September 16, 2010

Slides Presented at September 1-2, 2010 Next Generation Nuclear Plant (NGNP) Public Meeting

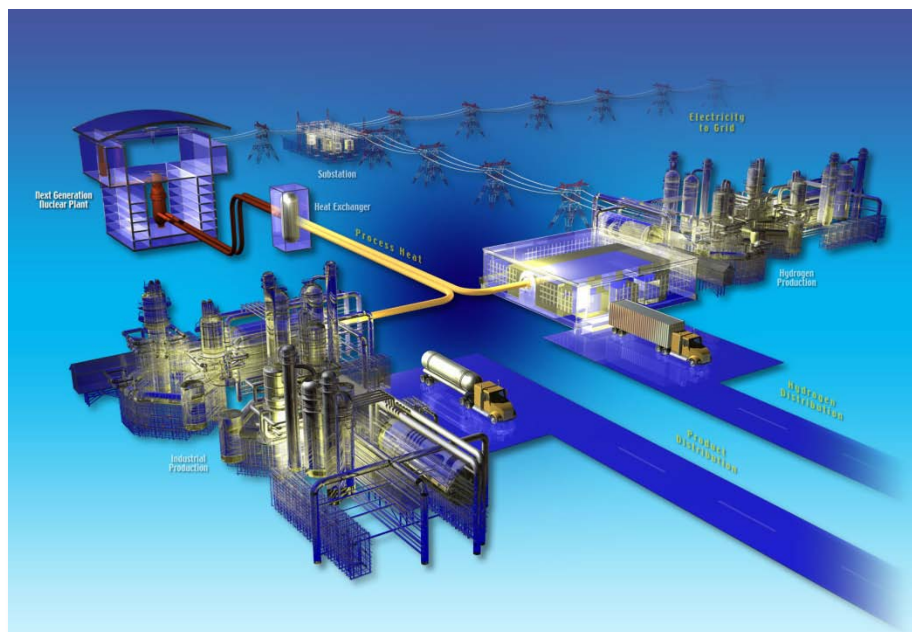
White Papers on High Temperature Materials, Fuel Qualification, and Mechanistic Source Term

NGNP Next Generation Nuclear Plant

NGNP High Temperature Materials

Presentation to NRC Staff by Next Generation Nuclear Plant Project

September 1, 2010



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White Paper:
INL/EXT-09-17187
Next Generation Nuclear Plant (NGNP) High Temperature Materials White Paper

June 2010

ML 101800221

Next Generation
Nuclear Plant

Overview

- Introduction
- Purpose and Objectives
- Regulatory Basis
- Material Selection and Qualification
  - Selection Approach
  - Design Considerations
- Metallic Material
- Graphite Material
- Ceramic Material
- Composite Material
- Outcome Objectives
- Q&A
- Public Comment
- Adjourn

Introduction

- High Temperature Gas-cooled Reactor (HTGR) concepts currently under consideration for NGNP include both pebble-bed and prismatic-block reactors
- The design of the HTGR is in the initial Conceptual Design Phase
  - Final component specification and material selection has yet to be performed
- The scope of the High Temperature Materials White Paper is to review the existing policies, regulations, and guidance associated with acceptance of materials for nuclear reactor applications and to assess the basis for implementation in the system components for the HTGR
- Principal materials proposed for application in the NGNP primary system are identified, along with the proposed approaches for establishing regulatory compliance

Primary Objectives

- Summarize existing regulatory policies and guidance
- Describe an approach for selecting materials, identifying properties, qualification, and accepting materials
- Describe influence that material selection and code requirements may have on licensing basis events, including design basis accidents
- Discuss any needed codes and standards work, including status of activities already in progress
- Identify policy and technical issues that should be discussed and resolved

Outcome Objectives – Metallic Materials

Does application of the ASME Code (including code cases) for metallic materials provide a reasonable basis for the design and qualification of components, such as for:

- ASME Code Case N-201-5
  - Core support structure for temperatures above 371 C
  - Materials: Type 316H SS, 2.25Cr-1Mo and Alloy 800H
- ASME Code Section III, Subsection NH
  - Extended temperature limits
  - Materials: Alloy 800H, Type 316H SS, and Mod 9Cr-1Mo, and 2.25Cr-1Mo
- Acceptability of the extended role of materials in HTGR safety case
  - Role of material properties in HTGR safety case
  - HTGR reliance on material properties for passive safety features
- Acceptability of alternate material qualification path
  - Qualification by review of materials analysis packages
  - Qualification of materials for specific HTGR components use

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Outcome Objectives – Nonmetallic Materials

- Does emerging ASME Code for graphite provide a reasonable basis for design and qualification of components
- Does experimental program characterizing the fluence/temperature response of graphite provide reasonable basis for licensing the initial operation of a HTGR lead plant
- Does Reliability and Integrity Management (RIM) Program provide reasonable basis for assessing the condition of graphite components in service
- Does application of the requirements for design and manufacture of the ceramic internals provide a reasonable basis for the design and qualification of the core structure ceramic (CSC)
  - That is, other ceramic components (insulation and carbon fiber reinforced carbon (CFRC))
- Is selection of CFRC reasonable for certain structural components

Regulatory Basis

Regulatory Basis

- In most cases, current NRC regulations are applicable for metallic materials of the NGNP plants
- Nonmetallic materials not fully covered in current regulations
- Where interpretation and application are required, consider
  - Differences between the safety functions of the HTGR and the LWR technologies
  - Inherent characteristics and passive capabilities of HTGRs
- ASME Code development is underway that will require further NRC review and approval that will support HTGR design application

Regulatory Basis (cont)

- 10 CFR 50, for example:
  - 10 CFR 50.55a, “Codes and Standards”
- NRC Regulatory Guides, for example:
  - Regulatory Guide 1.84, “Design, Fabrication, and Material Code Case Acceptability”
  - Regulatory Guide 1.87, “Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors”
- Other guidance documents
  - Standard Review Plan (NUREG-0800)
  - NUREG/CR-6824 “Materials Behavior in HTGR Environments”

Regulatory Precedents

- Peach Bottom 1, Fort St. Vrain, Large HTGRs
  - Similarities – Fuel form, Helium coolant, He-Water SG in primary system
  - Many lessons learned from graphite use and properties that will improve next generation of HTR graphite
    - Graphite types used in these previous designs (e.g., H-451) no longer available

- MHTGR, GT-MHR, and Pebble Bed Modular Reactor (PBMR) pre-application
  - NUREG -1338 MHTGR draft SER issued
  - 2001 NRC pre-application review of Exelon (PBMR) submittals

Phenomena Identification and Ranking Table (PIRT) Workshops

- 2007 PIRT workshops
  - NUREG/CR-6944 *Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)*, 2008
  - Vol. 4 – High Temperature Materials
  - Vol. 5 – Graphite
- 2009 follow-on workshop on graphite
- NGNP Research and Development Program Plan

ASME Code Development

- Working Group on Nuclear High Temperature Gas–Cooled Reactors
  - Develop rules for HTGRs within Section III, Division 5
- Roadmap for the Development of ASME Code Rules for High Temperature Gas Reactors (draft)
- Key committees that directly support HTGRs in areas relevant to materials include:
  - Subgroup on High Temperature Reactors
  - Subgroup on Elevated Temperature Design
  - Subgroup on Graphite Core Components
  - Special Working Group, High Temperature Gas-Cooled Reactors (Section XI)

Extensive ASME ST LLC Efforts

- Task 1 - Allowable Stresses in Section III, Subsection NH on Alloy 800 H and Grade 91 Steel
- Task 2 - Regulatory Safety Issues in Structural Design Criteria
- Task 3 - Improvement of Subsection NH Rules for Grade 91 Steel
- Task 4 - Updating Nuclear Code Case N-201
- Task 5 - Creep-Fatigue Data and Evaluation Procedures for Grade 91 Steel and Alloy XR
- Task 6 - Operating Condition Allowable Stress Values
- Task 7 - ASME Code Considerations for the IHX
- Task 8 - Creep and Creep-Fatigue Crack Growth
- Task 9 - Update NH - Simplified Elastic and Inelastic Methods
- Task 10 - Update NH - Alternative Simplified Creep-Fatigue Design Methods
- Task 11 - New Materials for NH
- Task 12 - NDE and ISI Technology for HTRs (Funded by NRC)
- Task 13 - Recommend Allowable Stress Values (awarded in 2010)
- Task 14 - Corrections to Stainless Steel Allowable Stress (awarded in 2010)

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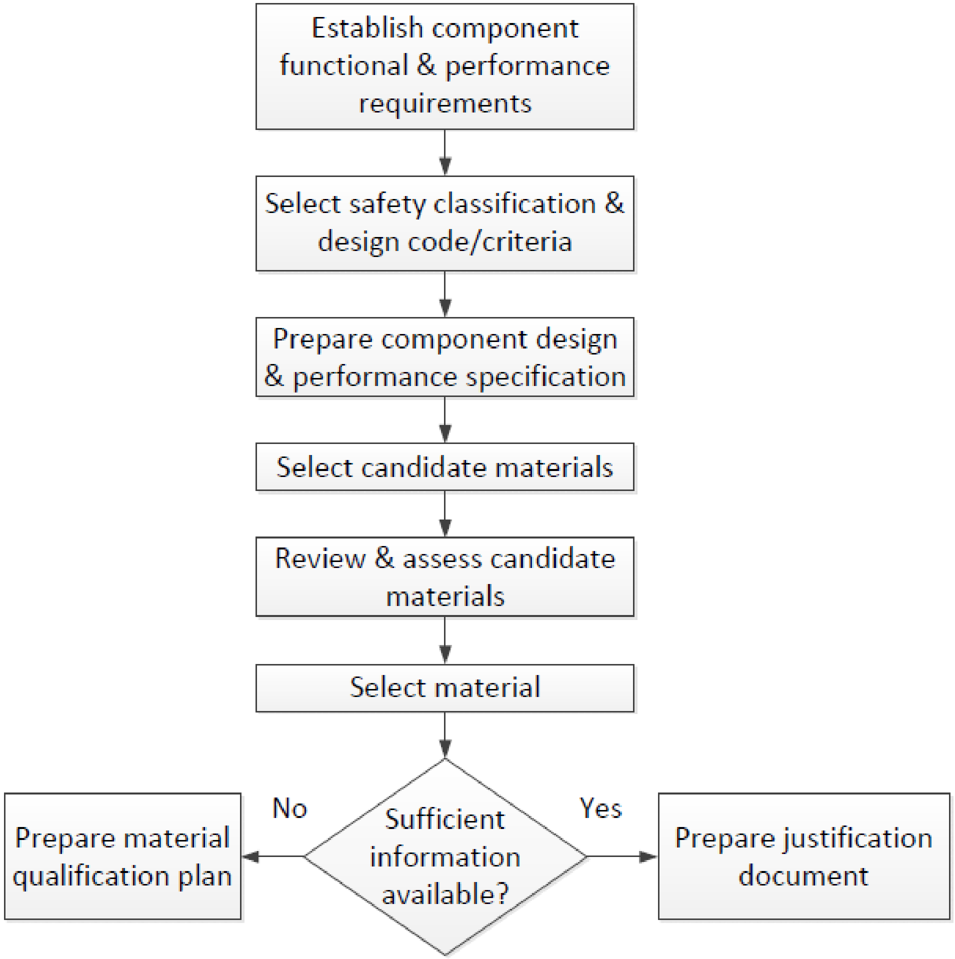
Material Selection and Qualification

Selection Approach



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Design Considerations - Metallics

- Availability of materials
  - Commercially available
  - Current application in nuclear or non-nuclear industries
- Suitability of material
  - Material properties at applicable temperatures
  - Helium gas service conditions
- Have these materials been characterized to the extent necessary for the specific HTGR component application
- Is material codified and/or standardized
  - Codes (ASME, RCC-M), Standards (ASTM, KTA)
- The manufacturability issues
  - Forging size
  - Thick section welding issues and performance
  - Inspection requirements

Design Considerations – Metallic (cont)

- Is material suitable/qualified for nuclear application
  - ASME Section III
  - NRC review and acceptance RG 1.84
  - Qualification by analysis and/or testing
- Material selection process must consider
  - Required properties at normal and accident reactor conditions
  - Operating environment
  - Size, form, availability, cost and manufacturability
- For operations at higher temperatures
  - Material strength becomes an important factor
  - Creep becomes an issue
  - Operating environment; helium with impurities becomes a factor
  - Creep-fatigue, environment, fluence interaction is more prominent
  - Change from metals to ceramics or composites might be considered

Design Considerations – Metallic (cont)

- For component design the following factors must be identified and considered:
  - Key system or component functions
    - Normal system functions
    - Safety functions
  - Anticipated operational environment (normal and accident)
    - Temperatures
    - Loadings
    - Fluences
    - Chemistry
  - Important issues with respect to high temperature material performance, examples:
    - Material emissivity for RV
    - High temperature fatigue for control rods

HTGR Components

- Primary system vessels
- Hot duct
- Reactor internals
- Shutdown cooling system
- Helium purification system piping and valves
- Main circulator
- Heat exchangers
  - Steam Generator (He to steam/water)
  - Reactor cavity cooling system exchanger (air to water or air to air)
  - Shutdown cooling system heat exchanger (He to water)
  - Core conditioning system heat exchanger (He to water)
- Helium pressure boundary piping
- Helium high temperature valves
- Vessel safety relief valve

Metallic Materials

Metallics - Candidate Materials

- Potential near term HTGR concepts
  - SA-508 Grade 3 Class 1 / SA-533 Type B Class 1
  - Type 316H Stainless Steel
  - Alloy 800H
  - Alloy X
  - Modified 9Cr-1Mo Grade 91
  - 2.25Cr-1Mo Grade 22
- Long term concepts (Advanced HTGRs or VHTR)
  - Alloy 617
  - Alloy 230
  - Mod 9Cr-1Mo Grade 91 (for vessel material)
  - Alloy XR
- Other metallics may be identified as design progresses

SA-508 Grade 3 Class 1 / SA-533 Type B Class 1

- Potential applications
  - RV, SG vessel and cross vessel
- Important considerations
  - Strength, creep fatigue, emissivity, thermal diffusivity, oxidation and irradiation
- Related experience
  - Pressure boundary components in LWRs for over 40 years
    - Historically used with success at operating temperature of 325°C
- Current qualification status
  - NRC approved and ASME Code Section III, Subsection NB (< 371°C limit for normal operation)
    - Current HTGR objective calls for vessel operating temperature within LWR operating envelope due to industry concerns with operating close to the 371°C limit for long durations
  - ASME Code Case N-499-2 approved for limited high temperature excursions up to 427°C for 3000 hours and 538°C for 1000 hours
  - Irradiation effects addressed by 10 CFR 50 and detailed in Reg. Guide 1.99
  - Further evaluation: Emissivity and oxidation

Type 316H Stainless Steel

- Potential applications
  - Metallic internals with temperature above 650°C during normal operation or accident
    - Core support structures, including core barrel
- Important considerations
  - High temperature strength, creep strength, irradiation resistance, relative low cost
- Related experience
  - The austenitic stainless steels of Type 304 and Type 316 are commonly used for light water reactor internals, such as fuel support structures, core barrel and flow baffle plates. These are however all low temperature applications, in aqueous conditions and materials used all comply to the low-carbon versions i.e. Type 304L or Type 316L.
- Current qualification status
  - Maximum temperature limit
    - 427°C Section III (time independent stress limit)
    - 816°C Section III Subsection NH (Elevated temperature Class 1)
    - 816°C Section VIII (Non-nuclear pressure vessels)
    - 816°C Code Case N-201-5 (Core support structure)
  - Further evaluation: High temperature strength, creep strength of large grain products (relevant to large forgings), irradiation effects

Alloy 800 H

- Potential Applications
  - SG, control rod sleeves
- Important considerations
  - High temperature strength, time dependent stress effects, irradiation dose
- Related experience
  - SG tubing and heat exchanger components at FSV, AVR and THTR
    - Cumulative operation of 34 years, with AVR operating 20 years
- Current qualification status
  - Maximum temperature limit
    - 427°C Section III (time independent stress limit)
    - 760°C Section III Subsection NH (Elevated temperature Class 1)
    - 899°C Section VIII (Non-nuclear pressure vessels)
  - Some internals exceed limit during normal operation or accident
    - Draft German standard KTA 3221 allows use up to 1000°C
    - Joint ASME & DOE effort to obtain and use this data to increase Code allowable temperature is underway
  - Further evaluation: Temperature limit, emissivity, oxidation and irradiation

Alloy X

- Potential applications
  - Metallic internals with temperature above 750°C during normal operation or accident
    - Control rods, CR guide tubes, upper plenum shroud and hot duct liner
- Important considerations
  - High temperature strength, time dependent stress effects, irradiation, dose
- Related experience with Alloy XR
  - Japanese HTTR hot duct liner and IHX tubing for over 10 years
    - Material used with success at operating temperature of 850°C with excursions up to 950°C
- Current qualification status
  - Alloy X
    - Maximum temperature limit of 427°C for Section III and 899°C for Section VIII Division 1 (Guidance)
    - Industry experience reports the potential for a normal operating limit of 871°C and abnormal condition limit of 938°C for <3000 hours and <1000 psi stress, but this will need to be supported by QAed references
    - Cobalt (0.5-2.5 wt%) could potentially create high dose issues for Hastelloy X

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Modified 9Cr-1Mo (Grade 91)

- Potential applications
  - Metallic core support structure, including core barrel
- Important considerations
  - High temperature strength, time dependent stress effects, emissivity, thermal diffusivity, irradiation effects and corrosion resistance
- Related experience
  - Tubing in super-heaters of power boilers at about 600°C for over 20 years
  - Extensive studies on high temperature and time dependent properties
    - Twice strength of 2.25Cr-1Mo at 500°C
- Current qualification status
  - Maximum temperature limit
    - 371°C Section III NG (time independent stress limit)
    - 650°C Section III Subsection NH (Elevated temperature Class 1)
  - Ideally Code Case N-201-5 would be expanded to consider Mod 9Cr-1Mo
    - Currently incorporated into proposed Div 5
  - Further evaluation: Emissivity, irradiation and oxidation effects

2 ¼ Cr – 1Mo (Grade 22)

- Potential applications
  - Cold end SG tubing, Core support structures
- Important considerations
  - High temperature strength, time dependent stress effects, thermal conductivity, corrosion resistance
- Related experience
  - Japanese HTTR reactor and heat exchanger vessels, which have operated at about 400°C for over 10 years.
  - FSV and THTR cold end SG tubing
  - Extensive studies on high temperature and time dependent properties
    - Comparable strength to Mod. 9Cr-1Mo up to about 430°C
- Current qualification status
  - Maximum temperature limit
    - 371°C Section III (time independent stress limit)
    - 593°C (300,000 hrs) Code Case N-201-5 (reactor internals)
    - 650°C (1000 hrs) Code Case N-201-5 (reactor internals)
    - 650°C Section III Subsection NH (Elevated temperature Class 1 1000 hrs)
  - Further evaluation: none - well characterized material

Long Term Concepts

- Alloy 617, Alloy 230
  - Potential IHX material
  - Work is underway at INL to characterize the high temperature performance of Alloy 617

- Modified 9Cr – 1Mo
  - Potential vessel material
  - Large forging availability
  - Thick section welding

Metallic Materials

- In present NGNP concepts, primary helium pressure boundary vessels, including reactor vessel will employ conventional metallic materials that are currently approved for nuclear service within ASME Section III
- Reactor vessel could operate in the region of negligible creep during normal operation and for anticipated operational occurrences that are expected within lifetime of a plant
  - Potential exists that the temperature limits of Section III, Subsection NG would be exceeded for short periods of time during low frequency design basis events involving conduction cooldown
  - Apply ASME Code Case N-499-2, for short-term operation and limited frequency

Current ASME Code Limits

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Alloy	Applicable ASME Code Section	Prescribed Limits
SA-508/SA-533	Section III, Subsection NB, NC, ND	371°C (700°F)
SA-508/SA-533	Section III, Subsection NB Code Case N-499-2	371°C (700°F) to 427°C (800°F) for 3,000 hours (Level B) 427°C (800°F) to 538°C (1000°F) for 1,000 hours (Level C or D) Maximum of 3 events over 427°C
SA-508/SA-533	Section VIII, Division 1	427°C (800°F)
316 SS	Section III, Subsection NH	816°C (1500°F)
316 SS	Section III, Subsection NB, NC, ND	427°C (800°F)
316 SS	Section III, Subsection NG with Code Case N-201-5	816°C (1500°F)
Alloy 800H	Section III, Subsection NB, NC, ND	427°C (800°F)
Alloy 800H	Section III, Subsection NG with Code Case N-201-5	760°C (1400°F)
Alloy 800H	Section III, Subsection NH	760°C (1400°F)
Alloy 800H	Section VIII, Division 1	899°C (1650°F)
Alloy X	Section III, Subsection NB, NC, ND	427°C (800°F)
Alloy X	Section VIII, Division 1	899°C (1650°F)
Alloy 617	Section VIII, Division 1	899°C (1650°F)
Modified 9Cr-1Mo	Section III, Subsection NB, NC, ND	371°C (700°F)
Modified 9Cr-1Mo	Section III, Subsection NH	649°C (1200°F)
Modified 9Cr-1Mo	Section VIII, Division 1	649°C (1200°F)
2¼ Cr–1 Mo, Grade 22	Section III, Subsection NB, NC, ND	371°C (700°F)
2¼ Cr–1 Mo, Grade 22	Section III, Subsection NG with Code Case N-201-5	593°C (1100°F)
2¼ Cr–1 Mo, Grade 22	Section III, Subsection NH	593°C (1100°F)
2¼ Cr–1 Mo, Grade 22	Section VIII, Division 1	649°C (1200°F)

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Metallic Materials

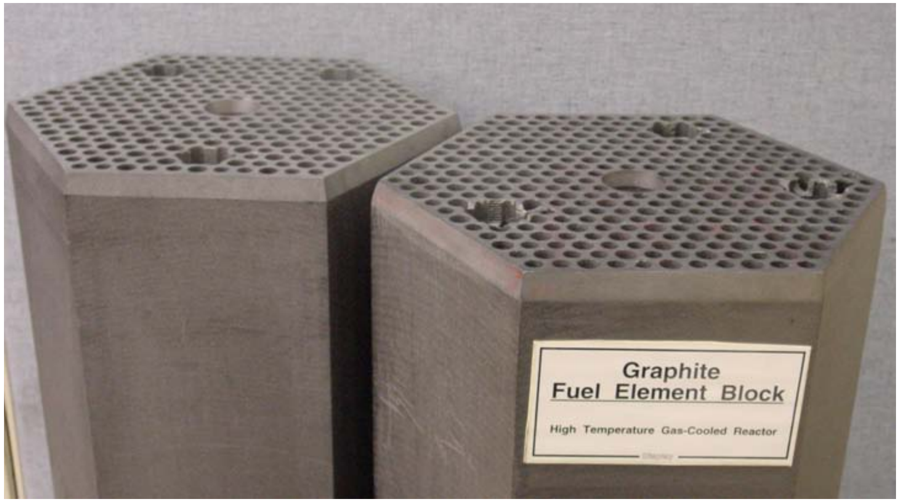
ASME Code activities

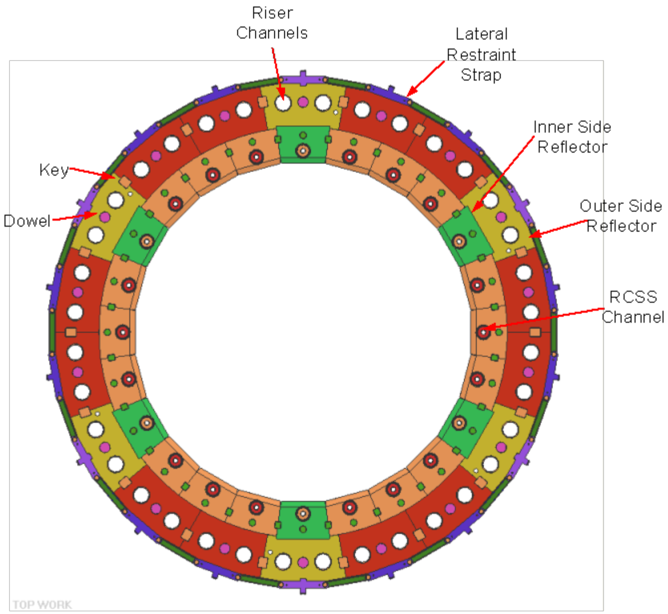
- SA 508/SA 533, Mod 9Cr- 1Mo, 2 ¼ Cr- 1Mo
  - Revisit 371 C (700 F) temperature limit
- Alloy 800H qualification under Section III, Subsection NH and Code Case N-201-5
  - Extend operating temperature above 800 C
- Include Alloy X/XR, and Alloy 617 in ASME Section III, Subsection NH and Code Case N-201-5

