel Handling & Storage System System Design Considerations

- Requirements & Specifications
- Applicable Documents & Models
- Requirements
  - System Definition
  - Characteristics
    - Performance
    - Modes and States
    - Functions

# Fuel Handling & Storage System

## (Cont'd)

- Interface Requirements
- Physical Characteristics
- Availability
- Reliability
- Maintainability
- Environmental Condition
  - Seismic Loading
  - External thermal & hydraulic conditions
  - Vibration
  - EMI
  - Transportation shocks and vibrations
  - Storage
  - Transportability

# Fuel Handling & Storage System

## (Cont'd)

- Design & construction
- Documentation
- Logistics
- Personnel & Training
- Major Components Characteristics
  - Sub-systems
    - Fuel & Graphite Circulating
    - Spent Fuel Storage
    - Fresh Fuel Storage & Feed

# Fuel Handling & Storage System

## (Cont'd)

- Sub-Systems (cont'd)
  - Graphite Replenishing
  - Used Fuel & Graphite Storage & Feed
  - First Core Graphite Loading
  - Spent Fuel & Last Core Removal
  - Gas Evacuation
  - Fuel Handling & Storage Control
  - FHSS Logistical Support

# Fuel Handling & Storage System

## (Cont'd)

- Functional Analysis (input to process flow diagram)
- Integrated Logistical Support Plan
  - Summary of System Characteristics
    - System overview
    - Operational profile and parameters
    - Maintenance parameters

# Fuel Handling & Storage System

## (Cont'd)

- Reliability program
  - Availability predictions
  - Redundancy
- Maintainability program
- Testing Program
- Human Factors Engineering
- Safety Engineering

# Fuel Handling & Storage System

## (Cont'd)

- Customer/Contractor ILS Planning Process
- Plan for Support
- Maintenance concepts
  - System Characteristics
  - Repair Levels & Functions at each level
    - Condition Monitoring
    - Preventive Maintenance
    - On-site Repair

# Fuel Handling & Storage System

## (Cont'd)

- Repair Levels & Functions at each level  
(cont'd)
  - Workshop
  - Regional (off-site)
  - Factory
- Requirements and Policies
  - Spares & Stores
  - Personnel
- Design Codes & Standards
- Design Review

# Fuel Handling & Storage System

## (Cont'd)

- Design Reports
  - Process Flow Analysis
  - Simulation Requirements
  - Simulation Data
  - Simulation Results
    - Different modes and states
  - Analysis of effect of external parameters
  - Level of Confidence
  - Appendices

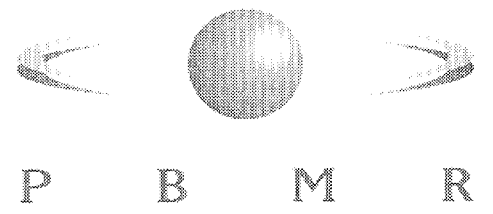
# Fuel Handling & Storage System

## (Cont'd)

- Thermo Hydraulic Control Strategy
- System Simulink Input Report
- System Transient Analysis Report
- CFD Simulation of Spent Fuel Tanks
- Thermal Hydraulic Design Report
- Thermo Hydraulic Flownet Input Report
- Fuelnet Simulator Input Report
- Development Specification
- Industrialization Plan

# SUMMARY

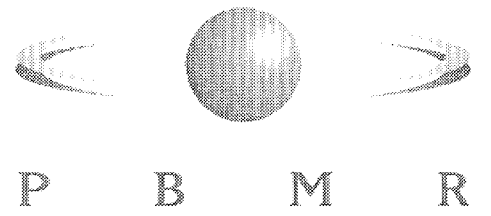
- A comprehensive, consistent and rigorous review process used in developing the System design
- Design principles, processes and considerations are similar to other Nuclear design projects
- Fuel Handling and Storage System design documentation is available for NRC examination



# Control of Chemical Attack in the PBMR

Albert Koster DSc - PBMR Pty (Ltd)

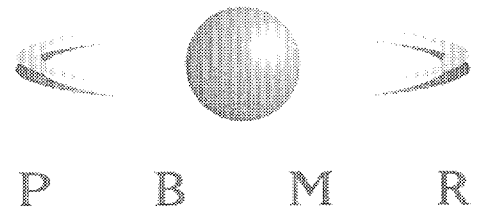
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1999-2000



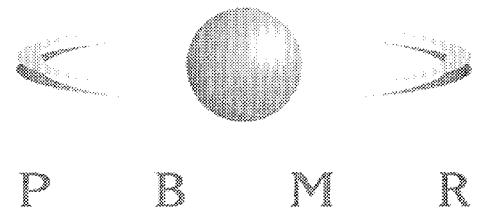
# Purpose of Presentation

- To discuss the PBMR safety design approach to control of chemical attack

# Background



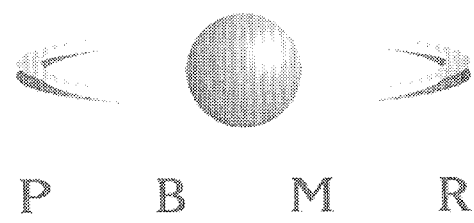
- Severe accidents in nuclear reactors (e.g. Windscale, Chernobyl) have resulted in graphite fires
- Water ingress at the AVR and Ft. St. Vrain HTGRs resulted in lengthy downtimes
- Graphite can corrode at elevated temperatures due to reactions with air or water oxidants
- PBMR safety design approach explicitly focuses on control of chemical attack



# Graphite Fire Severe Accidents

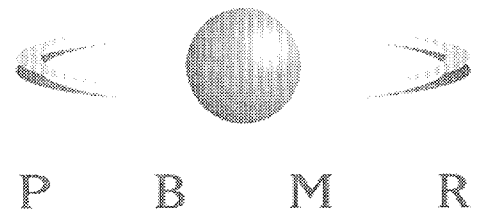
- Windscale event
  - Caused by buildup of radiation-induced Wigner energy in graphite at low temperature operation.
  - Release of energy caused burning of first metallic fuel and then graphite by air reactor coolant.
  - Fire and radionuclide release aggravated by the open cycle chimney air flow.
- Chernobyl event
  - Caused by severe reactivity excursion that destroyed the reactor core.
  - Metallic fuel and graphite burned ~ 20 hours after the explosion opened the core to air ingress.

# Comparison of Windscale and PBMR Air Ingress Resistance

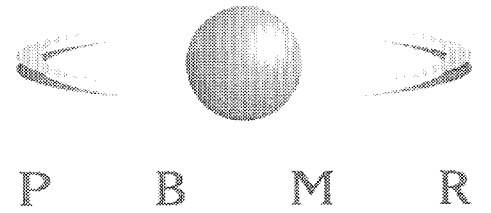


	<u>Windscale</u>	<u>PBMR</u>
• Initiating Event	Wigner energy	n/a
• Coolant	air	inert helium
• Fuel	metallic	ceramic
• Air Supply	unlimited	limited
	(open to atmos by design)	

# Comparison of Chernobyl and PBMR Air Ingress Resistance



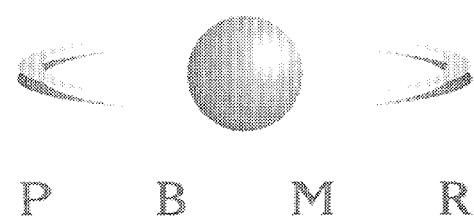
	<u>Chernobyl</u>	<u>PBMR</u>
• Initiating Event	positive reactivity	n/a
• Coolant	water	inert helium
• Fuel	metallic	ceramic
• Air Supply	unlimited (open to atmos)	limited



# Comparison of AVR and PBMR Water Ingress Resistance

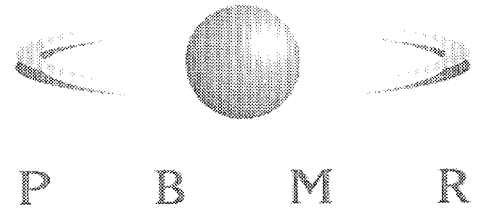
	<u>AVR</u>	<u>PBMR</u>
• Water Source	steam generator	direct cycle
• Coolant	inert helium	inert helium
• Fuel	ceramic	ceramic
• Graphite Type	nuclear grade	nuclear grade

# Comparison of FSV and PBMR Water Ingress Resistance



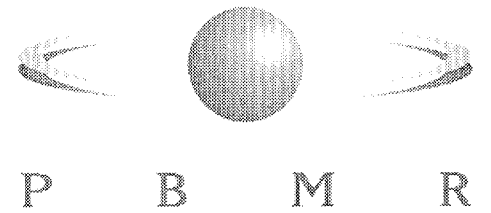
	<u>Ft St Vrain</u>	<u>PBMR</u>
• Water Source	water bearings	magnetic bearings
• Coolant	inert helium	inert helium
• Fuel	ceramic	ceramic
• Graphite Type	PGX core support	higher grade