Additional activities

- Consideration on how to incorporate chemical environmental, thermal aging and irradiation effects on material properties into design particularly those components operating at high temperatures

Graphite Material





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Key Functions

- Contain and protect fuel
- Maintain core geometry
- Provide undisturbed access for the insertion of reactivity control material
- Passively transport core heat, primarily by radial conduction from the fuel to the core barrel, during off-normal events when forced cooling is not available
- Resist chemical attack by limiting oxidation for off-normal events involving ingress of water or air gas mixtures
- Absorb thermal energy during transients
- Neutron moderator

Key Requirements

- Structural design considerations
- Nuclear and thermal characteristics
- Service conditions during normal operation
  - Temperatures, pressures, flow rates
  - Normal operating coolant chemistry
- Neutron irradiation
  - Highly dependent upon location within reactor
  - Basis for design life of most highly irradiated components
- Service conditions during off-normal events

Graphite Materials - HTGR Experience Base

Feature	Dragon	Peach Bottom	AVR	Fort St. Vrain	THTR	HTTR	HTR-10
Location	UK	USA	Germany	USA	Germany	Japan	China
Power (MWt/MWe)	20/ -	115/40	46/15	842/330	750/300	30/ -	10/ -
Fuel Elements	Cylindrical	Cylindrical	Spherical	Hexagonal	Spherical	Hexagonal	Spherical
Graphite							
- Fuel Elements	- -	- -	- -	H-327/H-451	- -	IG110	- -
- Core Structures	?	?	ASR/AMT	HLM/PGX	PXA2N	IG110/PGX	IG-11
He Temp (In/Out,°C)	350/750	377/750	270/950	400/775	270/750	395/950	300/900
He Press (Bar)	20	22.5	11	48	40	40	20
Pwr Density (MW/m3)	14	8.3	2.3	6.3	6	2.5	2
Reactor Vessel	Steel	Steel	Steel	PCRV[1]	PCRV[1]	Steel	Steel
Operation Years	1965-1975	1967-1974	1968-1988	1979-1989	1985-1989	1998 -	1998 -

[1] Prestressed Concrete Reactor Vessel

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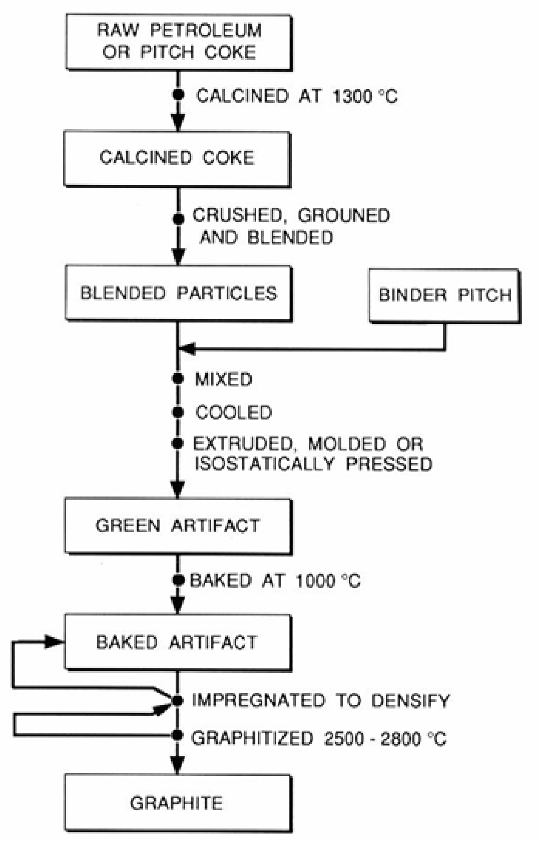
Other Experience Base

- Significant experience with graphite-moderated commercial power reactors was obtained with the U.K.-developed CO_2-cooled reactors
  - Magnox
    - 13 Magnox stations built in the U.K.
    - Additional stations built in Japan, Italy and France
  - Advanced Gas-Cooled Reactors (AGRs)
    - 7 AGR stations built in the U.K.

Manufacture

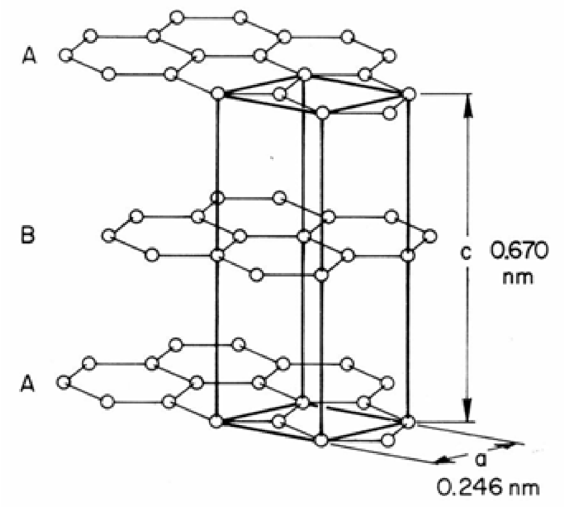


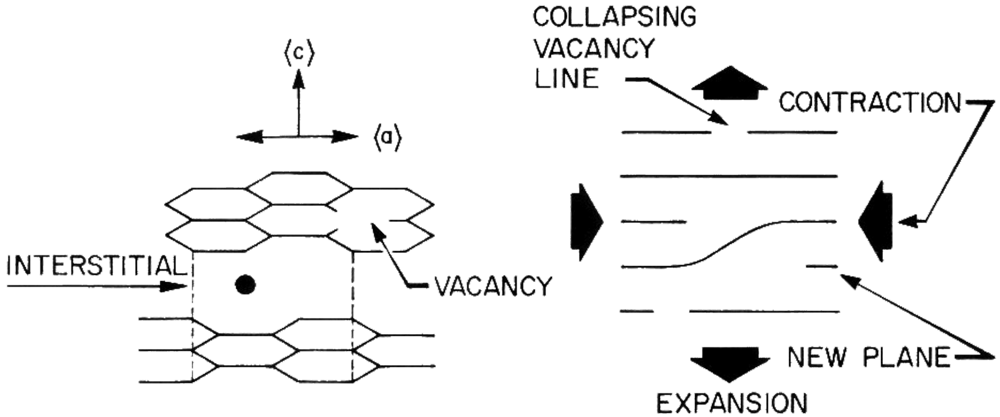
- Graphite is manufactured from calcined coke and a pitch binder
  - Multiple pitch impregnations to increase density
- Green forming technique influences the final microstructure
  - Want isotropic material response
- Properties and performance of graphite are significantly influenced by both raw materials and processing
- Nuclear graphite undergoes further purification steps

Irradiation Effects on Properties

- Important consideration in design of graphite-moderated reactors
- Significant changes occur in the following properties:
  - Dimensions
  - Strength and modulus
  - Thermal conductivity
  - Coefficient of thermal expansion
- Significant changes do not typically occur in the following properties:
  - Specific heat capacity
  - Emissivity

Graphite Structure & Irradiation Effects





Graphite Structure

Irradiation Effects

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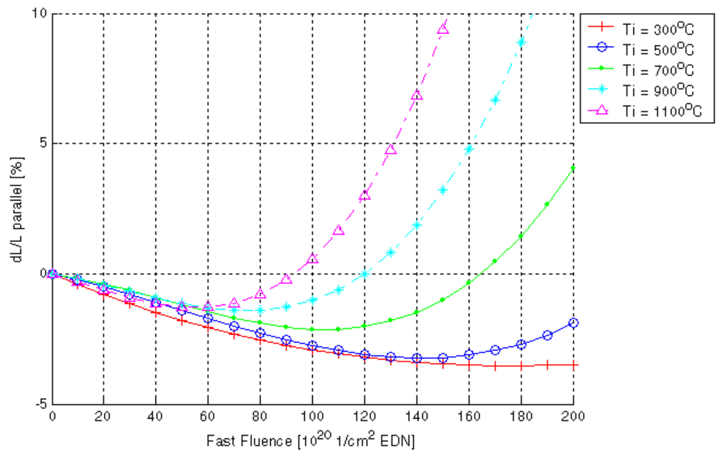
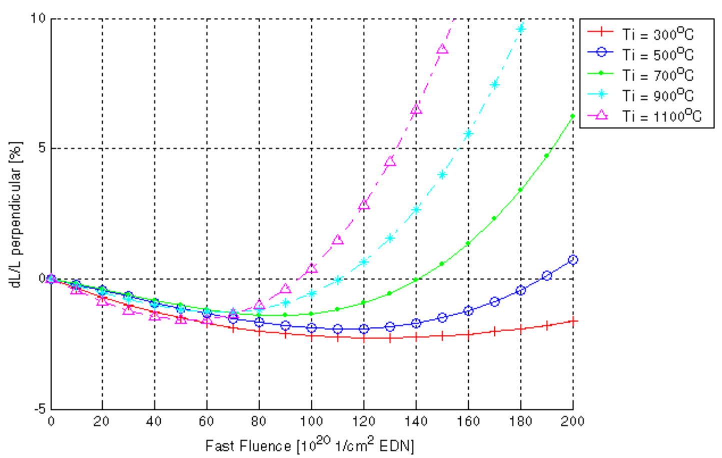
Dimensional Changes

- Microscopic cracks parallel to crystallographic planes (Mrozowski cracks) initially accommodate expansion in c-direction
  - Mainly a-direction contraction
  - Result: Graphite undergoes net volume shrinkage (**at first**)
- With increasing neutron dose, these cracks close due to c-direction expansion
  - Volume shrinkage rate falls, eventually reaches zero ("turnaround point")
  - Dimensional volume increase is rapid
- Rate of change is highly temperature dependent



Parallel Direction



Perpendicular Direction

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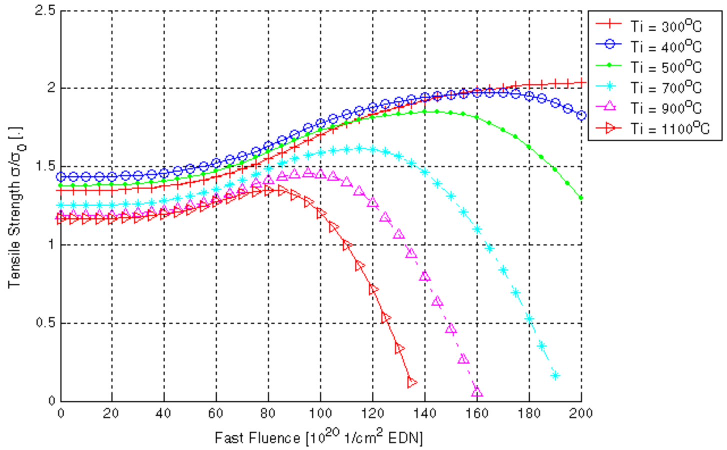
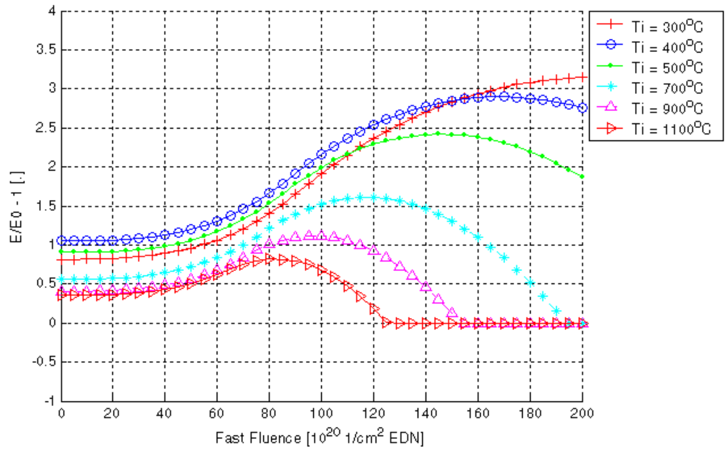
Strength/Modulus Changes

- Changes in strength and modulus somewhat parallel dimensional changes
- Strength/modulus initially increase
  - Maximum value is reached at approximately the turnaround point
- After turnaround pores start to form in microstructure.
  - As porosity forms, strength and modulus fall at increasing rate
- As with dimensional changes, strong dependence on irradiation temperature

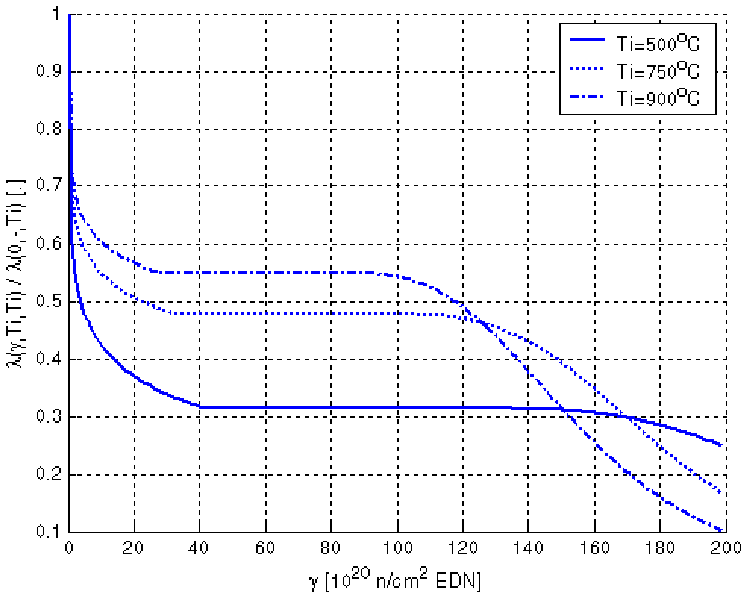


Tensile Strength



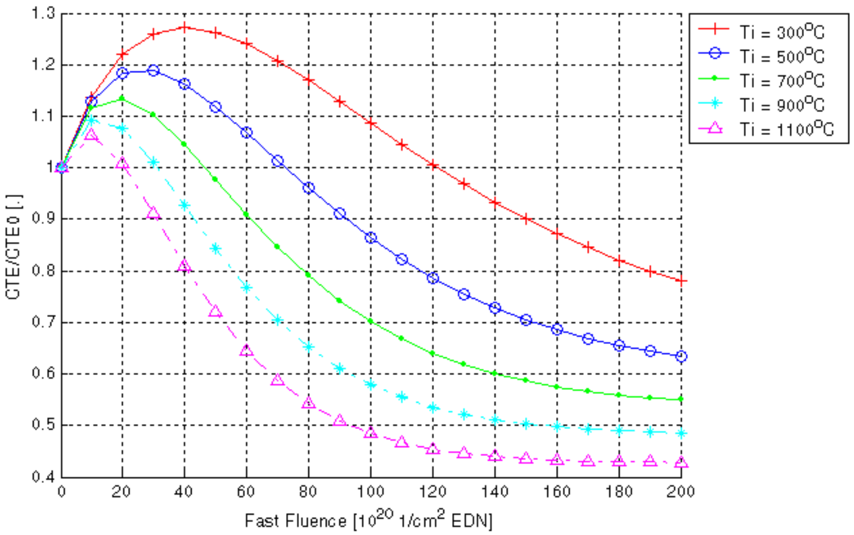
Modulus

Thermal Conductivity Changes



- Initial steep drop in conductivity followed by a saturation level
  - Point defect interruption of thermal conductance
- Pore generation after turn-around leads to further degradation
- At high operating temperatures irradiated and non-irradiated conductivity differences are small

Thermal Expansion Coefficient Changes



- Overall, graphite coefficient of thermal expansion (CTE) is low compared to other structural materials, e.g., metals
  - Implies potential for excellent shock resistance
- Important for design of the core and determining gaps between blocks

Irradiation Creep

- **Thermal creep strain** does not occur at normal reactor operating temperatures (<1200°C)
- **Irradiation-induced creep strain** reduces internal stresses resulting from dimensional changes
  - Results from dislocation movement due to irradiation
  - Creep strain rate generally increases with temperature
- The net effect is positive in that stresses associated with dimensional changes and differential thermal expansion under irradiation are reduced
- As the total fluence (dose) is increased, the ability to reduce internal stresses is increasingly important in attaining acceptable design lifetimes

Coolant Chemistry and Oxidation

- Normal Operation
  - Impurities within the helium coolant will have minimal effect on graphite components over the plant lifetime
- Off-Normal Events
  - Chemical attack/oxidation
  - Influence on component strength and, hence, structural integrity is not expected to be significant for events within the design basis

ASME Code for Graphite Core Components

- ASME Code for Graphite Core approved by ASME BNCS in early-2010
  - Developed by Section III Subgroup on Graphite Core Components
  - Expected to be published in 2011under Section III, Division 5 (High-Temperature Reactors)
- Key features:
  - Applies to fuel, reflector and shielding blocks, plus interconnecting dowels and keys; excludes fuel compacts and pebbles
  - Rules apply to both individual components and assemblies
  - Allowance of cracks in graphite components, provided that safety functions are retained
  - Design must account for statistical variations in graphite properties within billets and for different production runs
  - Design must account for irradiation effects on graphite properties

ASME Code for Graphite Core Components (cont)

Three methods are provided for assessing structural integrity

- Simplified Method
  - Simplified conservative method based on ultimate strength derived from Weibull statistics
- Full Analysis Method
  - Detailed structural analysis taking into account loads, temperatures and irradiation history
  - Weibull statistics used to predict probability of failure
  - Maximum allowable probability of failure defined for three Structural Reliability Classes, which relate to safety function
- Qualification by Testing
  - Partial or full-scale testing to demonstrate that failure probabilities meet criteria of full-analysis method

Operational Considerations for High Fluence

- Initial design life established conservatively, using probabilistic design methods consistent with the emerging ASME Code
- Components in high-fluence regions designed for replacement
- In early plants, material test reactor (MTR) data validating irradiation effects modeling to substantially lead actual plant operation
  - Design life to be appropriately adjusted as data become available
- Operational life of most highly irradiated components further evaluated via Reliability and Integrity Management (RIM) program

Reliability and Integrity Management

- Surveillance
  - Real-time confirmation of functions (e.g., core geometry, heat transport to RCCS, coolant chemistry) provided by normal operation
- Testing
  - Confirm ability to insert control rods (principal active function)
- In-service Inspection
  - Visual inspection of accessible areas during refueling or maintenance
  - Volumetric inspection - research presently ongoing
  - In-situ measurements
    - Microhardness – under development by JAERI for HTTR
    - Eddy current – employed successfully for crack detection in UK
    - Trepanning – employed in UK to assess critical area
- Maintenance
  - Prismatic: Reflector components replaced periodically during refueling
  - Pebble: Reflector components replaced as required

PIRT

- The most significant graphite phenomena identified through the 2007 PIRT evaluation were:
  - Irradiation effects on material properties
  - Consistency of graphite quality and performance over service life
- Theoretical models have been developed that appear to represent experimental data well; however, need to be:
  - Tested against data for new graphites
  - Extended to neutron doses and temperatures proposed for new HTGR designs
- Current and planned development efforts are responsive to these recommendations

Ceramic Materials

Potential Applications

Components are designed specific

- Ceramic insulation layer
- Typical placement is between lower levels of graphite bottom reflector/ core outlet plenum structure and underlying metallic core support structure
- To date, two classes of materials have been used in HTGRs
  - Baked carbon
  - Fused or sintered quartz

Functions and Requirements

- Functions
  - Maintain core geometry (support overlying graphite core structures)
  - Control the flow of heat to adjacent metallic components (e.g., core support structure)
- Requirements
  - Thermal conductivity
  - Environment and service requirements
  - Fluence (will be minimal for ceramic insulation)

Related Experience

- HTGRs Employing Baked Carbon Insulation
  - AVR
  - THTR-300
  - HTTR (ASR-0RB)
  - HTR-10

Ceramic Materials - Representative Properties

Property	Unit	ASR-0RB Carbon	NBC-07 Carbon	NBG-18 Graphite
Bulk Density	g/cm^3	1.6	1.7	1.87
Coefficient of Thermal Expansion	$\times 10^{-6}/°K$	4.4	4.6–4.8	4.5-4.6
Thermal Conductivity	$W.m^{-1}.K^{-1}$	10	4.9–5.0	140–145
Tensile Strength	MPa	17.8	15	20
Compressive Strength	MPa	50.4	138.5	77–78
Elastic Modulus	GPa	8.7	15.7	12
Ash Content	ppm	5000 max.	4100 max.	< 300 avg.