# PBMR Safety Design Approach

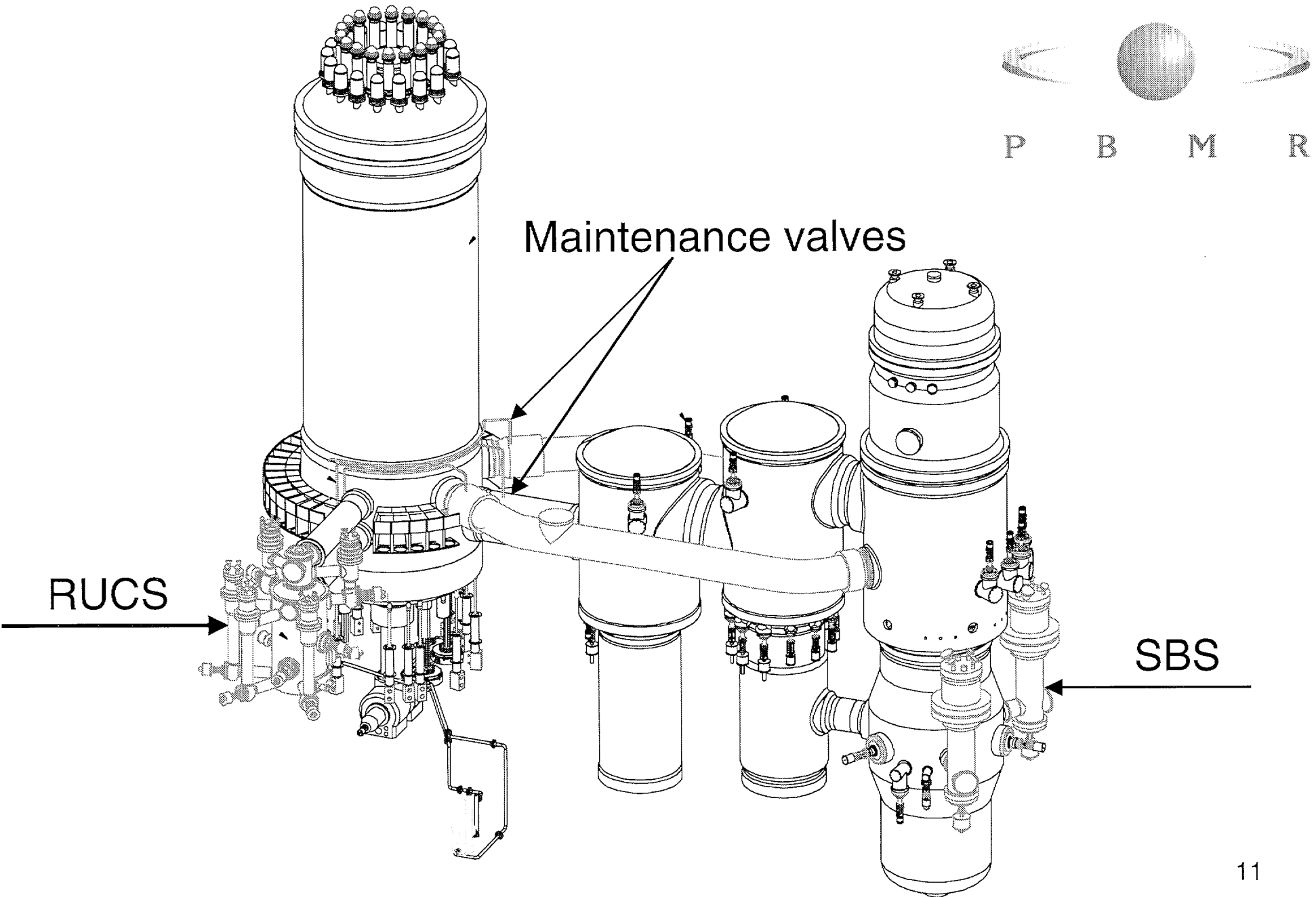


- Prevent water corrosion of graphite by limiting pressurized water sources and supply (e.g., no steam generators)
- Prevent air corrosion of graphite by providing reliable reactor isolation and limiting air supply
- Assure core heat removal and control of heat generation
- Retain radionuclides in SiC-coated fuel particles that are highly temperature and corrosion resistant

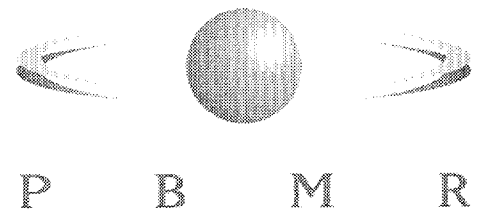
# PBMR Resistance to Water Ingress



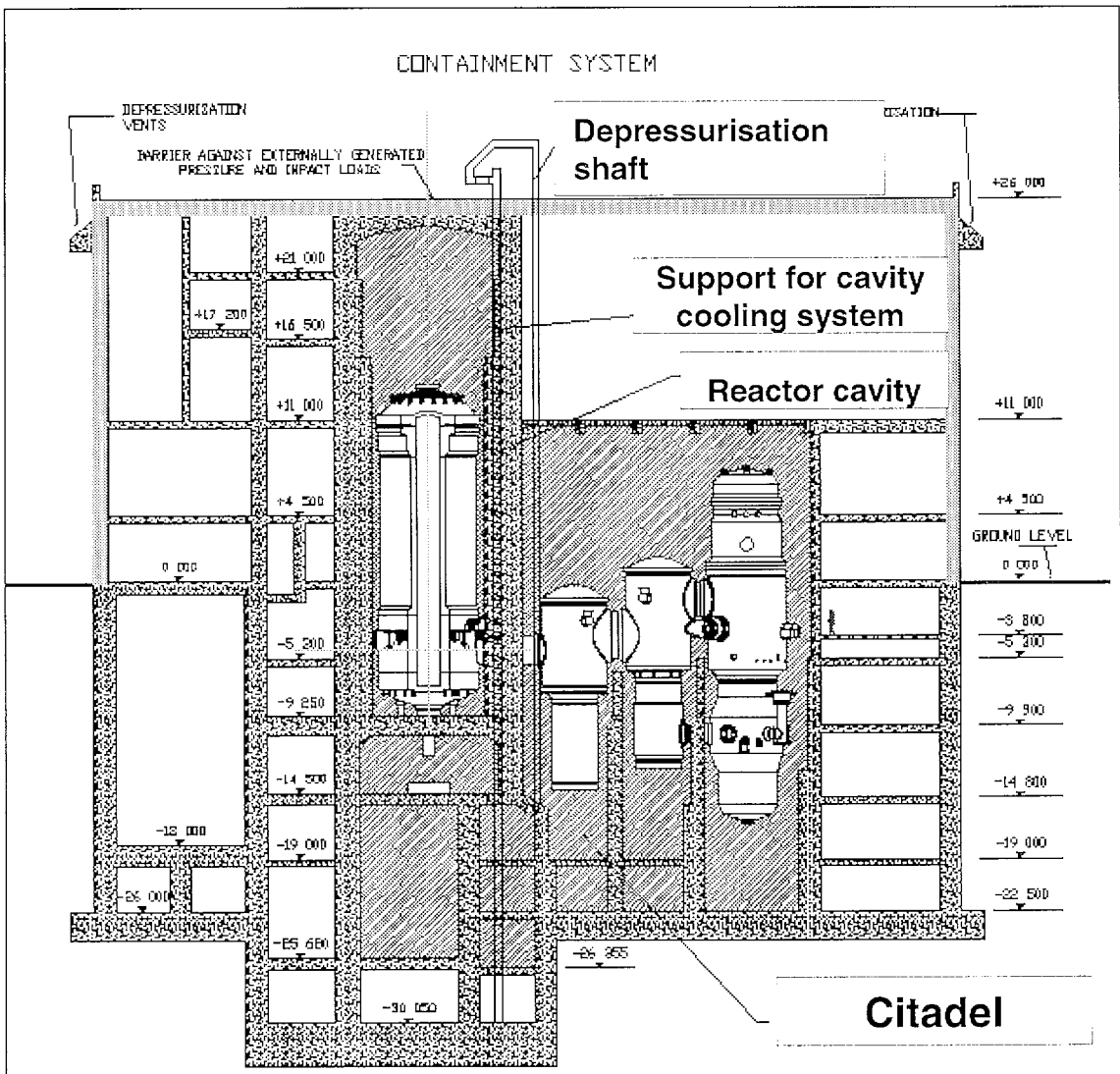
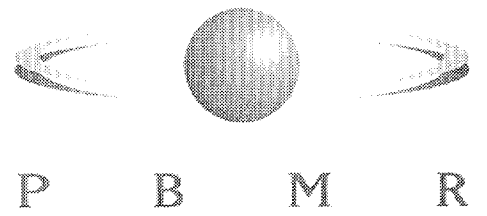
- During normal operation helium pressure always higher than the water in the secondary heat exchanger
  - Tube leaks result in helium blowdown of water thru secondary relief systems
- During depressurized shutdown events (e.g., maintenance at 1 atm), the Reactor Unit Conditioning System (RUCS) heat exchanger will be at a higher pressure
  - RUCS cools the core to below the graphite oxidation temperature
  - Water-graphite reaction is endothermic
  - A tube break results in water draining to the bottom of the RUCS vessel below the core
  - RUCS water inventory limited—if all hypothetically reacted, negligible core graphite reacts (<.001)