Design and Analysis Approach

- At present there is no applicable ASME Code
- Similarities to graphite suggest applying the newly-developed ASME Code for Graphite Core Structures
  - As with graphite, the Weibull distribution functions for strength would be experimentally determined
  - The reliability requirements associated with Structural Reliability Class 3 (SRC-3) would be applied
- Anticipate no significant irradiation-induced properties changes at fluence levels seen in service
- The Materials Qualification Plan for ceramic insulation to be developed dependant upon design

Composite Materials

Potential Applications

Components are design specific

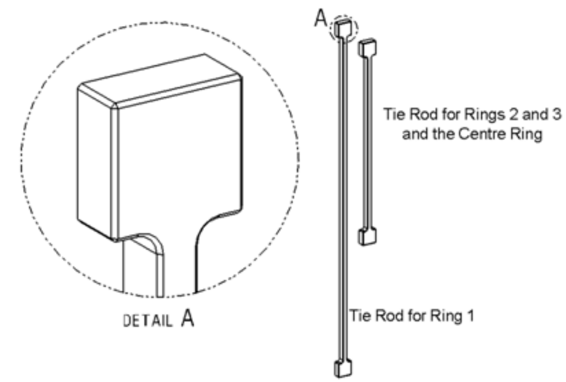
- Top reflector supports (pebble bed)
- Upper plenum insulation supports (prismatic)
- Upper core restraint devices (prismatic)
- Core lateral restraints
- Core outlet connection nozzle (between core outlet plenum and internal hot gas duct)
- Reactor control components (advanced application-significant fluence)
- Note:
  - The primary material candidate for all of the above components is Carbon Fiber Reinforced Carbon (CFRC)
  - For high-fluence applications, advanced materials may ultimately be required (e.g., SiC-based composites)

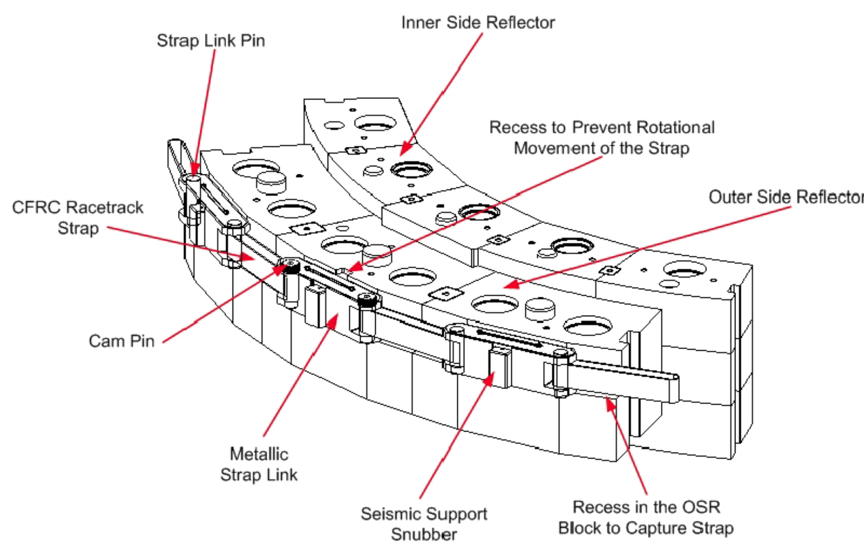
Important Considerations

- Heat resistance in an inert atmosphere to temperatures in excess of 2000°C
- High specific tensile strength and rigidity
- Low density and low thermal expansion
- Extremely high resistance to thermal shock
- Good to excellent thermal conductivity
- Anisotropy: in materials with aligned carbon fibers, the flexural and tensile strength and thermal conductivity have different values for orientations parallel and perpendicular to the fiber orientation
- Excellent fatigue resistance, even at high temperatures
- Excellent resistance to thermal creep at temperatures up to 1600ºC
- Relative chemical inertness
- Moderate resistance to fast neutron irradiation damage
- Production of high purity grades is possible

Related Experience

- Widely used in consumer and industrial products
- Limited development work to date for HTGR applications:
  - Previous USA reactor concepts
  - Manufacture and preliminary testing of components for pebble bed application
    - Top reflector supports (Tie Rods)
    - Core lateral restraint components (Racetrack Strap)





Composite Materials - Representative Properties

Property	1-D CFRC (parallel to fibers)	3-D CFRC	Fine Grained Isotropic Graphite
Density [g/cm^3]	1.7–1.8	1.7–1.8	1.75–1.85
Thermal Conductivity [$W \cdot m^{-1} \cdot °K^{-1}$]	400–600	100–200	90–200
Coefficient of Thermal Expansion [$10^{-6}/°K$]	0.1–2.0	0.1–0.2	2–5
Young's Modulus [GPa]	150–250	75–125	10–15
Bending Strength [MPa]	50–150	Values not available	40-70
Tensile Strength [MPa]	300–900	150–400	40–60
Compressive Strength [MPa]	200–500	100–200	100–200
Fracture Toughness [$MPa \cdot m^{1/2}$]	2–3	4–6	<1

Next Generation Nuclear Plant

Design and Analysis Approach

- At present there is no applicable ASME Nuclear Code for composites
  - Development of Code rules is being undertaken by the ASME Section III Subgroup on Graphite Core Components
- Proposed approach similar to Option 3 for graphite components:
  - Weibull distribution functions for strength would be experimentally determined for actual components or representative portions thereof
  - The reliability requirements associated with the appropriate Structural Reliability Class (SRC-1, -2 or -3) would be applied
- There are no significant irradiation-induced properties changes at fluence levels seen in service for components being considered
- For advanced applications (e.g., control rod components), irradiation effects would have to be taken into account

NGNP Next Generation Nuclear Plant

Qualification

- At the present state of development, Materials Qualification Plans would need to be developed for specific materials and for specific components or at least classes of components fabricated from similar materials (e.g., components cut from CFRC plate)

Outcome Objectives

Outcome Objectives – Metallic Materials

Does application of the ASME Code (including code cases) for metallic materials provide a reasonable basis for the design and qualification of components, such as for:

- ASME Code Case N-201-5
  - Core support structure for temperatures above 371 C
  - Materials: Type 316H SS, 2.25Cr-1Mo and Alloy 800H
- ASME Code Section III, Subsection NH
  - Extended temperature limits
  - Materials: Alloy 800H, Type 316H SS, and Mod 9Cr-1Mo, and 2.25Cr-1Mo
- Acceptability of the extended role of materials in HTGR safety case
  - Role of material properties in HTGR safety case
  - HTGR reliance on material properties for passive safety features
- Acceptability of alternate material qualification path
  - Qualification by review of materials analysis packages
  - Qualification of materials for specific HTGR components use

Next Generation Nuclear Plant

Outcome Objectives – Nonmetallic Materials

- Does emerging ASME Code for graphite provide a reasonable basis for design and qualification of components
- Does experimental program characterizing the fluence/temperature response of graphite provide reasonable basis for licensing the initial operation of a HTGR lead plant
- Does Reliability and Integrity Management (RIM) Program provide reasonable basis for assessing the condition of graphite components in service
- Does application of the requirements for design and manufacture of the ceramic internals provide a reasonable basis for the design and qualification of the core structure ceramic (CSC)
  - That is, other ceramic components (insulation and carbon fiber reinforced carbon (CFRC))
- Is selection of CFRC reasonable for certain structural components

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Q&A

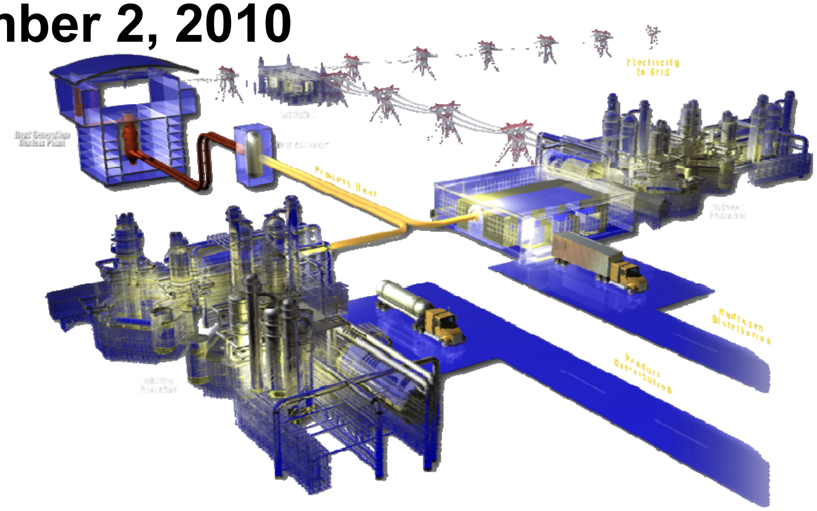
Next Generation Nuclear Plant

NGNP Fuel Qualification

Presentation to Nuclear Regulatory Commission Staff by

Next Generation Nuclear Plant Project

September 2, 2010



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White Paper

Next Generation Nuclear Plant Fuel Qualification White Paper INL/EXT-10-18610, Revision 0

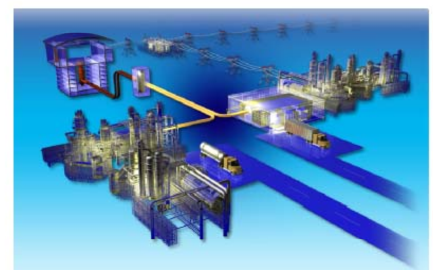
July 2010

ML 102040261

INL/EXT-10-18610
Revision 0

NGNP Fuel Qualification White Paper

July 2010



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Overview

- Introduction
- Outcome Objectives
- Regulatory Basis
- Coated Particle Fuel Experience Base
  - Common Considerations
  - German High Quality LEU-UO_2 Pebble Bed Fuel Experience
  - Prismatic Fuel Experience
- Design and Performance Requirements
  - Common Considerations
  - Pebble Fuel
  - Prismatic Fuel
- Fuel Qualification Program
- Questions and Answers
- Public Comment

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Introduction

- High temperature gas reactor (HTGR) concepts currently under consideration for the NGNP include both pebble bed and prismatic-block reactors
- Both concepts employ tristructural-isotropic (TRISO) fuel particles
  - Prismatic-block concept uses uranium oxycarbide (UCO) fuel particles made into cylindrical compacts
  - Pebble bed concept uses UO_2 fuel made into spheres
- TRISO fuel particles consist of a microsphere (kernel) of nuclear material encapsulated by multiple layers of pyrocarbon and silicon carbide layer
  - Engineered to retain fission products in the particle during normal operation and licensing basis events over the design lifetime of the fuel
- Fuel's ability to retain fission products is extremely important to the safety case and licensing approach for HTGRs

Purpose of the White Paper

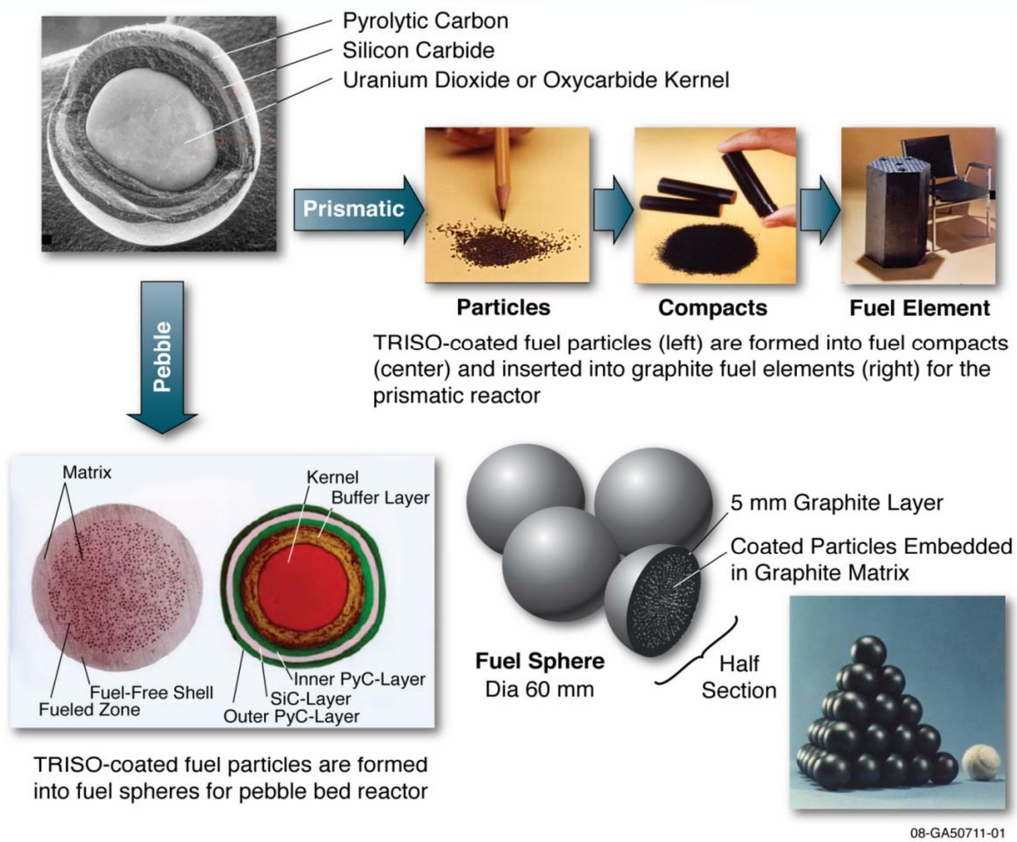
- Identify existing regulations, regulatory guidance, and licensing precedents relevant to the qualification of fuel for the NGNP project
- Summarize existing understanding, data, and analysis methods regarding coated-particle fuel performance
- Review reactor and fuel designs and resulting fuel service conditions and performance requirements
- Describe planned fuel fabrication, irradiation, testing activities, and planned approach to fuel qualification
- Obtain feedback from the Nuclear Regulatory Commission (NRC) on the planned approach to fuel qualification and information required for the combined license (COL) application

Outcome Objectives

- The primary issues for which feedback is requested include
  - Confirmation that plans established for qualification of the UO_2 pebble fuel type are generally acceptable
    - Utilization of German data for normal operation irradiation and transient/accident heat-up conditions
    - Performance of additional confirmatory irradiation and safety tests on fuel manufactured at a qualified facility to
      - (1) statistically strengthen the performance database and
      - (2) demonstrate that the fuel performs equivalent or better than the German fuel upon which the UO_2 pebble fuel design is based
  - Confirmation that plans established for qualification of the UCO prismatic fuel type are generally acceptable based on the NGNP/AGR Fuel Development and Qualification Program
  - Identification of any additional information or testing needed to meet NGNP fuel performance requirements

TRISO Fuel

- Fuel elements contain coated particles embedded in graphitic matrix
  - Cylindrical compacts inserted into hexagonal graphite blocks for prismatic reactor
  - Spheres for pebble bed reactor
- Coated particles
  - UO_2 kernel for pebble bed reactor
  - UCO kernel for prismatic reactor
  - Pyrolytic carbon and SiC layers surrounding the kernel are similar for both



- The multilayer TRISO coating system has been engineered to retain fission products during normal operation and all licensing basis events

Overview

- Present planned program for fuel qualification for NGNP
  - Both pebble and prismatic fuel options
- Great similarity between pebble and prismatic fuel forms because both are built on the worldwide historical database
  - Similar fabrication processes
  - Similar specifications
  - Similar quality control measures and statistical acceptance criteria
- Three important tenets for this fuel
  - High quality TRISO fuel can be fabricated in a repeatable consistent manner
  - Fuel performance with very low in-service failures is achievable under anticipated modular HTGR conditions
  - The broad historical international database is relevant to both fuel forms

Overview - cont

- Common objective to establish a design envelope within the expected capability of the fuel based on current understanding of TRISO fuel
  - Envelope is expressed as a set of normal operation and accident conditions that bound a broad set of historic modular pebble bed and prismatic designs
  - Both plan irradiation and accident testing
  - Both test statistically significant amount of fuel to confirm assumed failure rates under normal and accident conditions for pebble bed and prismatic design
  - Differences exist in part due to the degree of reliance on the historic database
- Exact reactor design and corresponding service conditions and performance requirements may change as the designs evolve
  - Corresponding fuel irradiation and accident test programs may need to be adjusted to address those changes

Approach to qualification of gas-cooled reactor fuel

– Advanced Gas Reactor (AGR) Fuel Development and Qualification Program deployed by Department of Energy in 2002. Program has since become part of NGNP Project and is now called NGNP/AGR Fuel Development and Qualification Program
– For the prismatic NGNP, AGR Fuel Program has focused on early testing of the UCO fuel type to demonstrate UCO TRISO fuel performance capability. Follow-on activities are concentrated on testing to qualify the UCO TRISO fuel form and its associated specification
– For the pebble bed NGNP, qualification of UO_2 TRISO fuel was based on a combination of existing German test data, early demonstration testing of UO_2 TRISO fuel in the AGR program and subsequent proof testing of fuel overseas to qualify UO_2 TRISO.
– Since international pebble bed UO_2 TRISO fuel programs have been curtailed, more work may need to be done under the AGR program to augment the existing German fuel performance database. White paper revisions will address changes in UO_2 fuel qualification.

Regulatory Basis

- 10 CFR 52.79, “Contents of application; technical information in final safety analysis report”
  - Provides content guide for COL application
- 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors”
  - Acceptance criteria for emergency core cooling systems for light water reactor (LWRs) are not directly applicable for NGNP fuel design due to design differences between LWRs and HTGR reactors
- 10 CFR Part 50, Appendix A, “General Design Criteria” provide some guidance
  - GDC 10, Reactor Design
  - GDC 35, Emergency Core Cooling

Regulatory Basis - cont

- 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants”
  - Applies to production of fuel performance data
- In summary, 10 CFR 50.43(e) requires a combination of analysis and test programs to demonstrate the performance of safety features and assure that sufficient data exist to assess the analytical tools used for safety analyses
  - This applies to NGNP design
  - This white paper describes how these requirements are met for HTGR fuel

Policy Statements/Regulatory Guidance

- SECY-93-092, “Issues Pertaining to the Advanced Reactor (PRISM, MHTGR and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements”
- NUREG-1338, “Pre-application Safety Evaluation Report for the MHTGR”
- Standard Review Plan (NUREG-0800), Section 4.2, “Fuel System Design”

Various NRC policy statements and regulatory guidance have been identified and will continue to be evaluated for relevance

Particle Fuel Phenomenon Identification and Ranking Tables (PIRTs) in NUREG/CR-6844

- NRC commissioned a panel to identify and rank the phenomena associated with TRISO coated particle fuel
- Results of PIRTs will
  - Identify key attributes of HTGR fuel manufacture that may require regulatory oversight
  - Provide reference for review of vendor HTGR fuel qualification plans
  - Provide insights for developing plans for fuel safety margin testing
  - Assist in defining test data needs for development of fuel performance and fission-product transport models
  - Develop of NRC's independent HTGR fuel performance code and fission product transport models
  - Develop of NRC's independent models for source term calculations
  - Provide insight for review of vendor HTGR fuel safety analyses

U.S. HTGR Precedents

- Peach Bottom 1, Fort St. Vrain (FSV), large HTGRs
  - Similarities
    - Coated particle fuel
    - Helium coolant
    - He-Water SG in primary system
  - Lessons learned that will improve next generation of HTGR design
- Others
  - Modular HTGR that began in 1985 (NUREG-1338)
  - In 2001, Exelon initiated pre-application interaction on the Pebble Bed Modular Reactor design (project closed in late 2002)

Coated Particle Fuel Experience Base

Coated Particle Fuel Experience Base

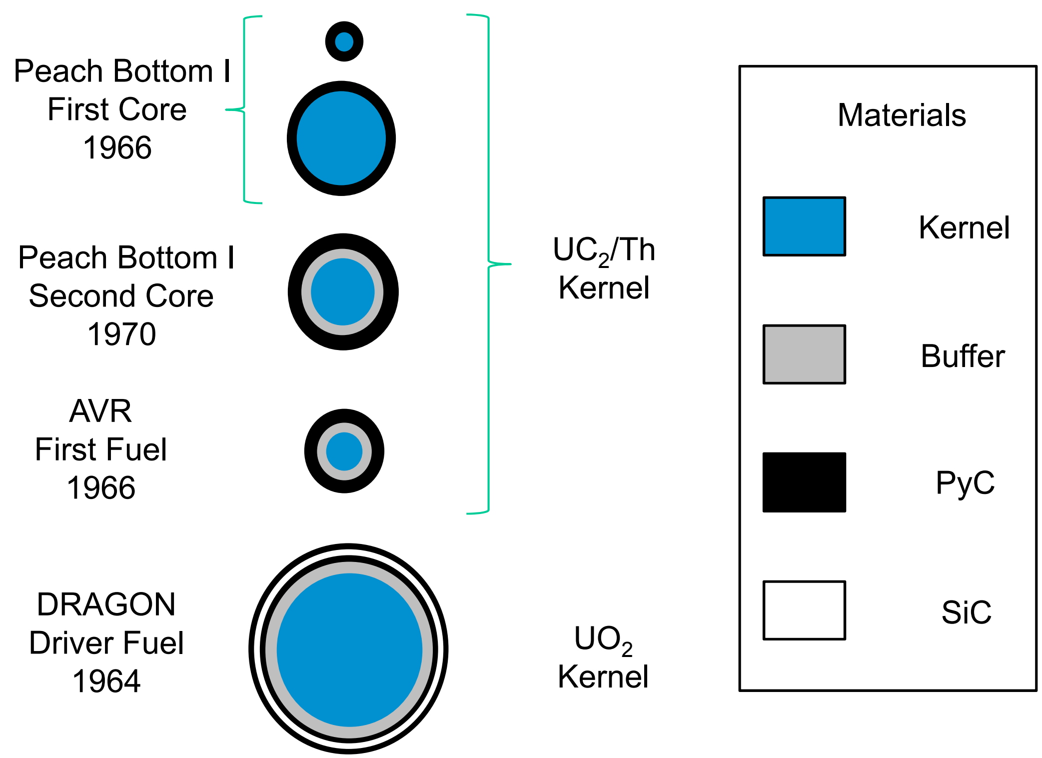
- Common Considerations
  - Experience (evolution of coated particle fuel)
  - Failure mechanisms
  - Coated particle design

- German LEU-UO_2 Pebble Fuel Experience
  - Fabrication
  - Irradiation
  - Safety testing
  - Analysis methods

- Prismatic Fuel Experience
  - Fabrication
  - Irradiation
  - Safety testing
  - Analysis methods

Evolution of Coated Particle Fuel

Early Fuel Particles



Property Variations

- Kernel
  - Diameter: 100-800 μm
  - Material: U, Th, Pu
  - Carbide, Oxide, Oxycarbide
  - Enrichment – Natural to highly enriched uranium (HEU)
- Coating configurations
  - PyC
  - Bistructural isotropic (BISO)
  - TRISO
- Fuel Forms
  - Assemblies, compacts
  - Prismatic blocks, compacts
  - Spheres

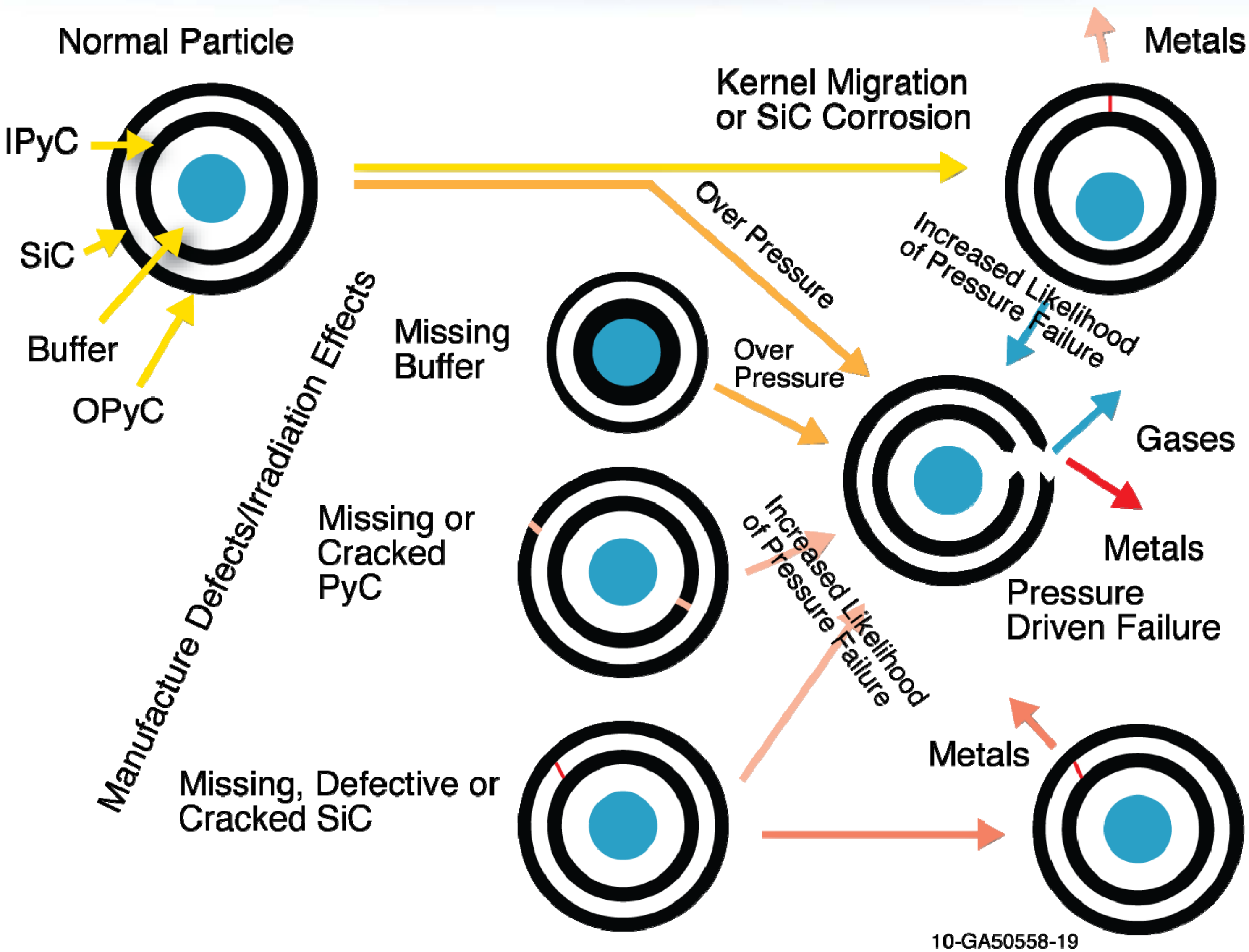
International LEU-UO_2 TRISO Experience

- Russia (spheres)
  - Irradiation Temperature: 400 to 1950°C
  - Burnup: 1 to 41% FIMA
  - Fast Fluence: 0.1 to 2.7×10^{25} n/m^2
- China (spheres)
  - Acquired and utilized German equipment and IP
  - Fabricated HTR-10 core
  - Irradiation testing in IVV-2M and HFR
  - Irradiation in HTR-10
- Japan (prismatic annular compacts)
  - Fabricated HTTR core
  - Irradiation testing in JMTR, HFIR
  - Irradiation in HTTR