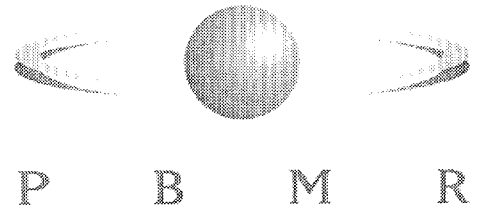
# PBMR Resistance to Air Ingress



- Air ingress events are infrequent---not expected in plant lifetime
- Helium pressure boundary (HPB) designed to ASME standards
- Citadel provides protection from external events
- Nuclear grade graphite blocks undergo limited air oxidation relative to other graphite and carbon forms
  - Reduced impurities limit catalytic and other oxidation enhancing effects
  - Electrode blocks of higher impurity which are more susceptible to oxidation at 500-600°C are routinely cooled in air during manufacturing
- Air supply limited by citadel volume

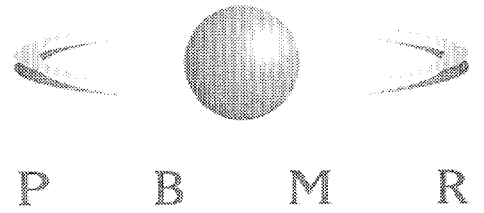


PBMR Citadel  
Provides External  
Protection  
and Limits Air  
Supply



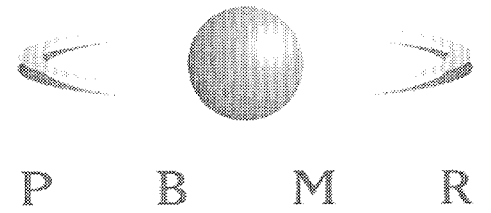
# Air Ingress Design Basis

- Helium Pressure Boundary(HPB) breaks in design basis-- not expected within lifetime of fleet of plants
  - Instrument lines (<10mm)
  - Fuel Handling and Storage System (FHSS) lines (<65mm)
  - Helium Inventory and Control System (HICS) lines (<65mm)
- Isolation of HPB possible depending on break location
  - Automatic or remote manual, if within FHSS or HICS
  - Remote manual, if within Power Conversion Unit HPB
- Reactor Pressure Vessel (RPV) designed with no piping above core support, ASME pressure vessel closures --- lighter helium prevents air ingress to core



## Air Ingress Design Basis (cont.)

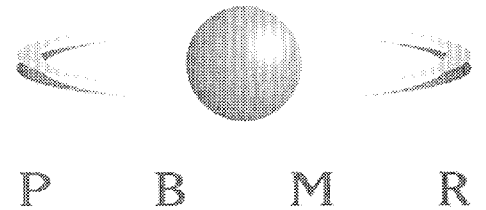
- Breaks  $<10\text{mm}$  ( $78\text{ mm}^2$ ) designed to vent slowly (hrs) through HVAC filters to environment
- For breaks  $<10\text{mm}$ , negligible air ingress and graphite oxidation
  - Opening too small (flow resistance)
  - Helium egress prevents air ingress
  - SBS and RUCS if available designed to cool core to below corrosion temperatures
  - If no action taken and entire RPV filled with air (80% is inactive nitrogen),  $<.00005$  of reactor graphite oxidized



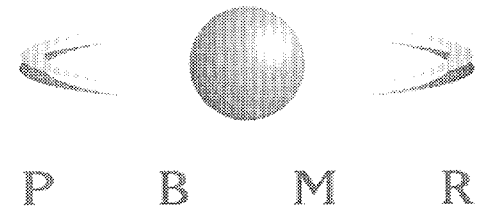
## Air Ingress Design Basis (cont.)

- Breaks <65mm (3318 mm<sup>2</sup>) designed to vent quickly (minutes) to environment thru containment system rupture disc with damper reclosure
- For breaks <65mm in RPV, insignificant graphite oxidation
  - Helium depressurization and core heatup cause outward expansion of helium for several days
  - Reaction is exothermic, but small contributor relative to decay heat
  - If no action is taken, as conduction cooldown to RCCS progresses, contraction of helium within HPB will result in air ingress to reactor
  - However, air ingress is limited by two moles (CO<sub>2</sub> or CO) forming for every mole of air reacting
  - Heated air slows down flow due to increased flow resistance
  - If entire citadel air supply hypothetically entered, <.002 of reactor graphite oxidized