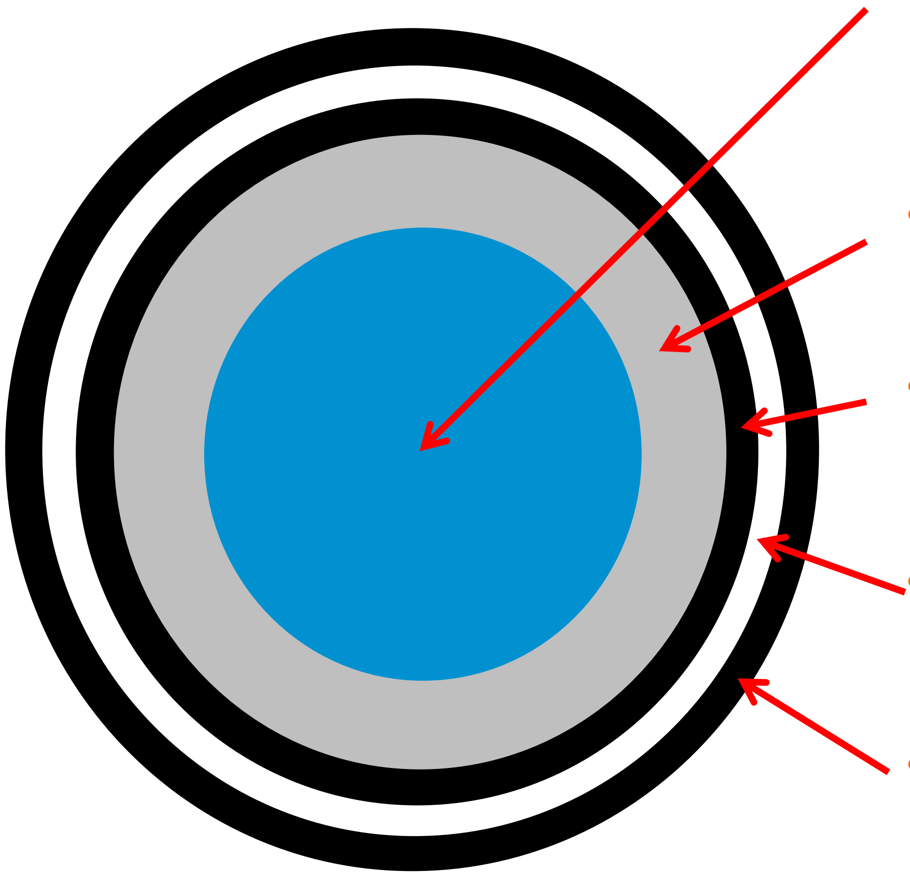
Coated Particle Failure Mechanism Control

- Structural/mechanical mechanisms
  - Excessive PyC irradiation induced shrinkage leading to SiC cracking
  - Pressure vessel failure
- Thermochemical mechanisms
  - Kernel migration
  - Corrosion of SiC
  - Thermal decomposition of SiC
- Control of failure mechanism
  - Coated particle design
  - Fuel specifications
  - Product upgrading (sieving, tabling)
  - Qualified characterization and acceptance procedures
  - Limits on service conditions (burnup, fluence, temperature, temperature gradients)

Coated Particle Failure Mechanisms



International Consensus Particle Design



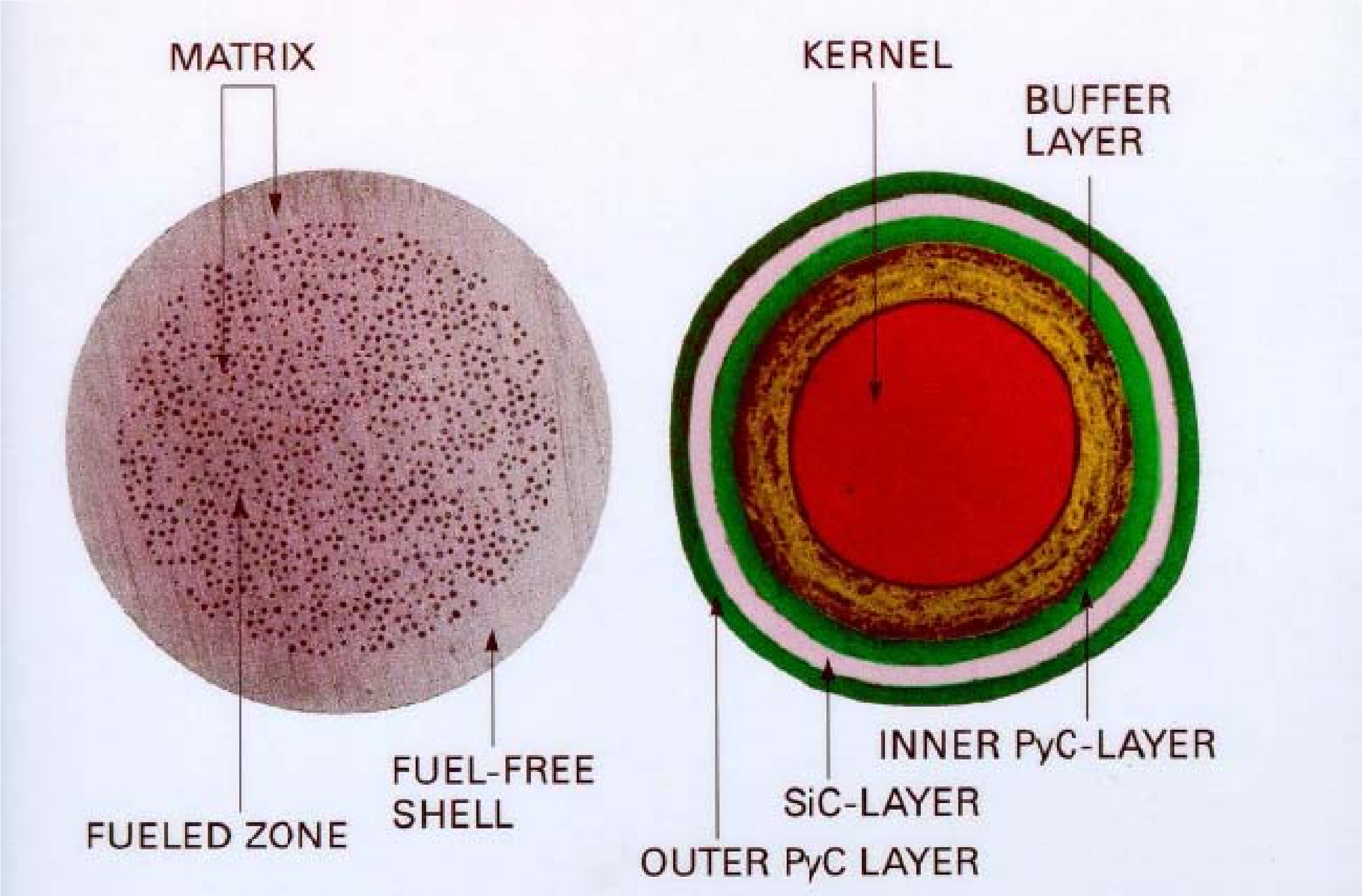
- Kernel
  - High density
  - Low enriched (8-20%)
  - UO_2 or UCO
- Buffer
  - Low density (~50% theoretical density [TD]) isotropic pyrocarbon
- Inner Pyrocarbon (IPyC)
  - High density (~85% TD) isotropic pyrocarbon
- Silicon Carbide (SiC)
  - High density (~99% TD)
  - Fine grain
- Outer Pyrocarbon (OPyC)
  - High density (~85% TD) isotropic pyrocarbon

UCO Fuel being Qualified as Improved Fuel

- UCO (UC_xO_y) is UO_2 with UC and UC_2 added
- UCO designed to provide superior fuel performance at high burnup
  - Kernel migration suppressed (most important for prismatic designs because of larger thermal gradients)
  - Eliminates CO formation; internal gas pressure reduced
  - Fission products still immobilized as oxides
  - Allows longer, more economical fuel cycle
- Reference fuel for NGNP prismatic reactor designs
- Potential higher burnup alternative for pebble bed HTGRs

Pebble Bed

German LEU TRISO Pebble Bed Fuel Experience



German LEU TRISO Fabrication Experience

Characteristic	Pre-1985 Production			Post-1985 Production	
Year of Manufacture	1981	1981	1983	1985	1988
Designation	GLE 3	LEU Phase I	GLE 4	GLE 4/2	Proof Test Phase 2
Matrix Material	A3-27	A3-27	A3-27	A3-3	A3-3
Irradiation Test Designation	AVR 19	HFR-K3 FRJ2-K13 HFR-P4 SL-P1 FRJ2-P27	AVR 21-1 FRJ2-K15	AVR 21-2 HFR-EU1 HFR-EU1bis	HFR-K5 HFR-K6
Approximate number of fuel spheres manufactured	24,600	100	20,500	14,000	200

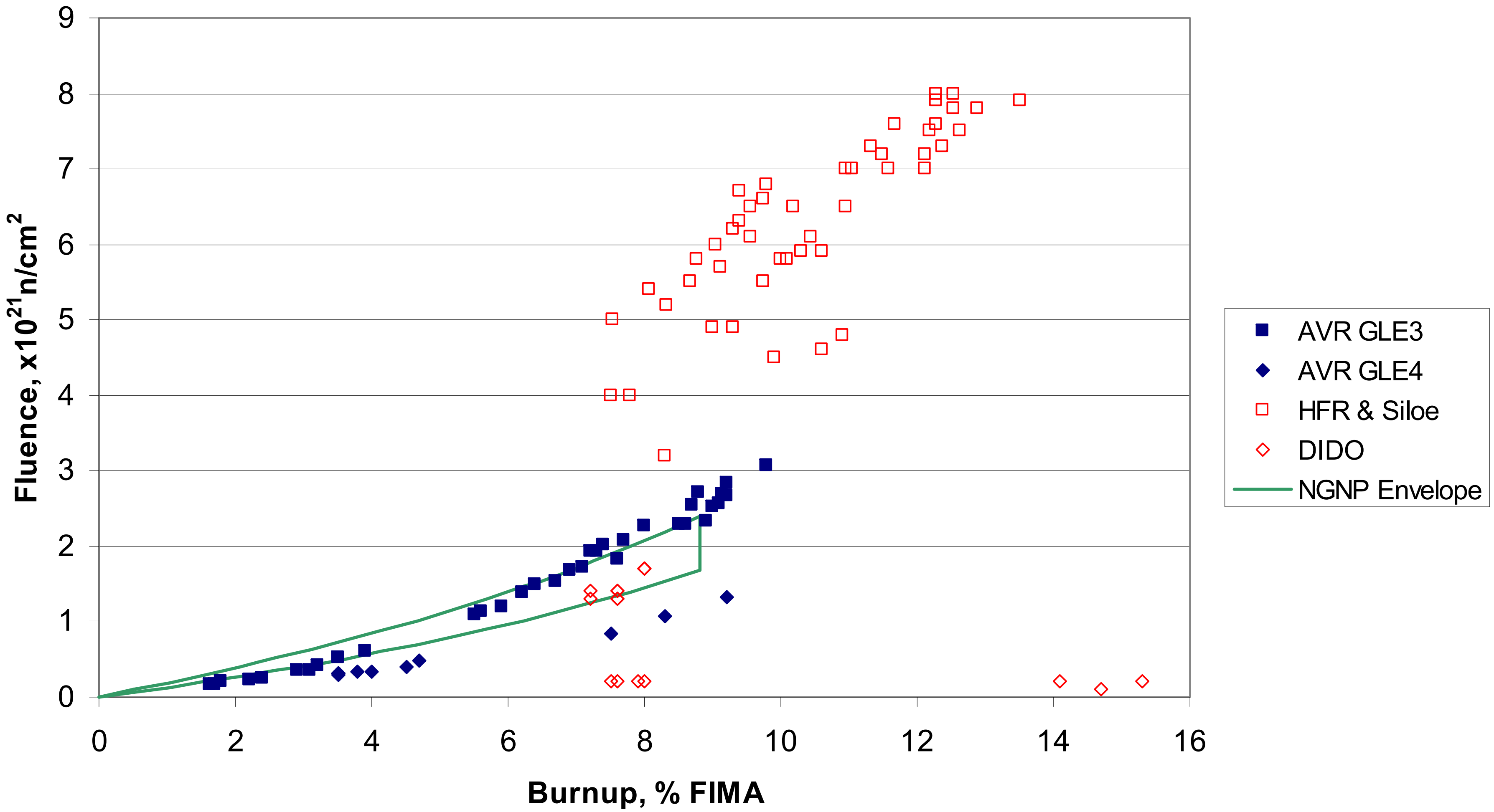
The symbols used in the 'Irradiation Test Designation' row have the following meanings:

1. The first three letters describe the reactor in which the test was done:
   AVR = *Arbeitsgemeinschaft Versuchsreaktor* in Jülich, Germany
   HFR = High Flux Reactor in Petten
   FRJ2 = DIDO reactor in Jülich
   SL = Siloe reactor in Grenoble
2. The next group of symbols describes the irradiation sample type and test number. In the case of AVR irradiations, the reload number is used (i.e., AVR 19), which means that the fuel spheres made up the 19th partial reload of the reactor. In other tests, the letter K designates a full-sized fuel sphere, the letter P designates coated particles in any other form—i.e., small spheres, compacts or coupons—and the number is the test number. Thus, FRJ2-P27 means irradiation test number 27 performed on coated particles in the DIDO reactor in Jülich.

Manufacturing Detail for German LEU TRISO Fuel

Characteristic	Pre-1985 Production			Post-1985 Production		Pebble-Bed Design Specification
Designation	GLE 3	LEU Phase I	GLE 4	GLE 4-2	Proof Test Phase 2	
Kernel Diameter (µm)	500	497	501	502	508	500
Kernel Density ($g.cm^{-3}$)	10.80	10.81	10.85	10.87	10.72	>10.4
Coating Thickness (µm)						
Buffer Layer	93	94	92	92	102	95
Inner PyC Layer	38	41	38	40	39	40
SiC Layer	35	36	33	35	36	35
Outer PyC Layer	40	40	41	40	38	40
Coating Density ($g.cm^{-3}$)						
Buffer Layer	1.01	1.00	1.01	1.1	1.02	<1.05
Inner PyC Layer	1.86	~1.9	1.9	1.9	1.92	1.9
SiC Layer	3.19	3.20	3.20	3.2	3.20	≥ 3.18
Outer PyC Layer	1.89	1.88	1.88	1.9	1.92	1.9
Fuel Sphere Loading						
Heavy Metal (g/FS)	10	10	6	6	9.4	7
Uranium 235 (g/FS)	1	1	1	1	1	0.545
Enrichment (% U-235)	9.82	9.82	16.76	16.76	10.6	7.8
Coated Particle per FS	16,400	16,400	9,560	9,560	14,580	11,200
Free-Uranium Fraction ($\times 10^{-6}$)	50.7	35	43.2	7.8	13.5	< 60

Irradiation Conditions, AVR and MTRs compared to NGNP Pebble Envelope



Sphere Irradiation Data Summary

No. of Spheres	No. of Particles	Burnup % FIMA	Fast Fluence 10^{25} n/m^2	Irradiation Temperature ˚C	Test Temperature ˚C	Exposed Kernels	SiC Defects
AVR Spheres							
13/14[1]	213,000/229,6000[1]	3.5 to 9.8	0.5 to 2.9		1600 to 1800	0	8
Materials Test Reactor Spheres							
4	65,6000	8 to 10.6	0.2 to 5.9	920 to 1200	1600 to 1800	0	1
12	182,240	7.5 to 10.3	0.2 to 5.9	903 to 1220	NA[2]	3	NA

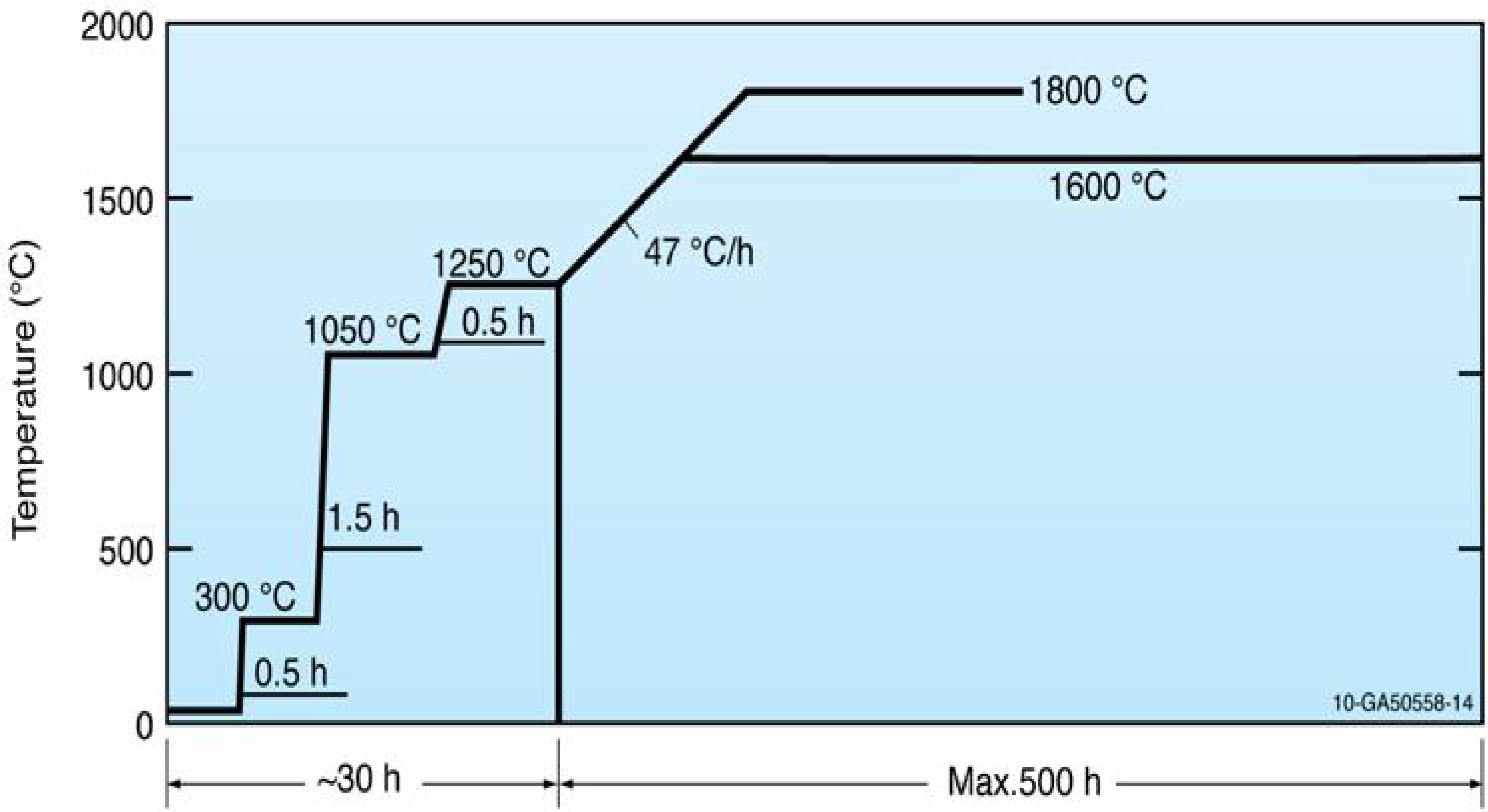
Analysis Summary of Irradiated Spheres

Parameter	Number of Particles	Total Particles	Maximum Parent Population Particle Fraction	
Confidence Level that Indicated Particle Fraction is not Exceeded in Parent Population			50%	95%
Exposed Kernels	3	477,400	**7.6×10^{-6}**	**1.6×10^{-5}**
SiC Defects	9	278,800	**3.5×10^{-5}**	**5.6×10^{-5}**

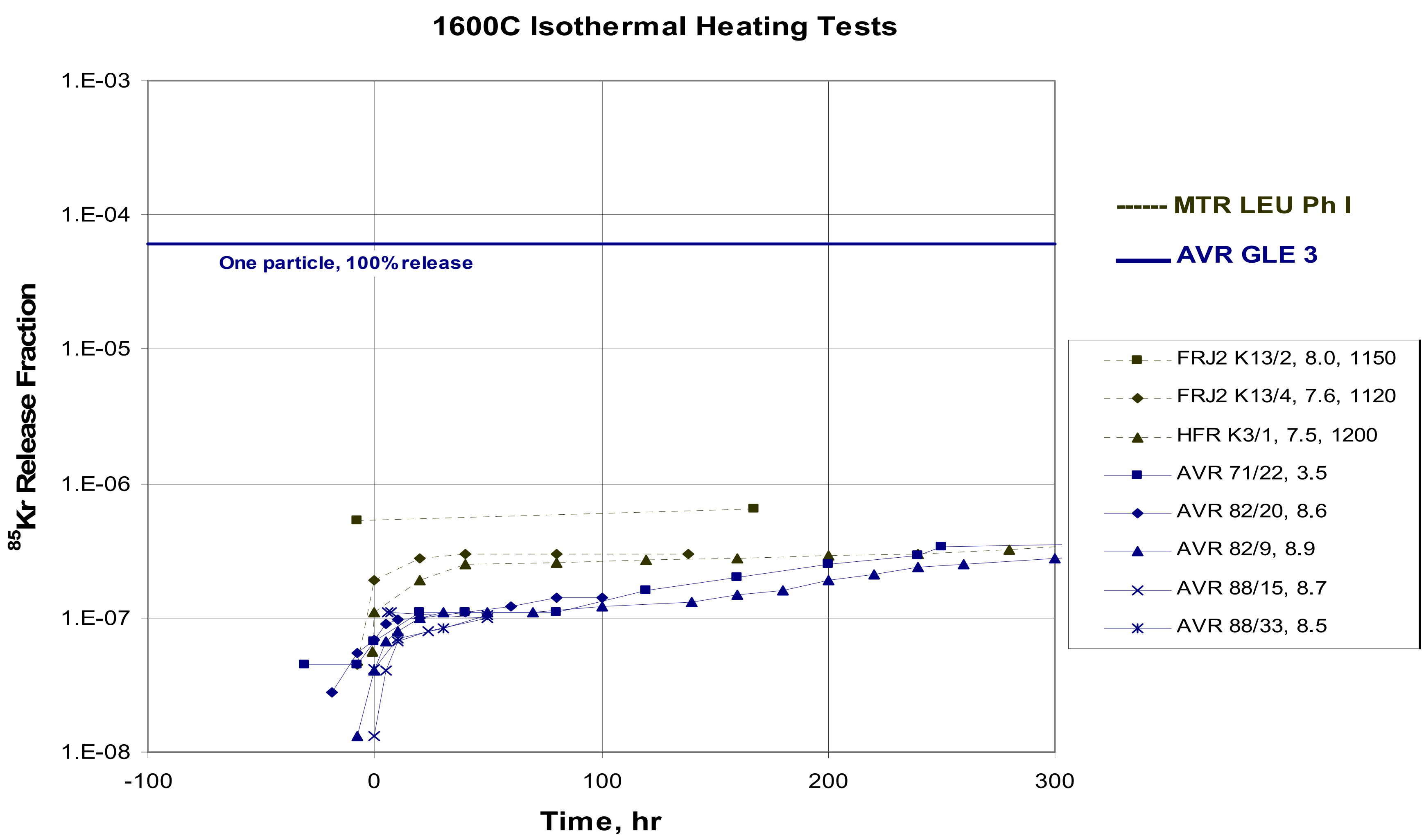
1 - The AVR 82/9 sphere data were not used for SiC defect determination because the heating test Cs release data were reported to have been distorted by contamination