# PBMR Resistance to Large Air Ingress



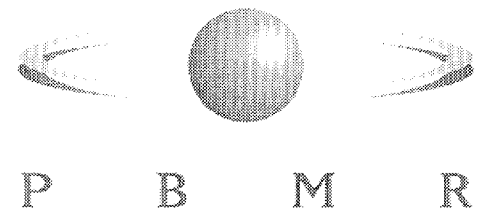
- Beyond the design basis---not expected within the lifetimes of a fleet of plants
- Breaks of between 65 and 170 mm (area 1330 – 23000 mm<sup>2</sup>) designed to vent thru blow out panels in top of citadel
  - Depending on location of break, two way flow through large breaks is conceivable
  - Depending on location of break, air transport to and through reactor core is possible
  - If no mitigative measures taken (e.g., blocking blowout panels in top of citadel) and entire reactor building supply of air entered, <.01 of reactor graphite oxidized, .07 of spheres or .12 of fuel free graphite in spheres
- Even with large amounts of local core oxidation, radionuclide retention is expected to be maintained within the ceramic fuel particles



# Air Ingress Research

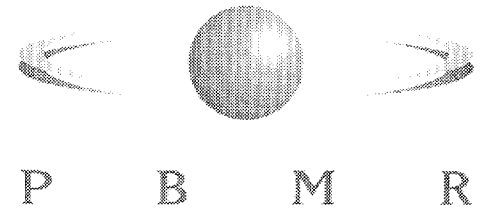
- International air oxidation tests have been performed (e.g., Veluna, Nacok), but provide limited insights
  - Non-representative core geometry, mass flow resistance, and reactor temperature distributions
  - Require top-bottom multiple openings or idealized failures to intentionally optimize natural convection
- Better strategy is to balance prevention measures within the design basis with a range of potential mitigative measures given the large times available for external actions

# Mitigation Strategies



- Given building is filled with helium-air-radionuclide mixture, conditions provide possibility of manned but contamination protected entrance
  - External dose rates (after 12 hrs) < 100  $\mu\text{Sv/hr}$
- Alternatively, remote external actions may also be possible
- Objectives are to to block leak with simple means and to slowly add inert gas to building, citadel, and/or core

# SUMMARY



- Limited water ingress potential leading to insignificant damage of graphite components
- Negligible air ingress through openings <10mm
- Air circulation through openings <65mm only after the core cools down with negligible public health impact
- Large HPB breaks beyond the design basis have acceptable risk
  - Extremely unlikely due to the design and choice of materials
  - Time available to take mitigating action before significant corrosion
  - Temperatures do not rise above the level that coated fuel particles are unable to retain radionuclides

# Pebble Bed Modular Reactor High Temperature Materials

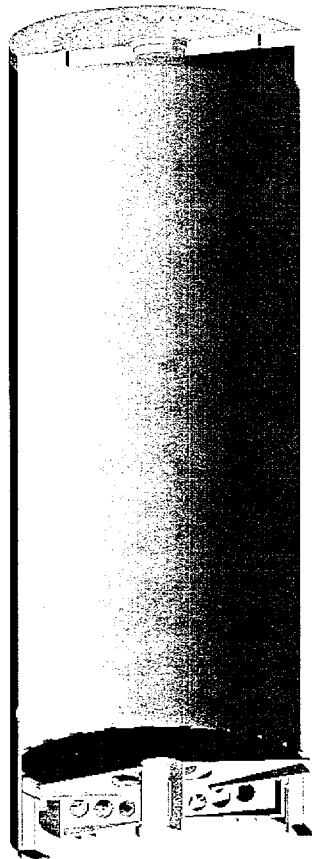
Mark Davies  
October 2001

# High Temperature Materials

- Core Barrel
- Control Rods
  - Chain
  - Segments
  - Secondary Shock Absorber
- Carbon-Carbon Composites

# CORE BARREL AND SUPPORT STRUCTURE

## Material: Type 316 Stainless Steel

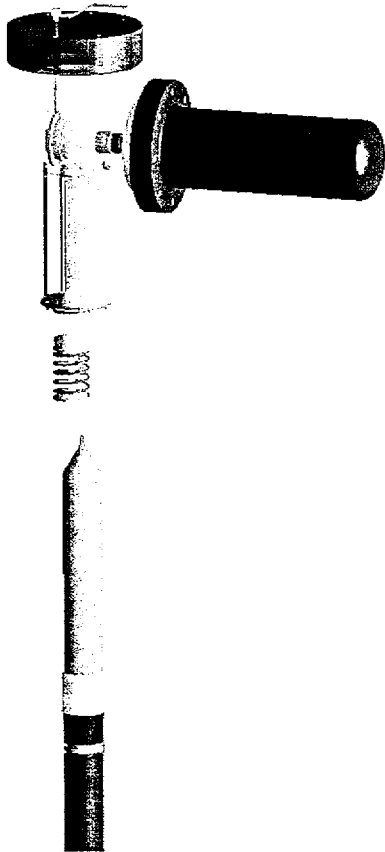


- Operating Temperature: ~490°C
- Abnormal Temperatures:
  - PLOFC, 590°C max
  - DLOFC, 711°C max
- Fast Fluence ( $E > 0.1 \text{ MeV}$ ):  
 $1 \times 10^{19} \text{ n/cm}^2$ .
- Design Code: ASME III, Subsection NG and ASME Code Case N-201.
- This material is qualified for use in accordance with ASME III, Subsection NG (Core Support Structures) at service temperatures up to 429°C and for service temperatures between 429°C and 816°C by ASME Code Case N-201 for a design life up to 300 000hrs.