2 - Not Applicable – These spheres were not subjected to heating test (exposed kernels were detectable by in-pile gas release)

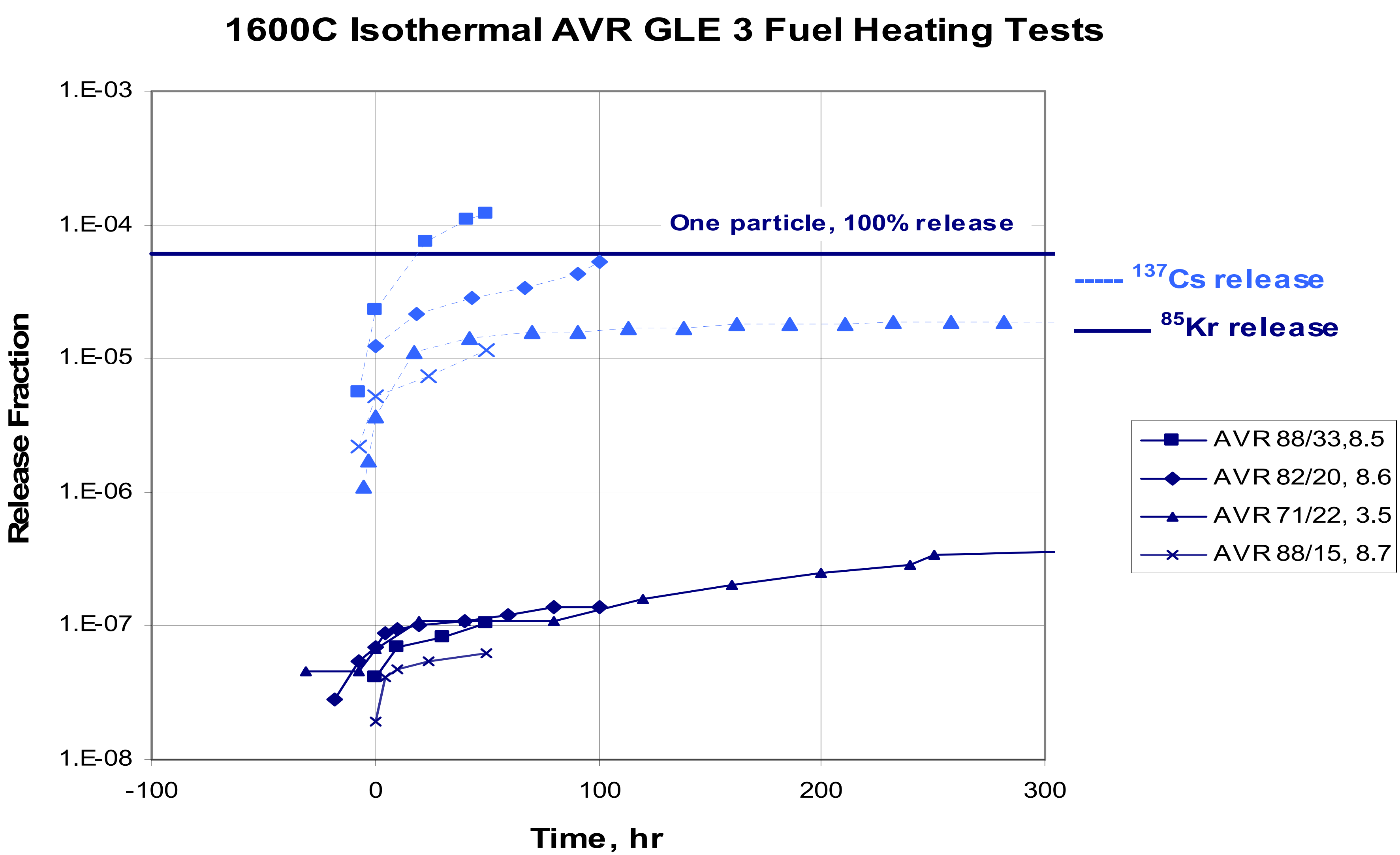
Heating Test Temperature Profile



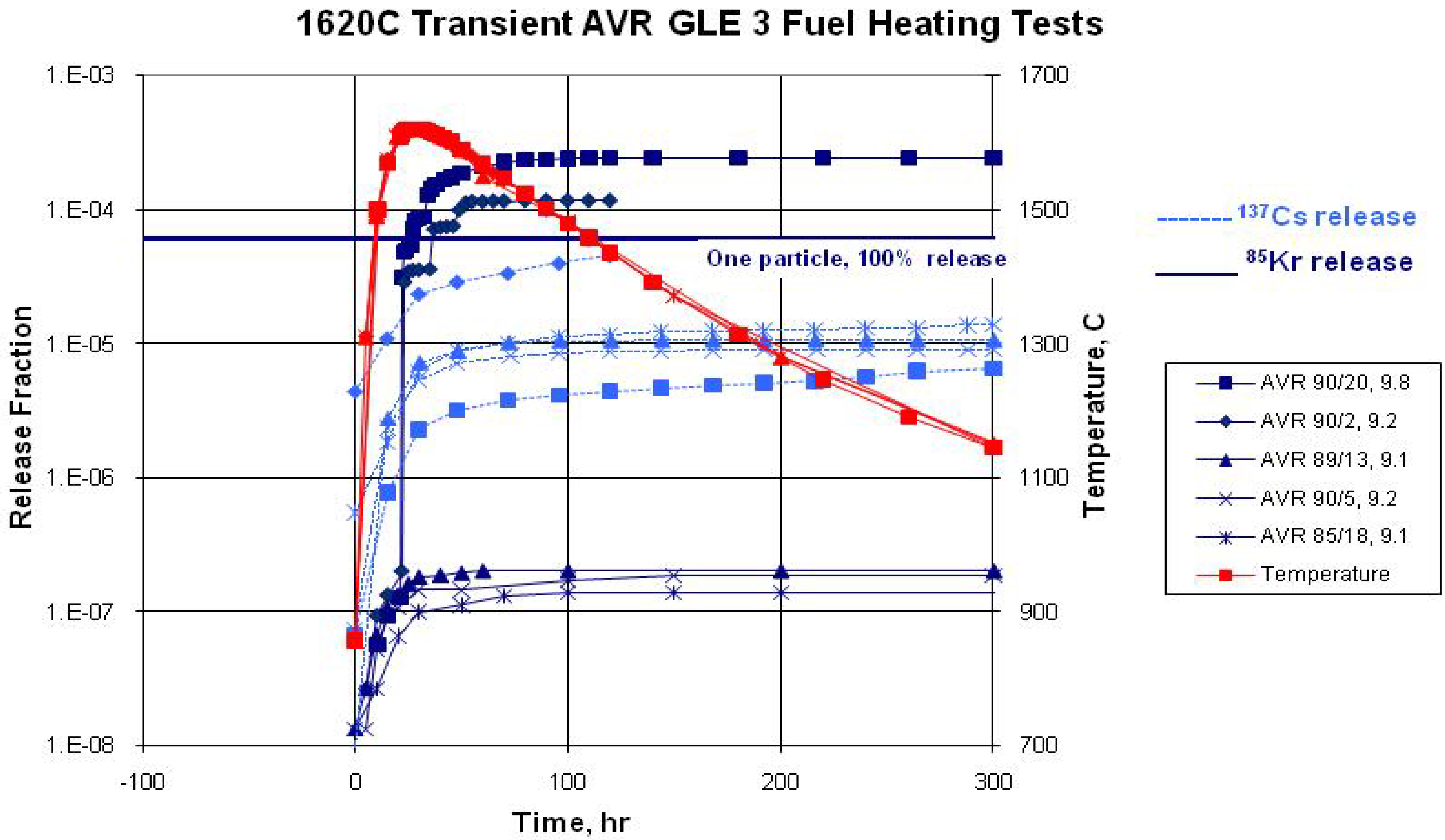
End of Irradiation Exposed Kernels



End of Irradiation SiC Defects



Maximum 1620°C Transient Heating Tests



1600-1620°C Heating Test 85Kr Release

Identifier	Burnup (% FIMA)	Fast Fluence (10^{21} n/cm^2)	Irradiation Temperature (°C)	Test Type	Number of Particles	Exposed Kernels
AVR Spheres (GLE 3)						
90/20	9.8	2.94	Not available	Transient	16,400	4
90/2	9.2	2.66	Not available	Transient	16,400	2
90/5	9.2	2.66	Not available	Transient	16,400	0
85/18	9.15	2.63	Not available	Transient	16,400	0
89/13	9.1	2.61	Not available	Transient	16,400	0
82/9	8.9	2.52	Not available	Isothermal	16,400	0
88/15	8.7	2.43	Not available	Isothermal	16,400	0
82/20	8.6	2.38	Not available	Isothermal	16,400	0
88/33	8.5	2.33	Not available	Isothermal	16,400	0
71/22	3.5	0.48	Not available	Isothermal	16,400	0
MTR Spheres (LEU Phase 1)						
HFR-K3/1	7.5	4	1,200	Isothermal	16,400	0
FRJ2-K13/2	8	0.2	1,150	Isothermal	16,400	0
FRJ2-K13/4	7.6	0.2	1,120	Isothermal	16,400	0
Average	8.3	2.2		Total	213,200	6
Maximum Parent Population Exposed Kernel Fraction, 50% confidence						3.1×10^{-5}
Maximum Parent Population Exposed Kernel Fraction, 95% confidence						5.6×10^{-5}

Heating Test Particle Failure Summary

Parameter	Test Temperature			
	1,600°C	1,700°C	1,800°C	1,800°C[a]
Average Burnup, % FIMA	8.3	7.6	6.5	8.4
Average Fast Fluence, 10^{21} n/cm^2	2.2	2.0	1.8	2.6
Number Particles	213,200	32,800	82,000	49,200
Number Exposed Kernels	6	20	69	12
Exposed Kernel Fraction[b] (50% confidence)	3.1×10^{-5}	6.3×10^{-4}	8.5×10^{-4}	2.6×10^{-4}
Exposed Kernel Fraction[b] (95% confidence)	5.6×10^{-5}	8.9×10^{-4}	1.03×10^{-3}	4.0×10^{-4}

a. Excludes AVR 74/10 and 70/33 test data.

b. Maximum parent-population exposed-kernel fraction.

Fuel Performance Analysis Methods

- Utilize manufacturing, irradiation and safety testing data
  - As-fabricated data from burn-leach tests
  - Normal operation data from in-pile gas release
  - Accident condition data from safety testing
- Analysis of in-pile data using NOBLEG code
  - Failed particle release
  - Contamination release
- Fuel performance calculation from empirical curve
  - Particle failure fraction as function of temperature
  - Derived from German irradiation and testing data
  - Encompasses normal operation and accident conditions

Prismatic

Prior Prismatic Fabrication Experience

- U.S. experience
  - Commercial fuel fabrication of prismatic TRISO fuel at General Atomics (GA) for FSV (Th, U)C_2 fissile and ThC_2 fertile particles
  - UCO TRISO fuel for 12 test capsules irradiated in U.S. and Sweden
    - 350 micron LEU kernels – HRB-14, 15A, 15B, 16, 17, 18, 21 and R2-K13
    - 200 micron HEU kernels – NPR-1, 2, and 1A
- German experience
  - AVR – 5,354 fuel spheres TRISO UCO (21% of reactor core)
  - FRJ-P24 – annular TRISO UCO compacts containing 300 micron kernels
- Japanese experience
  - Annular compacts with TRISO UO_2 fuel for HTTR

Prior Prismatic Irradiation Experience

- U.S. experience
  - UCO fuel in U.S. tests generally failed to meet HTGR prismatic performance requirements due to poor SiC coating performance unrelated to kernel composition
- TRISO-P fuel in HRB-21 and NPR irradiations suffered cracking of highly anisotropic IPyC leading to SiC failure
- German experience
  - FRJ2-P24 irradiation of TRISO UCO fuel under prismatic HTGR conditions of temperature and burnup (but very low fluence) showed excellent fission gas retention
  - Fission gas release-to-birth (R/B) remained low in Arbeitsgemeinschaft Versuchsreaktor (AVR) with 21% of core containing TRISO UCO fuel

Prior Prismatic Irradiation Experience - cont

- Japanese experience
  - TRISO UO_2 irradiated in HRB-22 test to a burnup of 4.8% FIMA with particle failure fraction 1E-4 measured by on-line radiation detectors
  - TRISO UO_2 irradiated in Oarai Gas Loop-1 to burnup of 3.7% FIMA with through-coating failures less than 3E-4
  - HTTR operating with TRISO UO_2 prismatic fuel is experiencing excellent performance R/B ~ 1E-8, including a brief period of operation at 950°C, although at very low burnup, less than 3.6% FIMA

Prior Prismatic Safety Testing Experience

- U.S. experience
  - Isothermal heating of 10-particle batches of TRISO UCO, UO_2, UO_2* and UC_2 for 10,000 hours at 1,200, 1,350, and 1,500°C
  - Eu-154 release greater from UCO particles than from UO_2 particles, however Eu retention by graphite minimizes release from core
  - Ag-110m release greater from UO_2 particles than from UCO particles, because SiC on UO_2 particles had columnar grain orientation (grain boundaries in radial direction) whereas in UCO particles SiC was laminar (circumferential direction)
  - Cs-137 release was measured only in particles with defective SiC
  - Tests indicate some effect of kernel chemistry in terms of Eu release, but this effect is likely not significant due to holdup in graphite

Prior Prismatic Safety Testing Experience

- German experience
  - Heated U.S. TRISO UCO fuel from R2-K13 irradiation to 2,500°C and total SiC thermal failure, so not informative of safety performance under modular HTGR conditions
  - Performed many safety tests of TRISO UO_2 fuel under modular HTGR conditions showing excellent performance
- Japanese experience
  - TRISO UO_2 fuel irradiated in HRB-22 tested at 1,600, 1,700 and 1,800°C
  - At 1,600°C, one particle out of 2,800 failed, a failure rate of 4E-4
  - At 1,700 and 1,800°C, metallic fission product releases varied greatly from particle to particle, but the causes of the variation were unclear
- Prior experience demonstrates accident performance of TRISO fuel depends on quality coatings not on kernel composition

Prismatic Analysis Methods Experience

- U.S. experience
  - Computer codes have been developed by General Atomics to model fuel performance
  - These codes are intimately connected to codes that track fission product release and transport within the core and primary cooling system that are key inputs to the mechanistic source term
  - A brief description of these codes is provided in the MST white paper
  - Current work on fuel performance model development, namely the particle fuel model code (PARFUME), is ongoing at Idaho National Laboratory in the NGNP/AGR Fuel Program

Design and Performance Requirements

Design and Performance Requirements

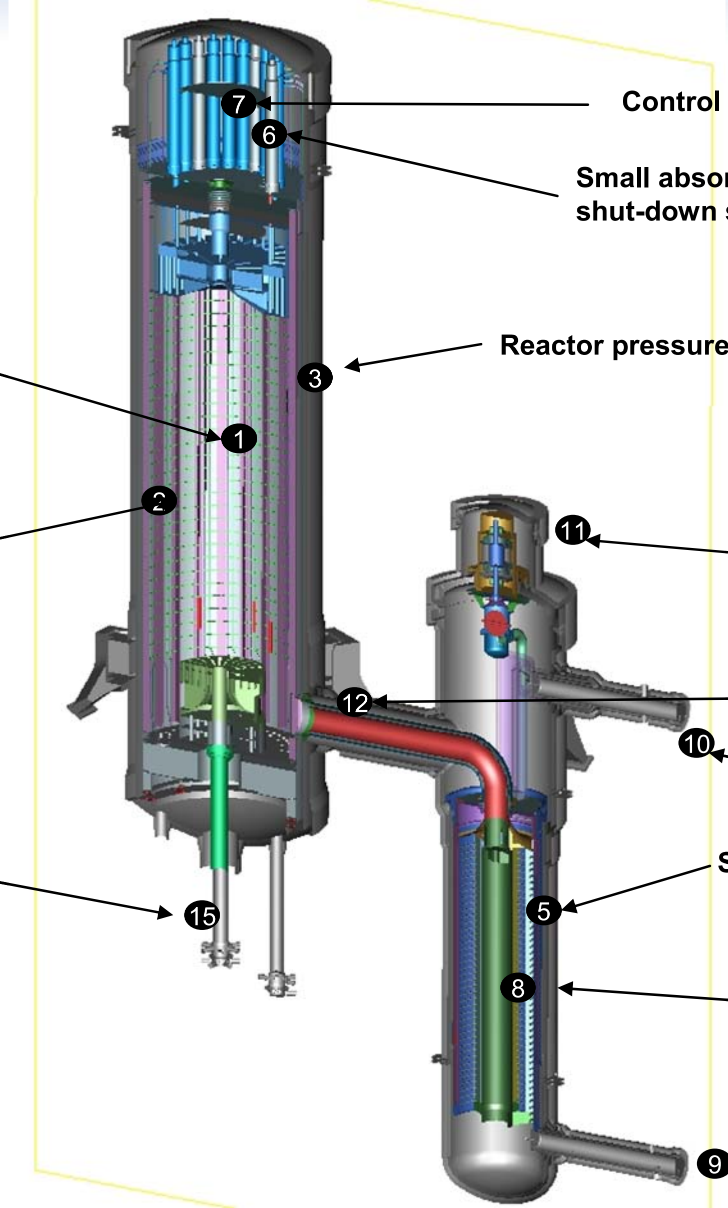
- Common considerations
  - Inherent characteristics
  - Dominant service condition parameters
- Pebble bed design
  - Reactor
  - Fuel
  - Fuel service conditions
  - Performance requirements
- Prismatic design
  - Reactor
  - Fuel
  - Fuel service conditions
  - Performance requirements

Common Considerations

- Reactor core characteristics
  - Low power density/high heat capacity with or without coolant
  - Negative temperature coefficient
  - Large neutron migration length
  - Single phase neutronically and chemically inert coolant
  - High thermal conductivity with or without coolant
- Fuel characteristics
  - High degree of fission product retention under normal operation or accident conditions
  - Large overpower and temperature margins
- Dominant service condition parameters
  - Burnup
  - Temperature (normal operation and accident conditions)
  - Temperature gradient during normal operation
  - Fast fluence

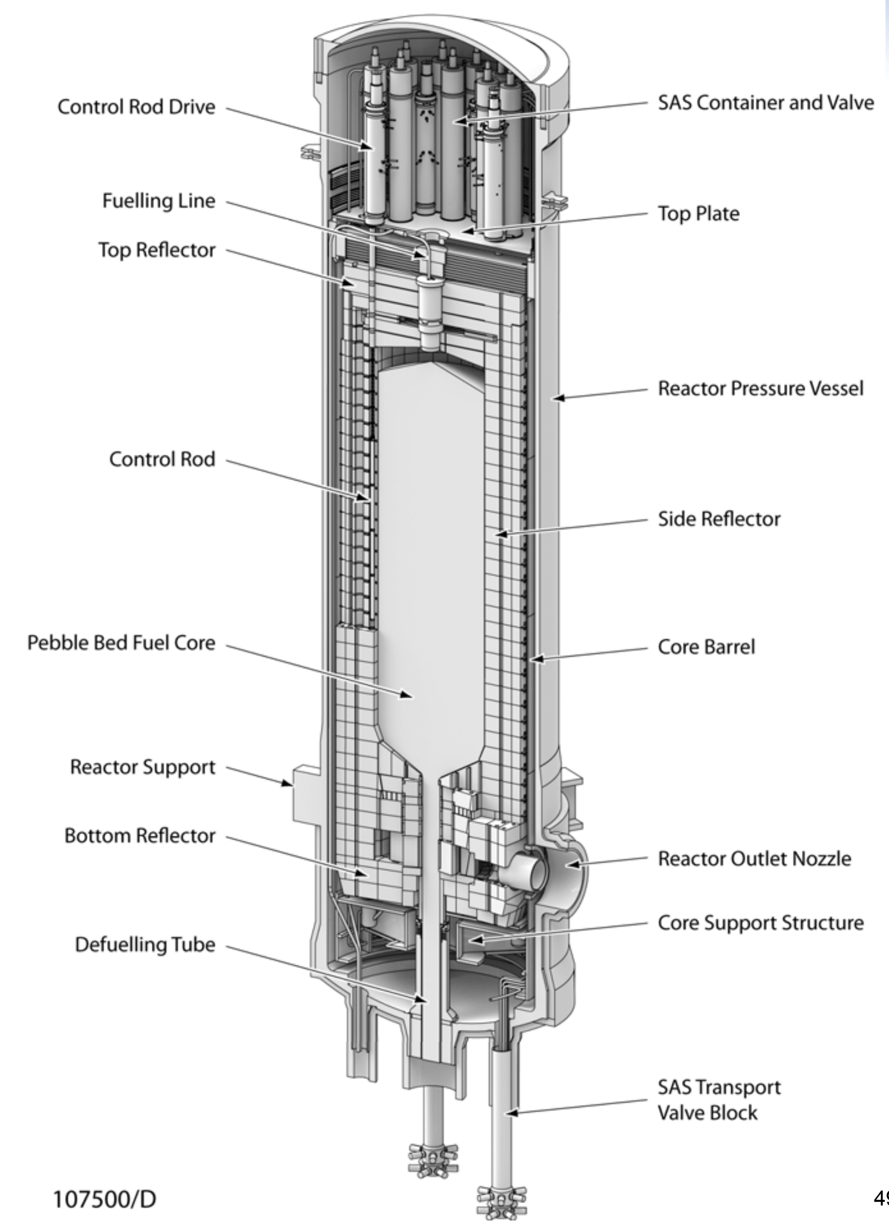
Pebble Bed Design

Pebble Bed Primary Circuit



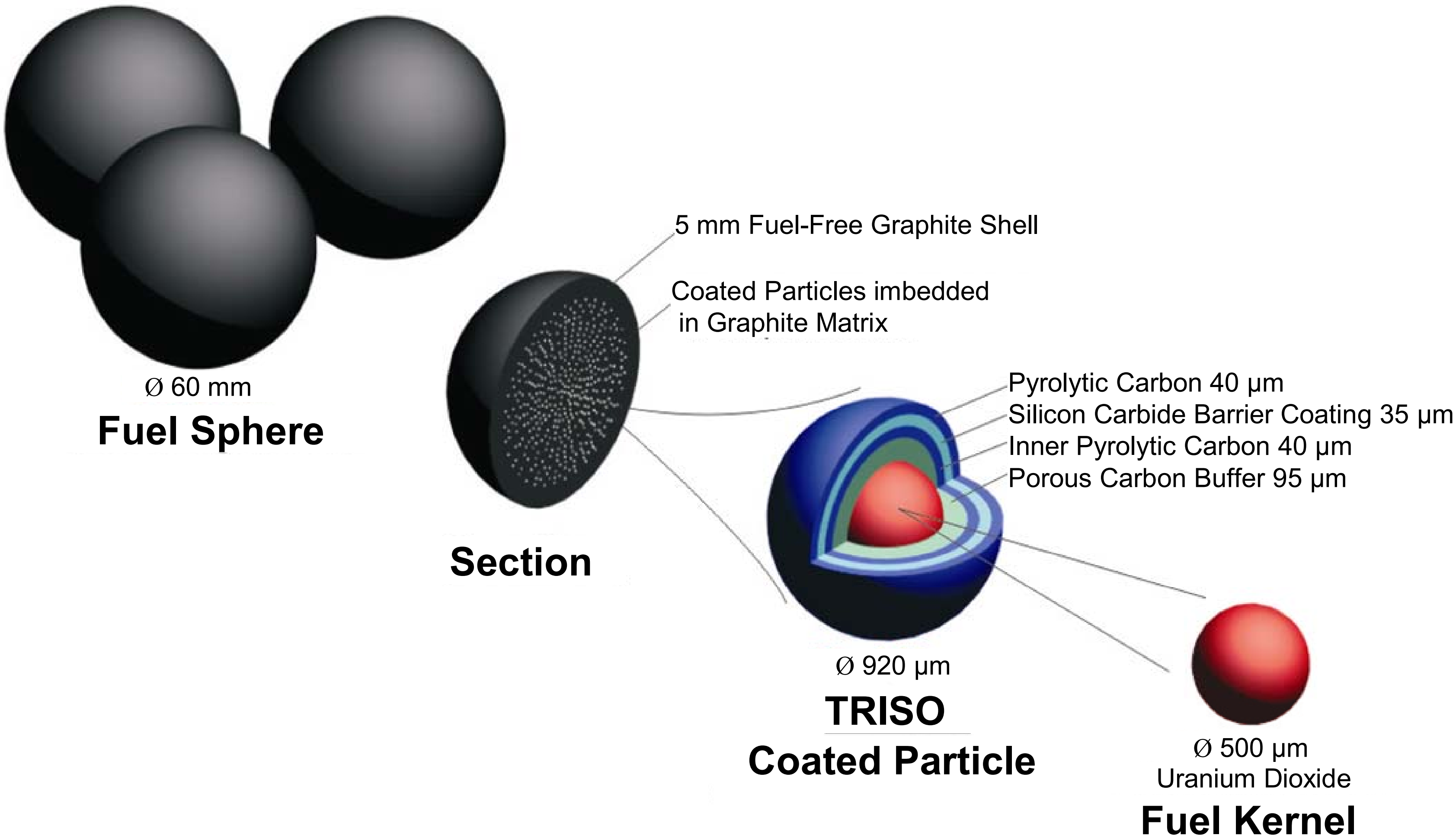
Power 250 MWt
T coolant/in = 250°C
T coolant/out = 750°C

Pebble Bed Reactor Unit



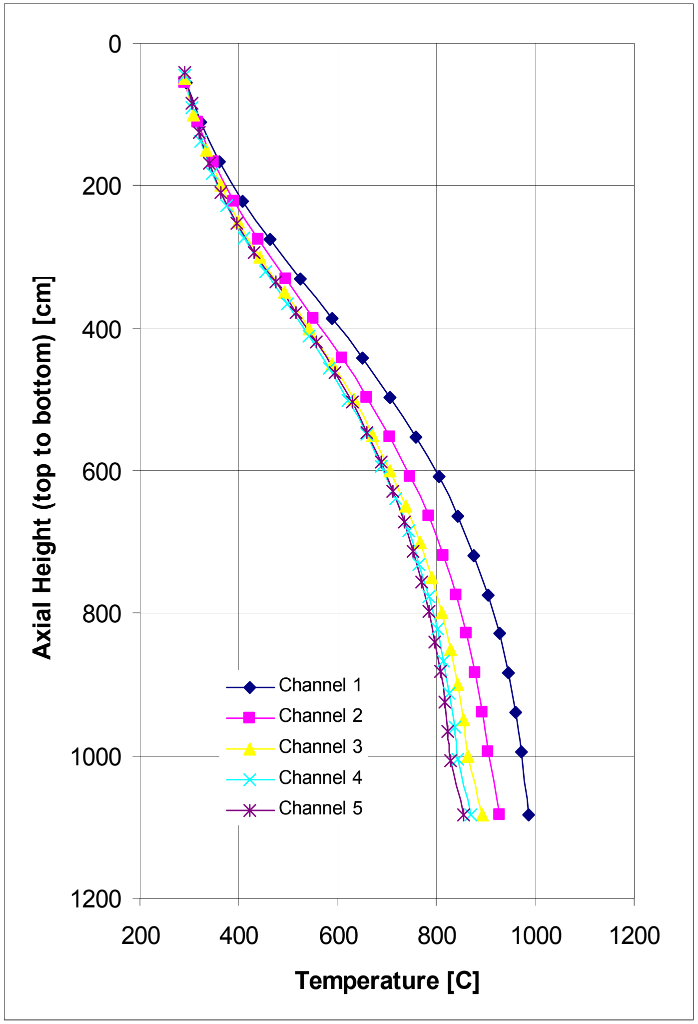
Pebble bed diameter 3.0 m
Pebble bed height 10.5 m
Fuel spheres in core ~400,000

Pebble Bed Fuel

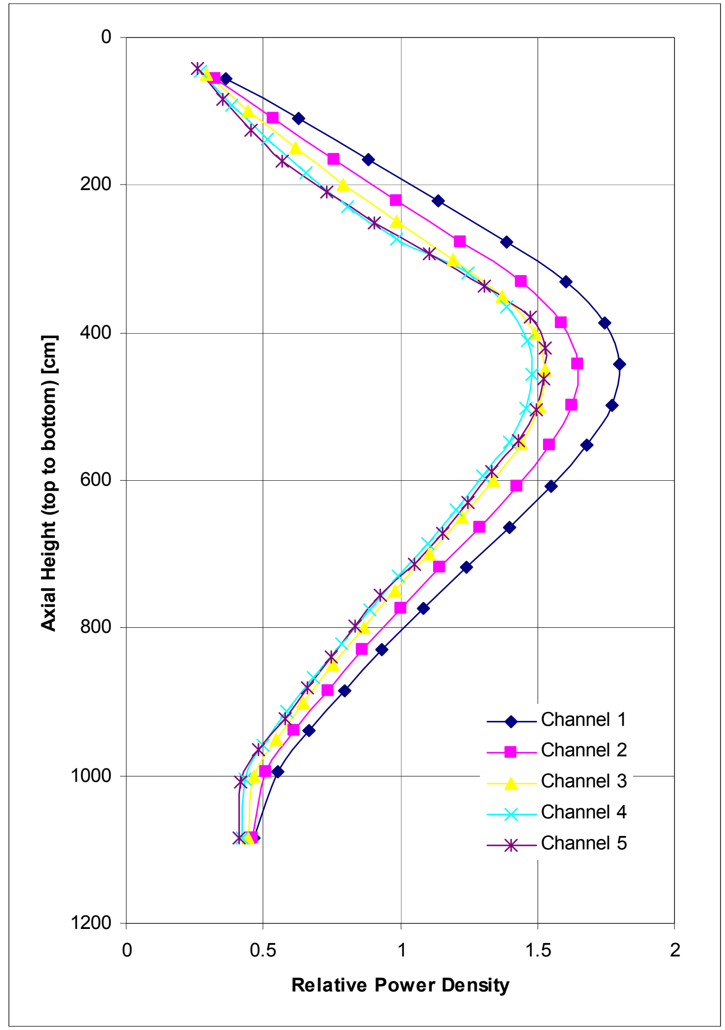


Pebble Bed Normal Operation Conditions

Temperature Distribution

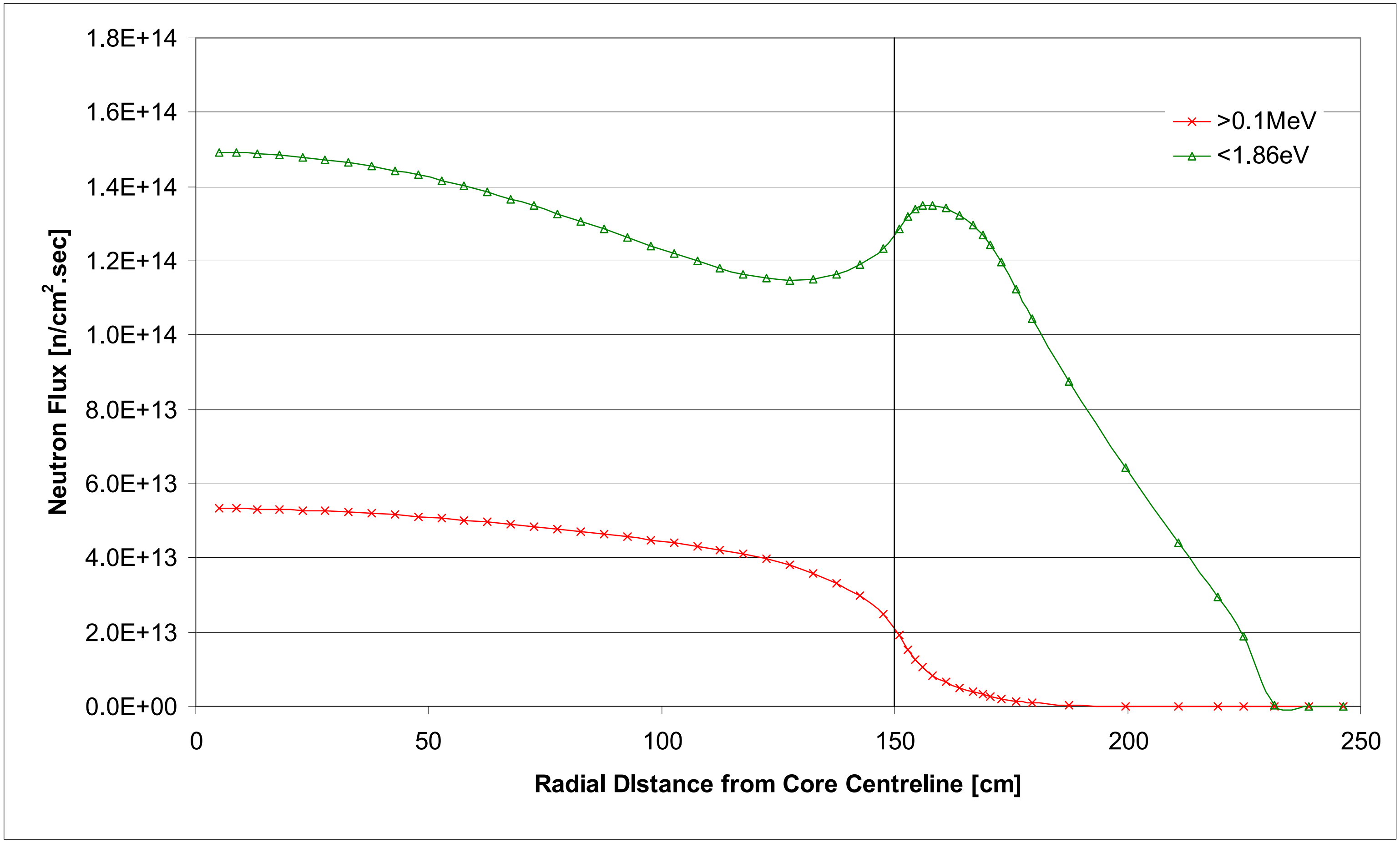


Power Distribution



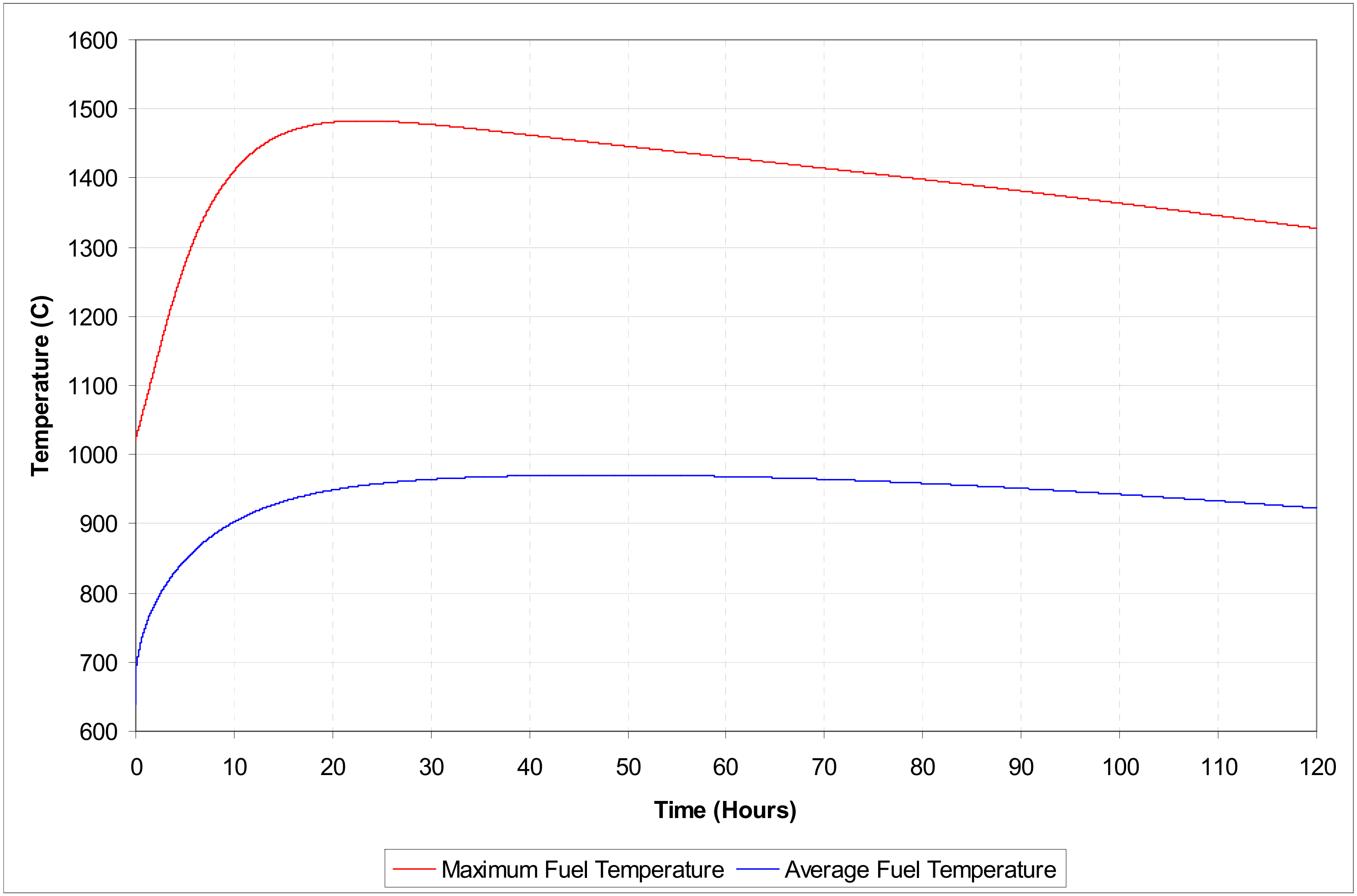
Pebble Bed Normal Operation Conditions

Fast and Thermal Flux Distributions



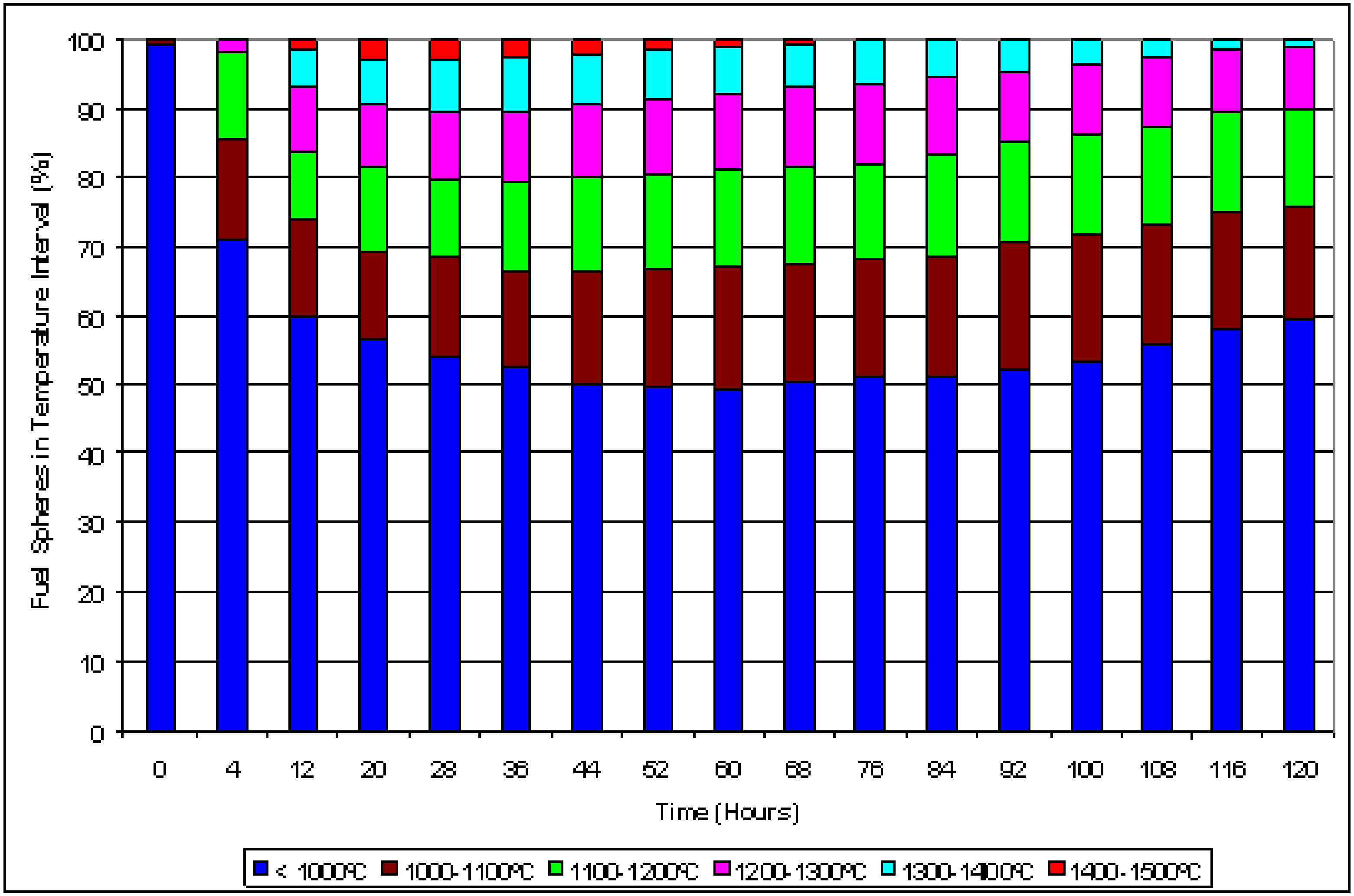
Pebble Bed Accident Conditions

Depressurized Loss of Forced Cooling



Pebble Bed Accident Conditions

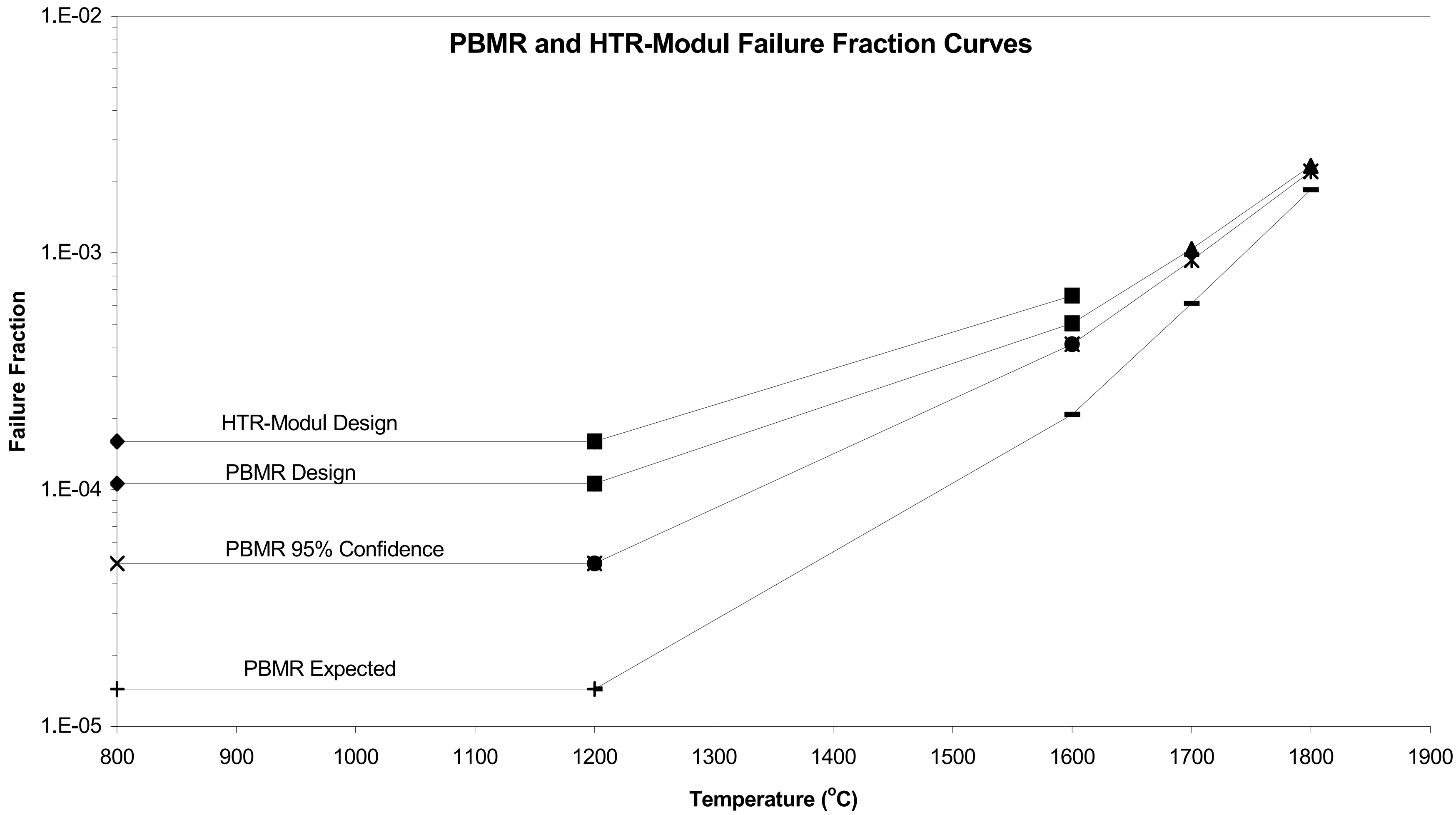
Depressurized Loss of Forced Cooling



Pebble Bed Fuel Service Conditions

Parameter	Core Average Value	Maximum Value
Normal Operation		
Discharge Burnup, % FIMA	8.31	8.75
Discharge Fast Fluence, 10^{21} n/cm^2 (E > 0.1 MeV)	2.01	2.39
Sphere Center Temperature, °C	644	1,048
Accident Conditions – Best Estimate Maximum Transient Conditions		
Sphere Temperature, °C	970	1,483