# REACTOR CONTROL RODS

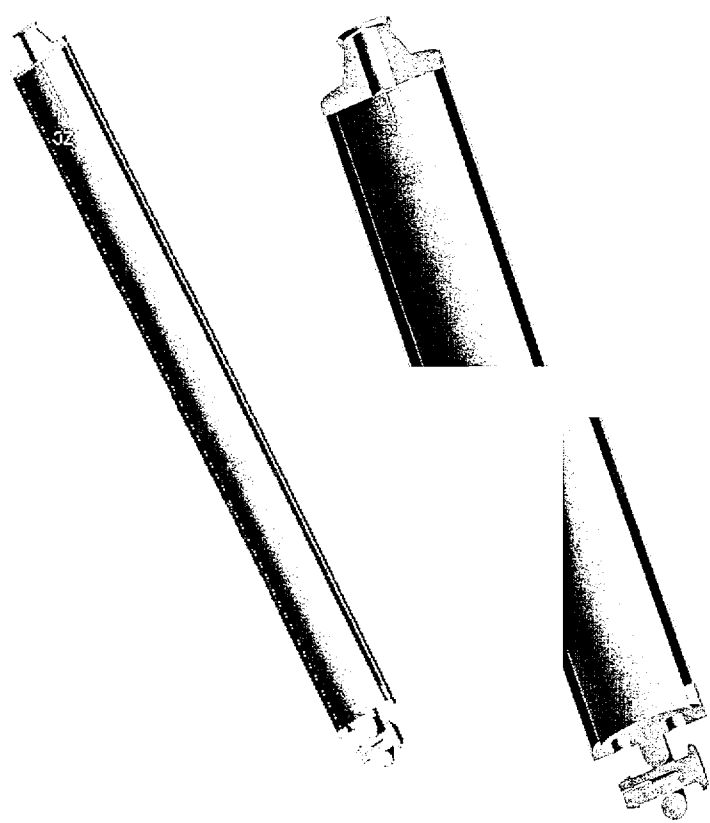
Material: Incoloy 800H

## Chain



- Operating Temperature: ~330°C
- Abnormal Temperatures: PLOFC, 870°C max:  
DLOFC, 900°C max
- Fluence (Thermal):  $<5 \times 10^{21} \text{ n/cm}^2$
- ASME III, subsection NH (Class 1 Components in Elevated Temperature Service) has qualified the use of Incoloy 800H to temperatures of up to 760°C, for service periods of up to 300 000hrs.
- Design data for temperatures up to 900°C and service periods of 300 000hrs as well as data for temperatures up to 1100°C and service periods of 100 000hrs have been provided in the guideline KTA 3221- Metallic HTR Components (in draft).
- The material's response to irradiation effects have been characterized in the German HTR development program. The test results envelop the following conditions:  
  
Hot tensile testing (at temperatures between 400°C and 900°C) of samples irradiated at 400-600°C, to a fluence of  $3 \times 10^{21} \text{ n/cm}^2$  (thermal and fast).