Pebble Bed Fuel Performance Requirements

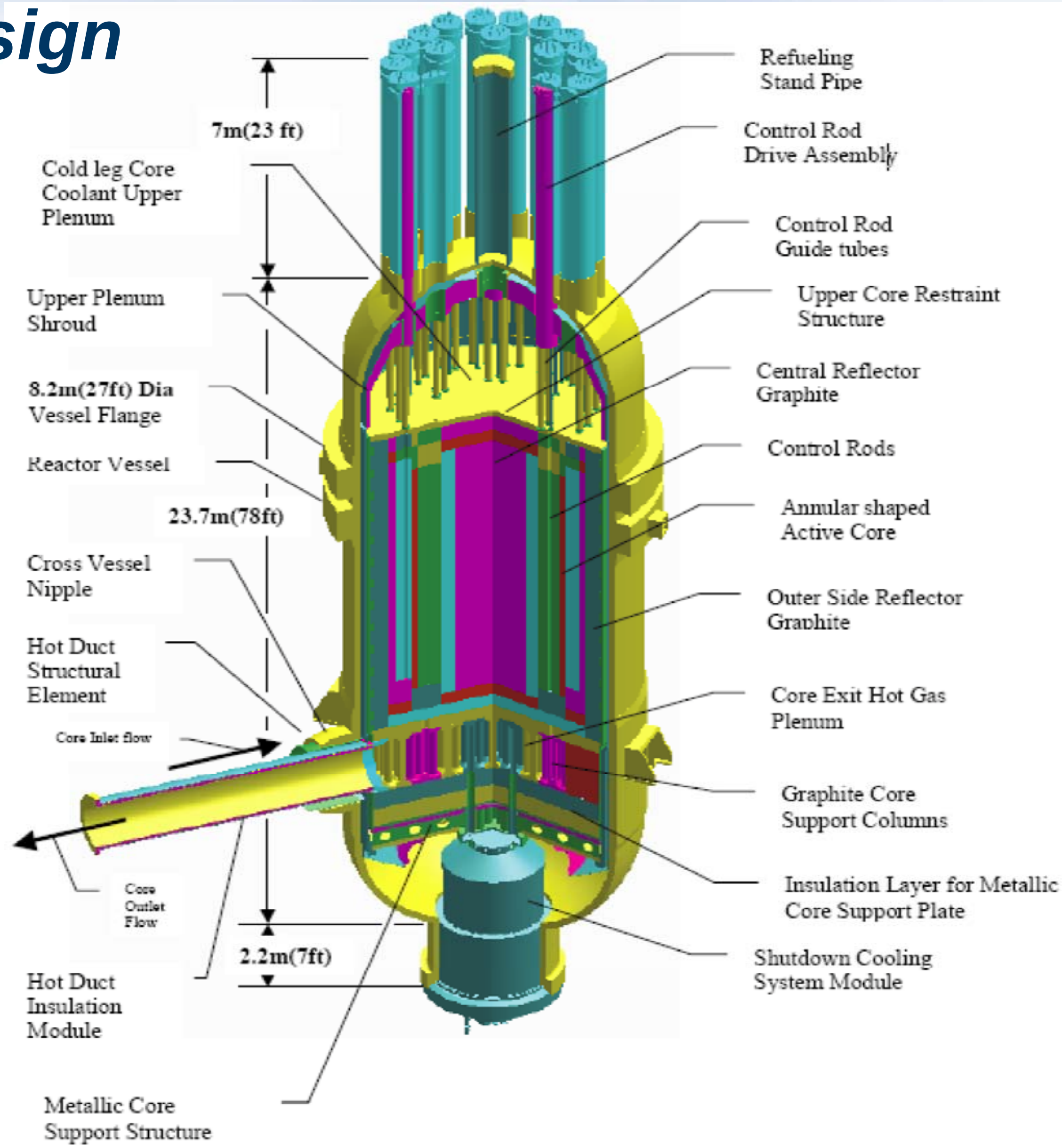


Prismatic Design

Prismatic Fuel Reactor Design

- Reactor is in beginning stages of conceptual design
- Point of departure is 350 MWth MHTGR
- Prismatic core consists of hexagonal (prismatic) graphite fuel elements (blocks) 793-mm long by 360-mm across flats
- Fuel blocks are stacked 10 high in 66 columns forming 3 rings (18, 24 and 24 columns) in an annular configuration
- Reflector graphite blocks are provided inside and outside the active core

Prismatic Fuel Reactor Design

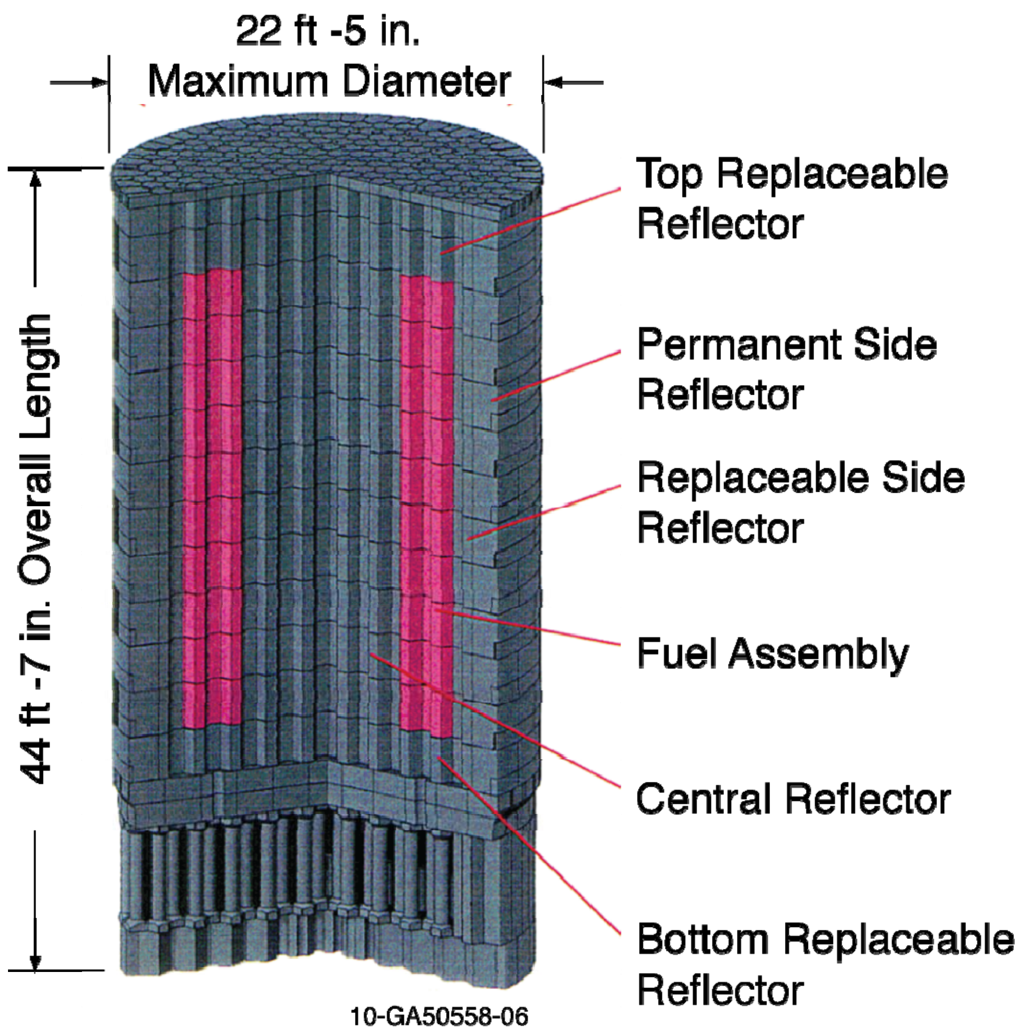


Prismatic Fuel – 600 MWt Thermal Core Design

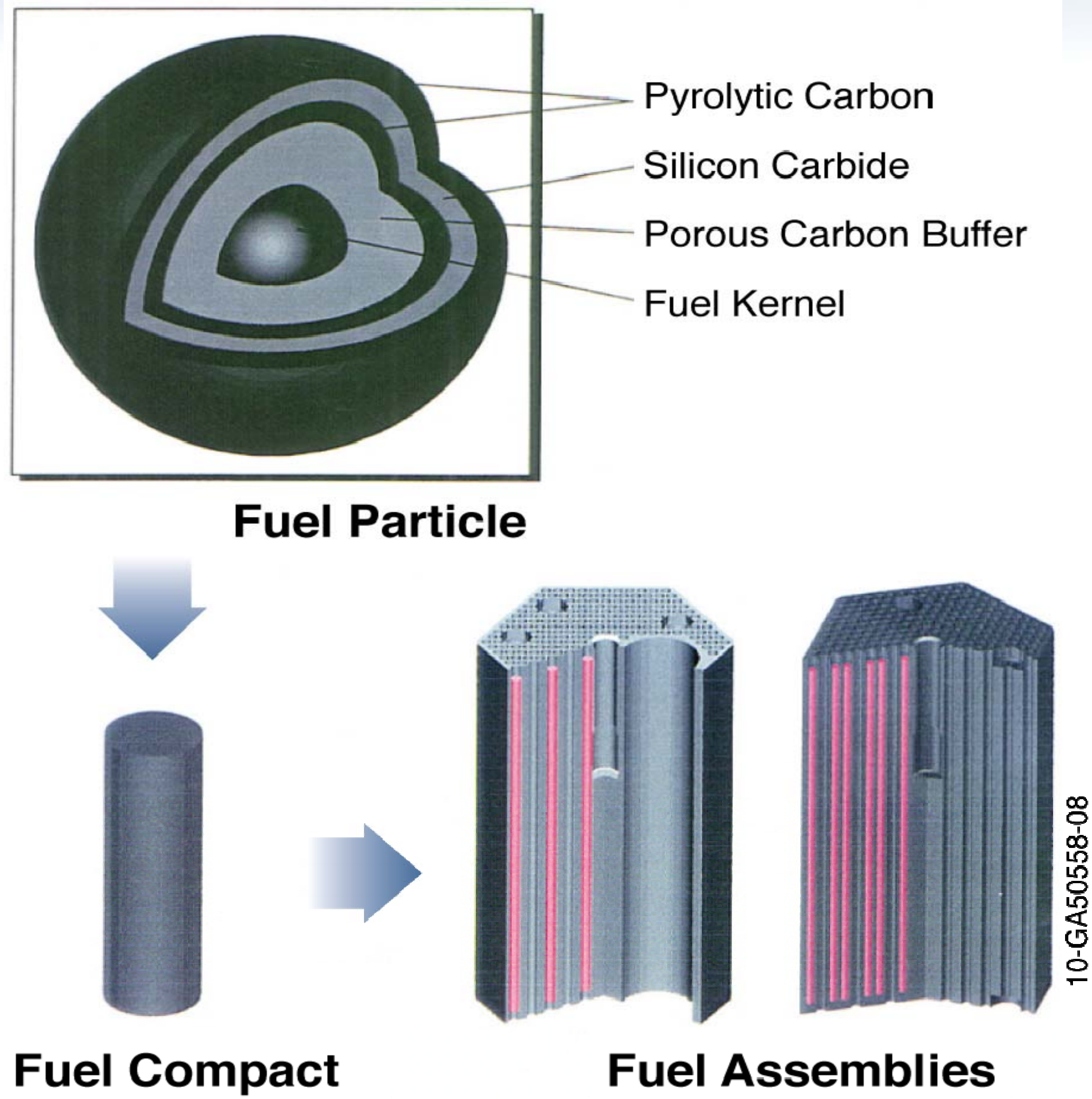
Material Graphite
102 Fuel Columns
Hexagonal Fuel Block Dimensions:
- Width Across Flats 0.36 m
- Height 0.8 m

Number of Fuel Blocks:
- Standard 720
- Control 120
- Reserve Shutdown 180

Number of Fuel Compacts 2919600
Mass 870 Tons



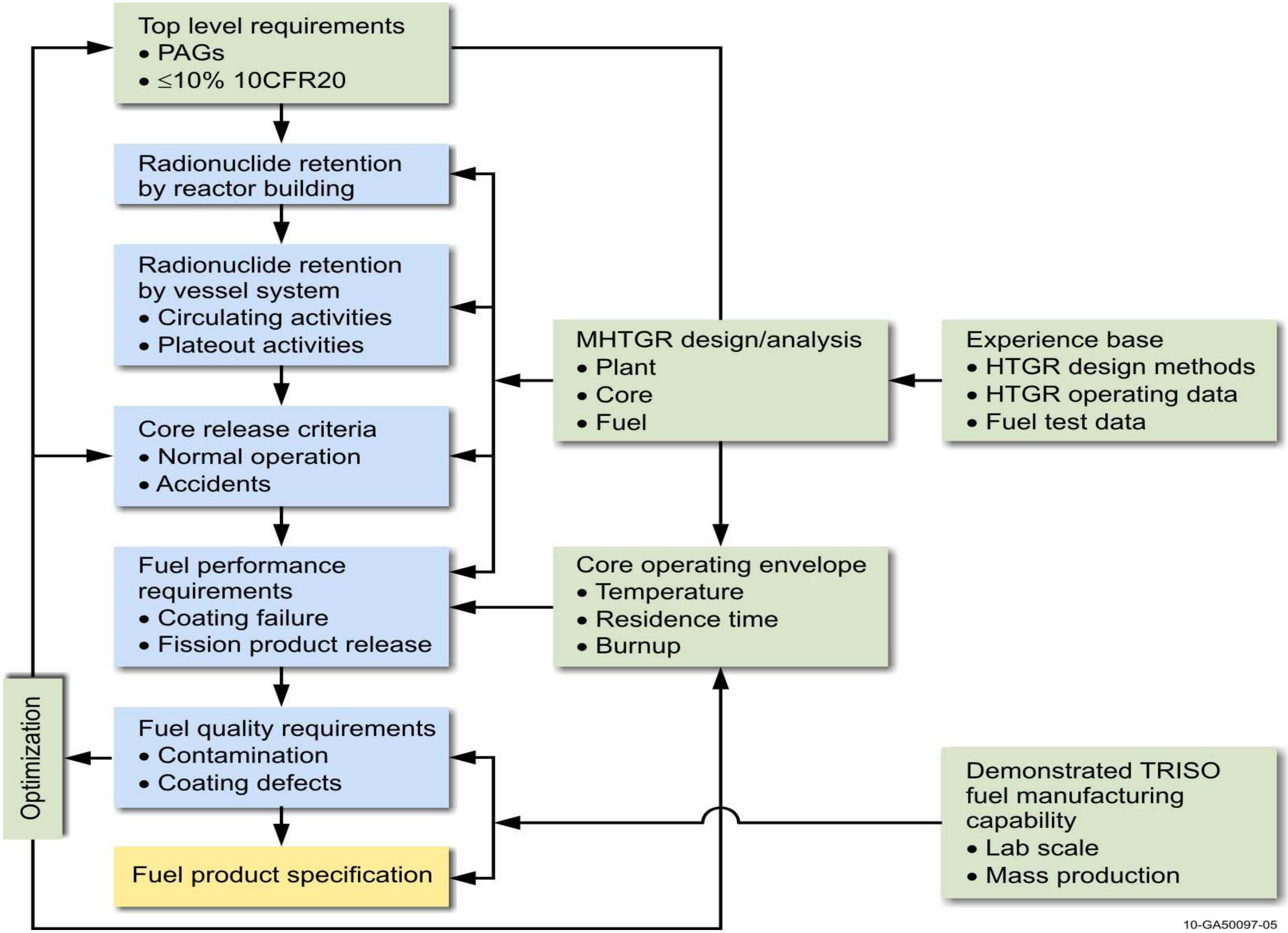
Prismatic Fuel



Prismatic Fuel – Anticipated Maximum Service Conditions for NGNP

Parameter	Maximum Value
Maximum fuel temperature – normal operation, °C	1,400
Maximum time averaged fuel temperature, °C	1,250
Fuel temperature (accident conditions), °C	1,600
Fuel burnup, % FIMA	17[a]
Fast fluence, 10^{25} n/m^2 (E > 0.18 MeV)	5
a. Estimated value for 14% enriched 425-μm reference fuel particle.	

Prismatic Fuel – Logic for Deriving Fuel Requirements



Preliminary Prismatic Fuel Performance Requirements

Parameter	NGNP – 750°C Core Outlet Temperature	
	Maximum Expected	Design
As-Manufactured Fuel Quality:		
HM contamination	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$
Missing or defective buffer	$\leq 1.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-5}$
Missing or defective IPyC	$\leq 4.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-4}$
Defective SiC	$\leq 5.0 \times 10^{-5}$	$\leq 1.0 \times 10^{-4}$
Missing or defective OPyC	0.01	0.02
In-Service Fuel Failure:		
Normal operation	$\leq 5.0 \times 10^{-5}$	$\leq 2.0 \times 10^{-4}$
Core heat-up accidents	$\leq 1.5 \times 10^{-4}$	$\leq 6.0 \times 10^{-4}$
Core Release Limits for Metals:		
^{137}Cs fractional release	$\leq 7.0 \times 10^{-6}$	$\leq 7.0 \times 10^{-5}$
^{110m}Ag fractional release	$\leq 2.0 \times 10^{-4}$	$\leq 2.0 \times 10^{-3}$

Fuel Qualification

Idaho National Laboratory

Fuel Qualification – Common Considerations

- Establishment of a fuel product specification
- Implementation of a fuel fabrication process capable of meeting the specification
- Implementation of statistical quality control procedures to demonstrate that the specification has been met
- Irradiation of statistically significant quantities of fuel with monitoring of in-pile performance and post irradiation examination to demonstrate that normal operational performance requirements are met
- Safety testing of statistically significant quantities of fuel to demonstrate that accident condition performance requirements are met
- Data from the programs are used to develop/improve models to predict fuel performance in the reactor

Fabrication: Pebble Bed and Prismatic

- Kernel
  - Enrichment
  - Diameter, density, stoichiometry, sphericity
    - Population means and upper and lower tolerance limits (*when applicable*)
  - Impurities in kernel
- Coating Layers
  - Thicknesses and densities (buffer, IPyC, OPyC, SiC)
    - Population means and upper and lower tolerance limits (*when applicable*)
  - Anisotropy of PyC layers
    - Upper limit
  - Faceting/sphericity
- Statistical methods used to ensure product meets specification at required confidence level

Fabrication: Pebble Bed Spheres (Fuel Element)

- Uranium loading
- Uranium homogeneity
- Fuel sphere carbon content
- Matrix graphite thermal conductivity
- Matrix graphite abrasion rate
- Matrix graphite corrosion rate at elevated temperature
- Matrix graphite coefficient of thermal expansion
- Impurities
- Exposed uranium (uranium not contained by intact SiC layer) via burn leach test
- Impact strength
- Crushing strength
- Diameter
- Fuel free shell thickness

Fabrication: Prismatic Compacts (Fuel Element)

- Uranium loading
- Uranium homogeneity
- Integrity and dimensions
- Crush strength
- Matrix density
- Heavy metal contamination by acid leach
- Defective SiC coating fraction
- Defective IPyC coating fraction (fuel dispersion)
- Defective OPyC fraction
- Impurities

Irradiation, PIE and Safety Testing: Pebble Bed

- German LEU UO_2 TRISO fuel irradiation and safety testing database provides a sound basis for projecting the performance of pebble bed fuel. Selected data will be used for fuel qualification
- Additional data to be generated include
  - As-manufactured pebble bed fuel characterization data to show that quality is at least equivalent to that of historic German fuel
  - Irradiation testing of pebble bed fuel that demonstrates fuel performance is consistent with historical German database
  - Safety testing at 1600 and 1800°C to provide additional data to establish accident fuel performance, fuel failure and associated margin within this envelope
    - The higher end of this envelope (e.g. 1800°C) is well beyond the anticipated transient and accident envelope

Pebble Bed Fuel Qualification Tests

- Plan established by PBMR based on including and building upon successful German LEU UO_2 TRISO fuel performance data
- The current status of PBMR causes uncertainty in details of the plan but not the objectives
- Objectives were focused on testing to *confirm* behavior of pebble bed fuel demonstrated in German testing
  - Performance demonstration irradiation of coated particles in AGR-2
  - Pre-qualification irradiation of early fuel from pilot fuel line
  - Full burn-up irradiation proof test demonstration (aka qualification)
    - Exact conditions anticipated to be typical of pebble bed reactors considering uncertainties (>9.8% FIMA; 900-1150°C, ~3.5×10^{25} n/m^2)
  - Safety testing of pebbles at 1600 and 1800°C to build upon existing database on accident performance
  - Post irradiation examination after irradiation and after safety testing to characterize fuel spheres, coated particles, and fission product behavior

Prismatic Fuel Qualification Tests

- The AGR Fuel Development and Qualification Program will qualify TRISO-coated UCO particle fuel and the associated specification
- TRISO-coated UCO particles will be fabricated at pilot scale for use in the formal qualification testing
- The testing program consists of irradiations, safety testing and post irradiation examinations that will characterize the behavior of TRISO-coated fuel under both normal and off-normal conditions
- Formal validation testing is also planned to validate fuel performance models required for core performance assessments and safety analysis

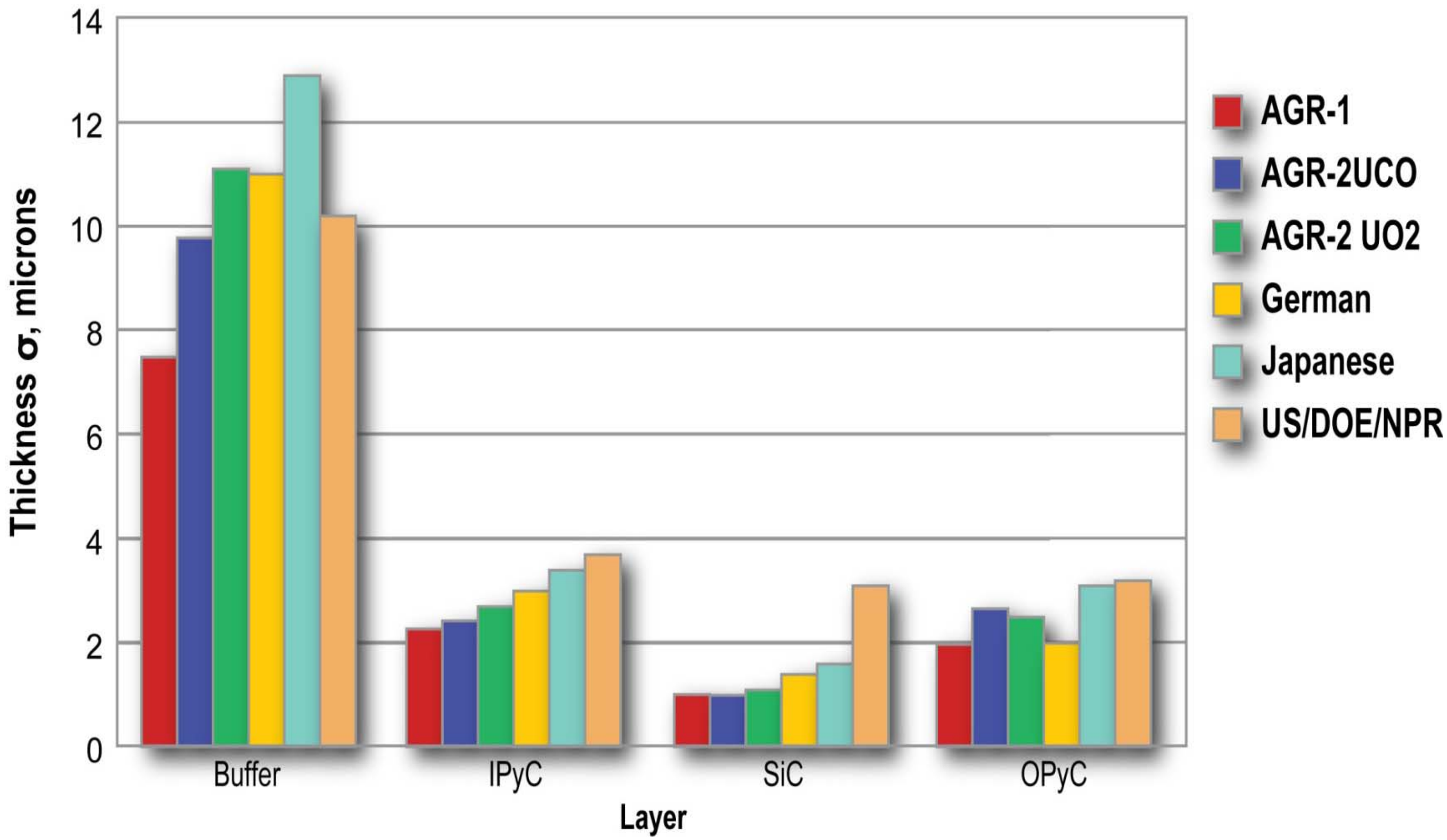
Fuel Qualification Irradiations

Capsule	Test Description	Test Objective/Expected Results
AGR-1	Shakedown Test/Early Fuel Performance Demonstration Test Contents included compacts made from UCO fuel particles coated in 2-inch laboratory scale coater at ORNL. A baseline fuel particle composite and three variant fuel particle composites were tested. The variants included two particle composites coated using different IPyC coating conditions and one particle composite coated using different SiC coating conditions.	Gain experience with multi-cell capsule design, fabrication, and operation to reduce chances of capsule or cell failures in subsequent capsules. Obtain early data on irradiated fuel performance and support development of a fundamental understanding of the relationship between fuel fabrication process and fuel product properties and irradiation performance. Provide irradiated UCO fuel for accident simulation testing (i.e. heating tests).
AGR-2	Fuel Performance Demonstration Fuel Contents to include compacts containing UCO particles made in large coater and UO_2 particles made by B&W, AREVA, and PBMR in different size coaters. AGR-2 will have 6 independently monitored and controlled capsules in a test train design essentially the same as demonstrated in AGR-1. One capsule of UCO fuel will be operated with a maximum time-averaged temperature of about 1400˚C as a performance margin test of the fuel.	Provide irradiation performance data for UCO and UO_2 fuel variants and irradiated fuel samples for PIE and post-irradiation heating test to broaden options and increase prospects for meeting fuel performance requirements and to support development of a fundamental understanding of the relationship between fuel fabrication process and fuel product properties and irradiation performance. Also, establish irradiation performance margin for UCO fuel.
AGR-5/6	Fuel Qualification Fuel specimens made by fuel vendor using process conditions and product parameters, based on best performance from successful AGR-1 and AGR-2 experience, using AGR program process development results and AGR-1, AGR-2 data. Variations in cell irradiation temperatures per test specification. .	Provide irradiation performance data for the reference fuel and irradiated fuel samples for PIE and post-irradiation heating tests in sufficient quantity to demonstrate compliance with statistical performance requirements under normal operation and accident conditions.
AGR-7	Fuel Performance Model Validation Contents to include same fuel type as used in AGR-5/6. The irradiation would test fuel beyond its operating envelope so that some measurable level of fuel failure would occur (i.e., margin test).	Provide fuel performance data and irradiated fuel samples for PIE and post-irradiation heating test and PIE in sufficient quantity to validate the fuel performance codes and models and to demonstrate capability of fuel to withstand conditions beyond AGR-5 and -6 in support of plant design and licensing.