# REACTOR CONTROL RODS 1

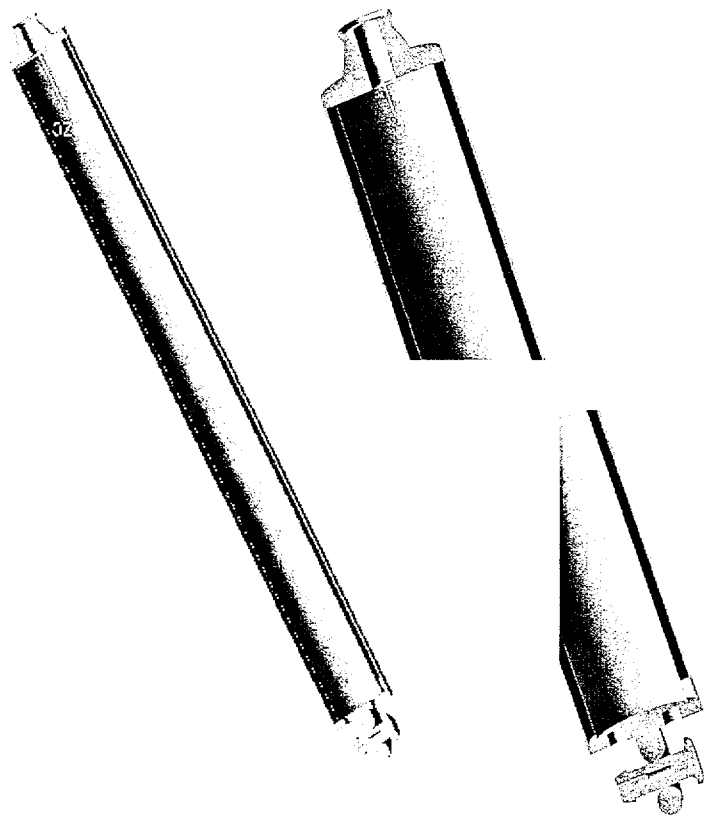


## Segments

Material: Incoloy 800H

- Operating Temperature:~650°C
- Abnormal Temperatures:  
    PLOFC, 926°C max  
    DLOFC, 1100°Cmax
- Fluence (Thermal):  
    5x10<sup>21</sup>n/cm<sup>2</sup>

# REACTOR CONTROL RODS 2



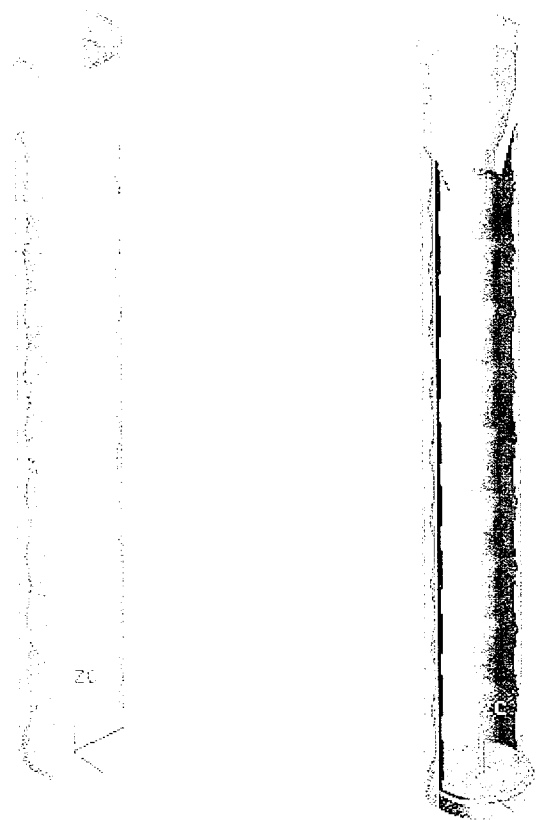
## Segments

**Material:  $B_4C$**

- Operating Environment: as above
- Density:  $2.51 \text{ g/cm}^3$
- Melting Point: around  $2,450^\circ\text{C}$
- Thermal expansion coefficient:  $5 \times 10^{-4} \text{ }^\circ\text{C}^{-1}$
- High resistance to chemical attack
- High thermal neutron absorption cross section  $\sim 4,000$  barns.
- Crystal structure: Rhombohedral
- Very high hardness: It is third hardest material next to diamond and cubic boron nitride (cBN).

# REACTOR CONTROL RODS

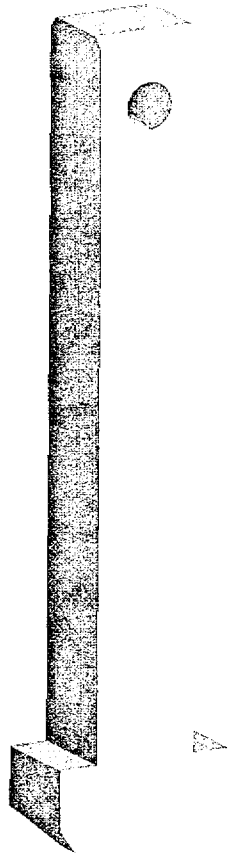
Material: Incoloy 800H



## Secondary Shock Absorber

- Operating Temperature:~900°C
- Abnormal Temperatures:  
    PLOFC, 1100°C max  
    DLOFC, 1100°Cmax
- Fluence (Thermal):  
    5x10<sup>21</sup>n/cm<sup>2</sup>

# Carbon-Carbon Composites



## Carbon Composites

- Top Reflector Tie rods
  - Required safety factor >20
  - Fast neutron dose to base of rod =  $1.56 \times 10^{20} \text{ n/cm}^2 \text{ EDN}$
  - Graphite Temperature = 1200 °C
- Restraints
  - Graphite Temperature = 1000 °C