Comparison of AGR-1 and AGR-2 Particles to Historical German data

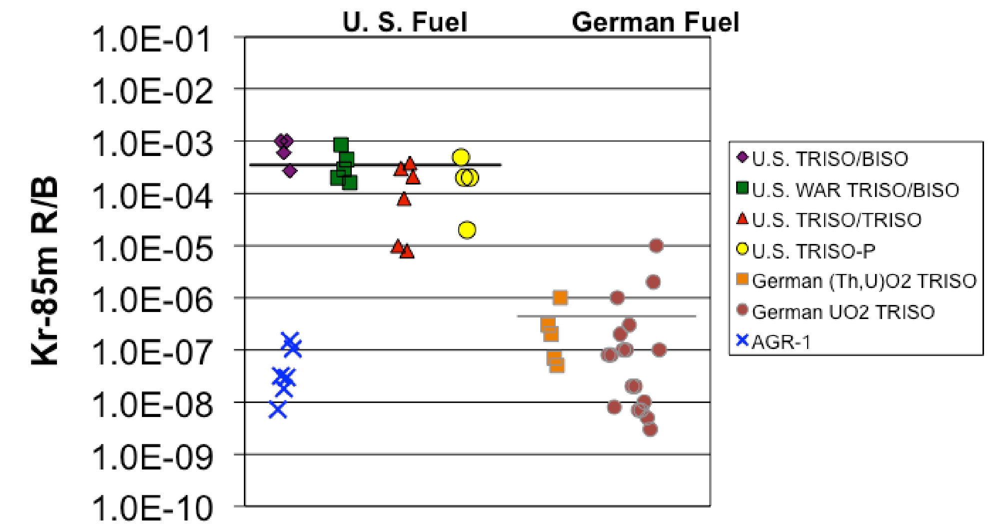
Mean properties	Buffer	IPyC	SiC	OPyC
Layer Thickness				
AGR-1, µm	102.4-104.2	39.4-40.5	35.0-35.9	39.3-41.1
AGR-2, µm	98.9	40.4	35.2	43.4
German, µm	92-102	38-41	33-36	38-41
Layer Density				
AGR-1 ρ, g/cm^3	1.1	1.90	3.208	1.90
AGR-2 ρ, g/cm^3	1.04	1.89	3.197	1.91
German ρ, g/cm^3	1.00-1.10	1.86-1.92	3.19-3.20	1.88-1.92
Anisotropy				
AGR-1 BAF_o		1.022/1.033		1.019/1.033
AGR-2 BAF_o		1.035/1.046		1.026/1.043
German BAF_o		1.042		1.024
Aspect Ratio				
AGR-1				1.055 ± 0.019
AGR-2			1.035 ± 0.011	1.051 ± 0.016
German			1.07 ± 0.02	1.09 ± 0.02

Standard deviations of AGR-1 and AGR-2 particle populations are as good as or better than historical fuels indicated better process control and higher characterization accuracy



AGR-1 Irradiation Demonstrates Performance Capability of UCO TRISO Fuel

- 300,000 particles irradiated under NGNP conditions to a peak burnup of 19% FIMA at peak temperatures <1250°C with no failures
  - Very low gas release from the experiment indicating high as-manufactured quality
  - No spikes on radiation monitoring equipment indicating no particle failures
  - Different fabrication conditions for PyC and SiC demonstrating robustness of fuel design and flexibility in fabrication process conditions
  - Currently undergoing post irradiation examination
  - Safety testing of this fuel anticipated later in 2011



	U.S.	German
Irradiation temperature (C)	930 - 1350	800 - 1320
Burnup (%FIMA)	6.3 - 80	7.5 - 15.6
Fast fluence (10^{25} n/m^2)	2.0 - 10.2	0.1 - 8.5

- Irradiation of engineering scale TRISO fuel (both UO_2 and UCO) is underway in the AGR-2 irradiation

Production Scale Fuel Qualification

- TRISO-coated UCO fuel activities in the AGR program are intended to develop and qualify a fuel manufacturing process that can be handed off to industry and serve as the foundation for commercial-scale TRISO-coated particle fuel production
- There is no fuel supplier at this time but options for developing the capability to produce fuel for NGNP have been established
- Once a production scale capability is established an irradiation proof test and safety testing will be needed to demonstrate acceptable performance of the fuel for NGNP
  - Testing will contain statistically significant quantities of fuel similar to that planned for AGR-5/6
  - These testing activities will occur in parallel with manufacture of the initial core fuel load for NGNP

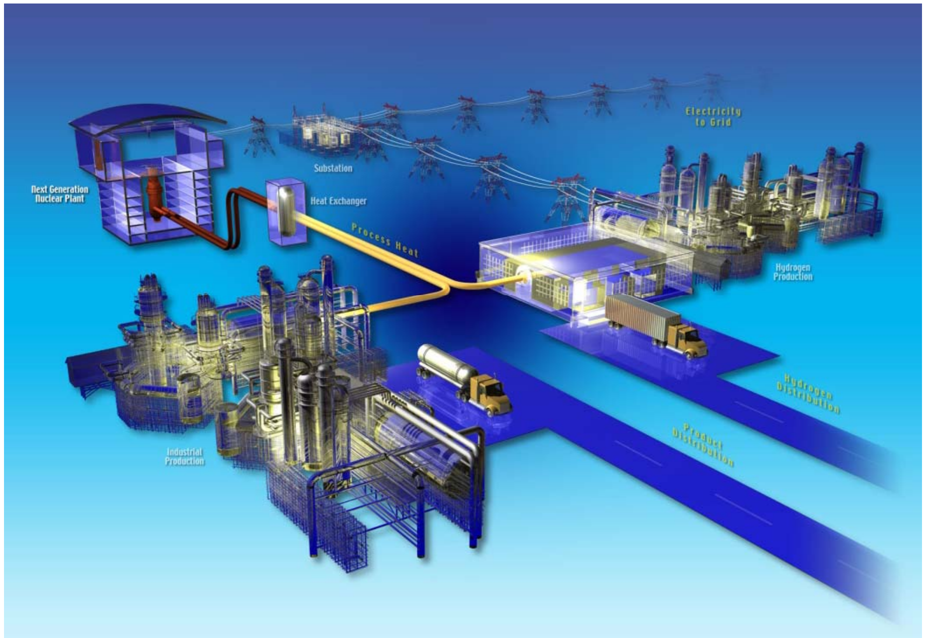
Outcome Objectives

- The primary issues for which feedback is requested include
  - Confirmation that plans established for qualification of the UO_2 pebble fuel type are generally acceptable
    - Utilization of German data for normal operation irradiation and transient/accident heat-up conditions
    - Performance of additional confirmatory irradiation and safety tests on fuel manufactured at a qualified facility to
      - (1) statistically strengthen the performance database and
      - (2) demonstrate that the fuel performs equivalent or better than the German fuel upon which the UO_2 pebble fuel design is based
  - Confirmation that plans established for qualification of the UCO prismatic fuel type are generally acceptable based on the NGNP/AGR Fuel Development and Qualification Program
  - Identification of any additional information or testing needed to meet NGNP fuel performance requirements

Mechanistic Source Terms White Paper

Presentation to NRC Staff by Next Generation Nuclear Plant Project

September 2, 2010



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Mechanistic Source Terms (MST) White Paper

July 2010

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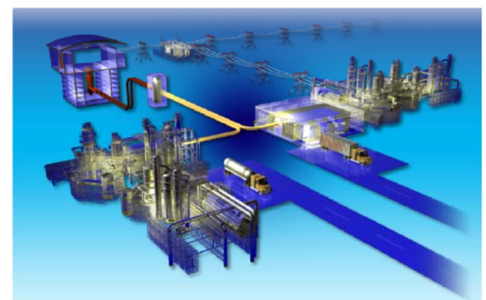
Mechanistic Source Terms White Paper

July 2010



Next Generation Nuclear Plant

Overview

- Introductions
- Mechanistic Source Terms
  - Introduction
    - Purpose
    - Issues for Discussion
  - Background
  - Regulatory Foundation
  - Approach to Mechanistic Source Terms
    - Radionuclide Transport and Retention
    - Fission Product Transport Codes Overview
    - Data from Operating HTGRs
  - Technology Development for Source Term Validation
  - Issues for Resolution
- Q&A
- Public Comment
- Adjourn

Introductions

Introduction (Section 1)

MST White Paper – Purpose (1.1)

- Define/describe proposed approach for developing event-specific MSTs for HTGR licensing

- Describe currently planned technology development programs needed to validate methods used to develop MSTs

- Obtain agreement from NRC that, subject to appropriate validation through the needed technology development program, the event-specific MST approach is acceptable

Issues for Discussion (1.3.1)

- Adequacy of the planned event-specific MST approach
- NRC feedback on any related issues that could significantly affect the NGNP COL schedule
- Specific agreement or feedback regarding
  - Acceptability of the definition of event-specific source terms for the HTGR
  - Acceptability of the MST calculational approach, subject to validation of design methods and supporting data
  - Acceptability of the approach of the planned fission product transport tests of the NGNP/AGR Fuel Development and Qualification Program, as supplemented by existing irradiation and accident testing data, to validate fission product transport analytical tools

MST White Paper is One of Three Closely Related White Papers (1.3.2)

- Fuel Qualification (Submitted to NRC on 7/21/10)
- Licensing Basis Event Selection (Planned for Sept 2010)
- Mechanistic Source Terms (Submitted to NRC on 7/21/10)

Background (Section 2)

HTGR Safety Basis (2.3)

- The objectives of the HTGR safety basis include:
  - Limit dose from releases so that regulatory requirements for protection of the health and safety of the public and protection of the environment* are met at an Exclusion Area Boundary (EAB) that is no more than a few hundred meters (e.g., 400 or 425 m) from the reactor
- This objective supports the licensing objective of establishing the facility plume exposure Emergency Planning Zone (EPZ) at the EAB, which provides for:
  - Reassessment of the requirements for emergency planning
  - Maximum flexibility in siting required for optimum use of the HTGR in co-located commercial applications (e.g., supply of steam, electricity, and process heat)
- To achieve these objectives, the HTGR is designed to:
  - Retain radionuclides at their source in the fuel
  - Limit the release of radioactive material

* Requirements are addressed in Section 4.2 of MST White Paper

HTGR Source Term Definition: (1 and 2.3.1)

- *Quantities of radionuclides released from the reactor building to the environment during Licensing Basis Events. This includes timing, physical and chemical forms, and thermal energy of the release.*

- HTGR Source Terms are:
  - Event-specific
  - Determined mechanistically using models of fission product generation and transport that account for reactor inherent and passive design features and the fission product release barriers
  - Different from the LWR source term that is based on a severe core damage event

HTGR Inherent/Passive Design Features (2.3.4)

- Large graphite moderator/reflector with high heat capacity
- Passive heat transfer path from fuel to ultimate heat sink
- Large, negative temperature coefficient of reactivity
- Low core power density
- Large core surface to volume ratio
- Helium coolant that is
  - Single phase
  - Chemically inert
  - Neutronically transparent
  - Low heat capacity/low stored energy

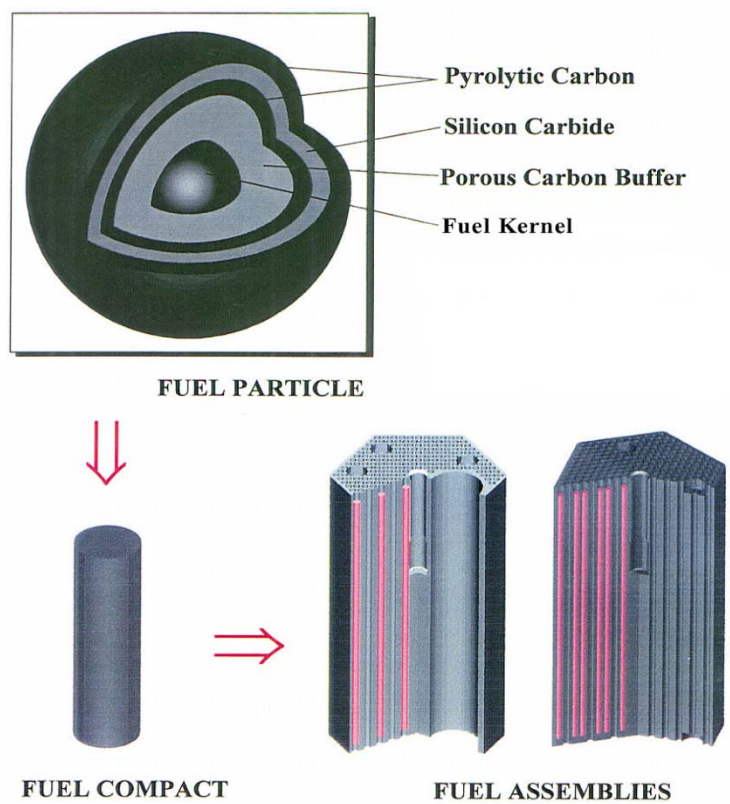
HTGRs Have Multiple Barriers to Fission Product Release that Provide a "Functional Containment" (2.3.2 and 2.3.3)

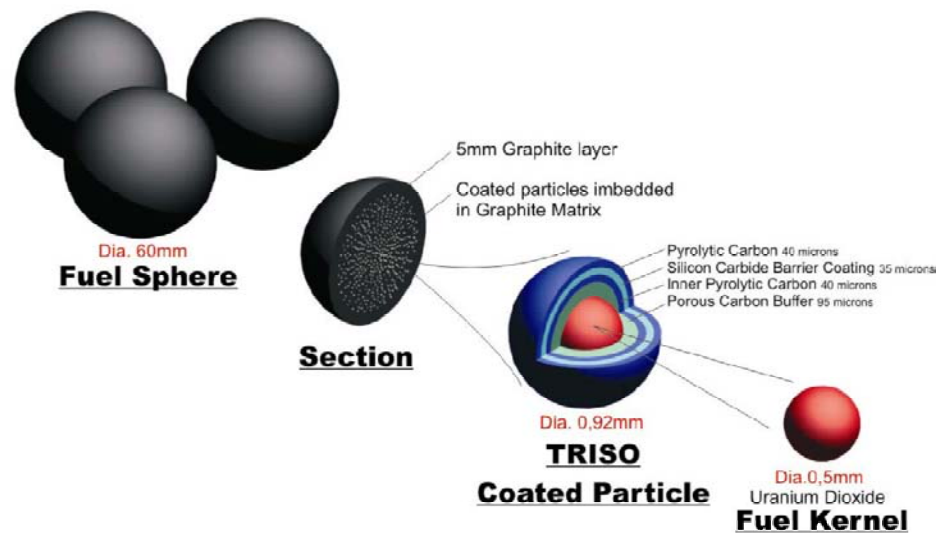
- Fuel Kernel
- Fuel Particle Coatings
- Matrix/Graphite

$\left.\right\}$ Prismatic or Spherical Fuel Element

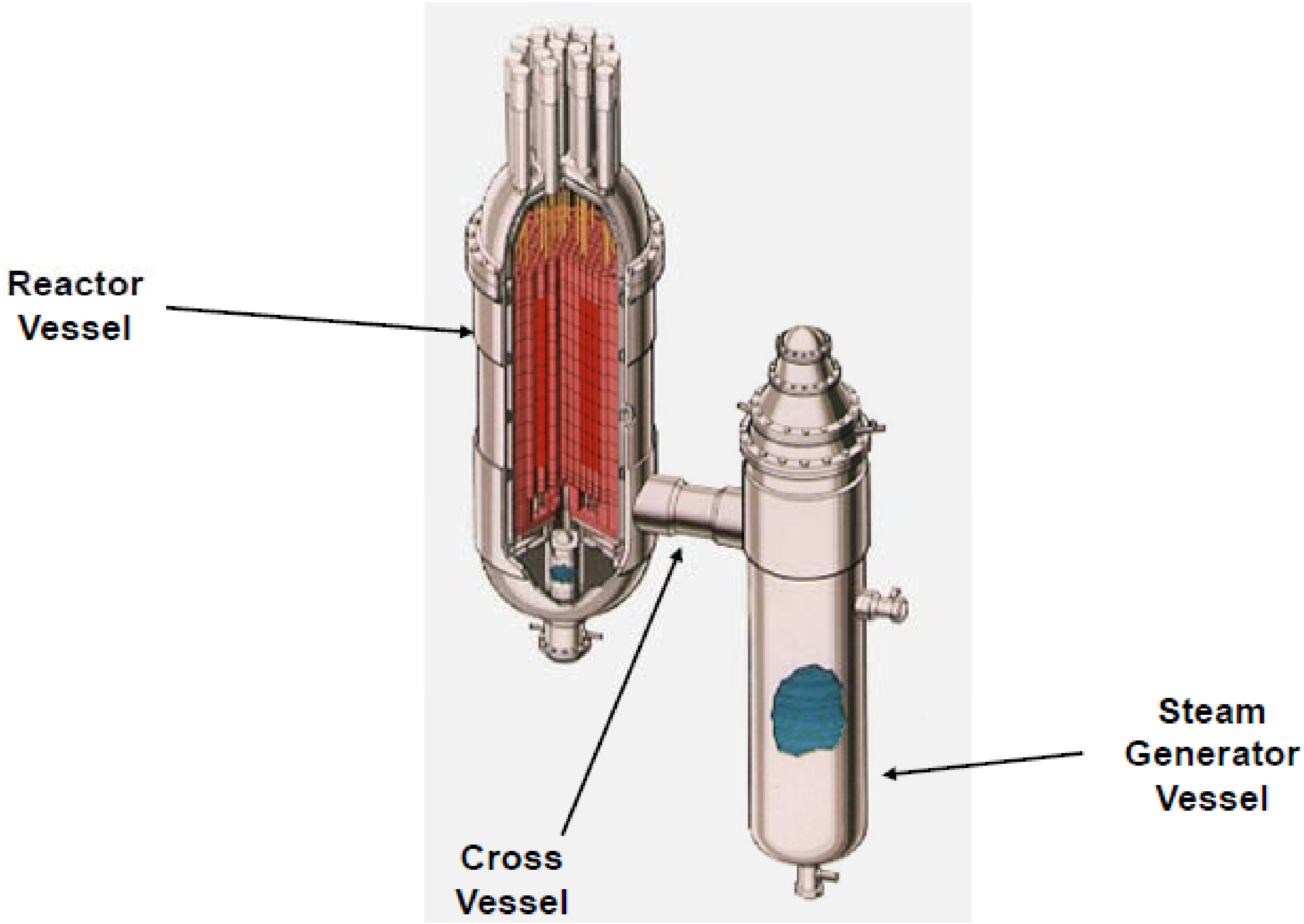
- Helium Pressure Boundary (Primary Circuit)
- Reactor Building

HTGR Fuel Designs (2.2)

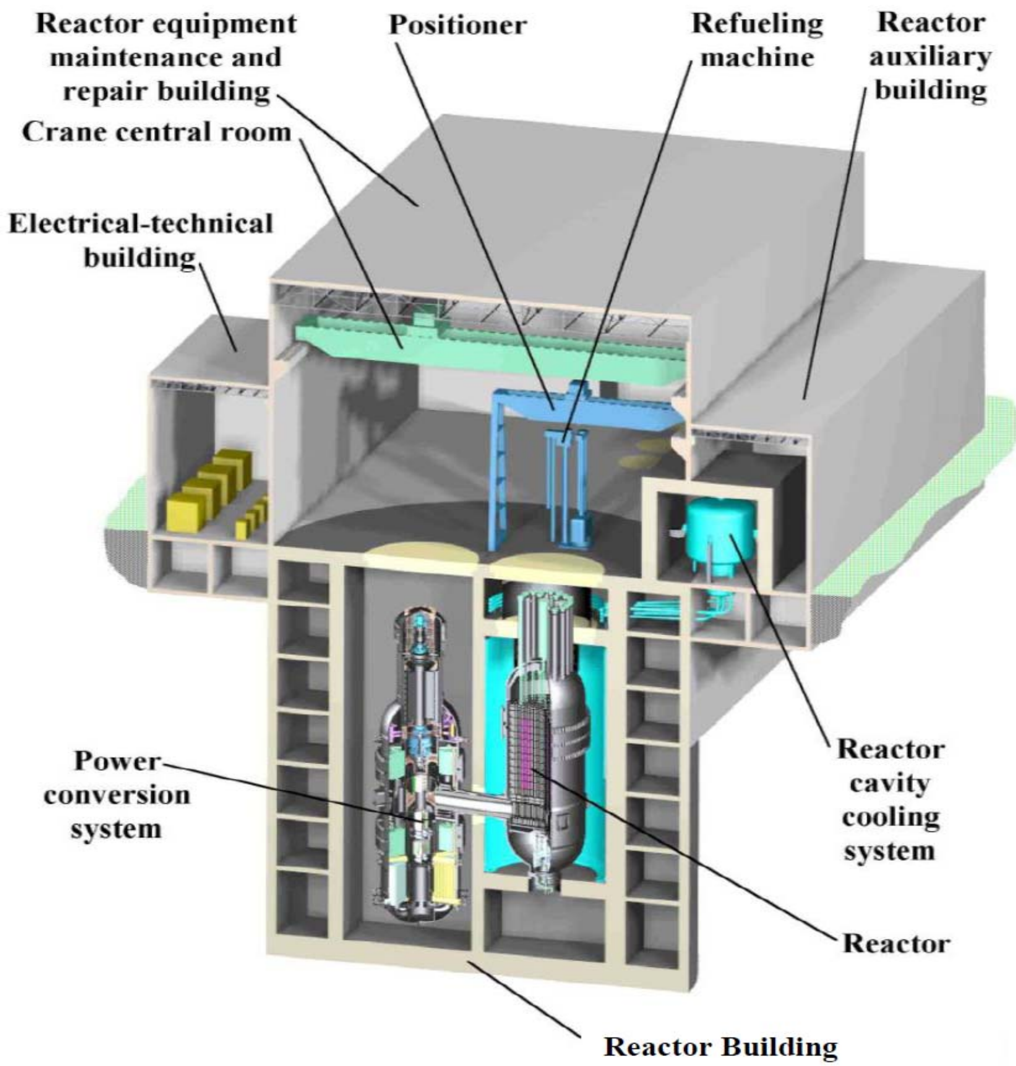




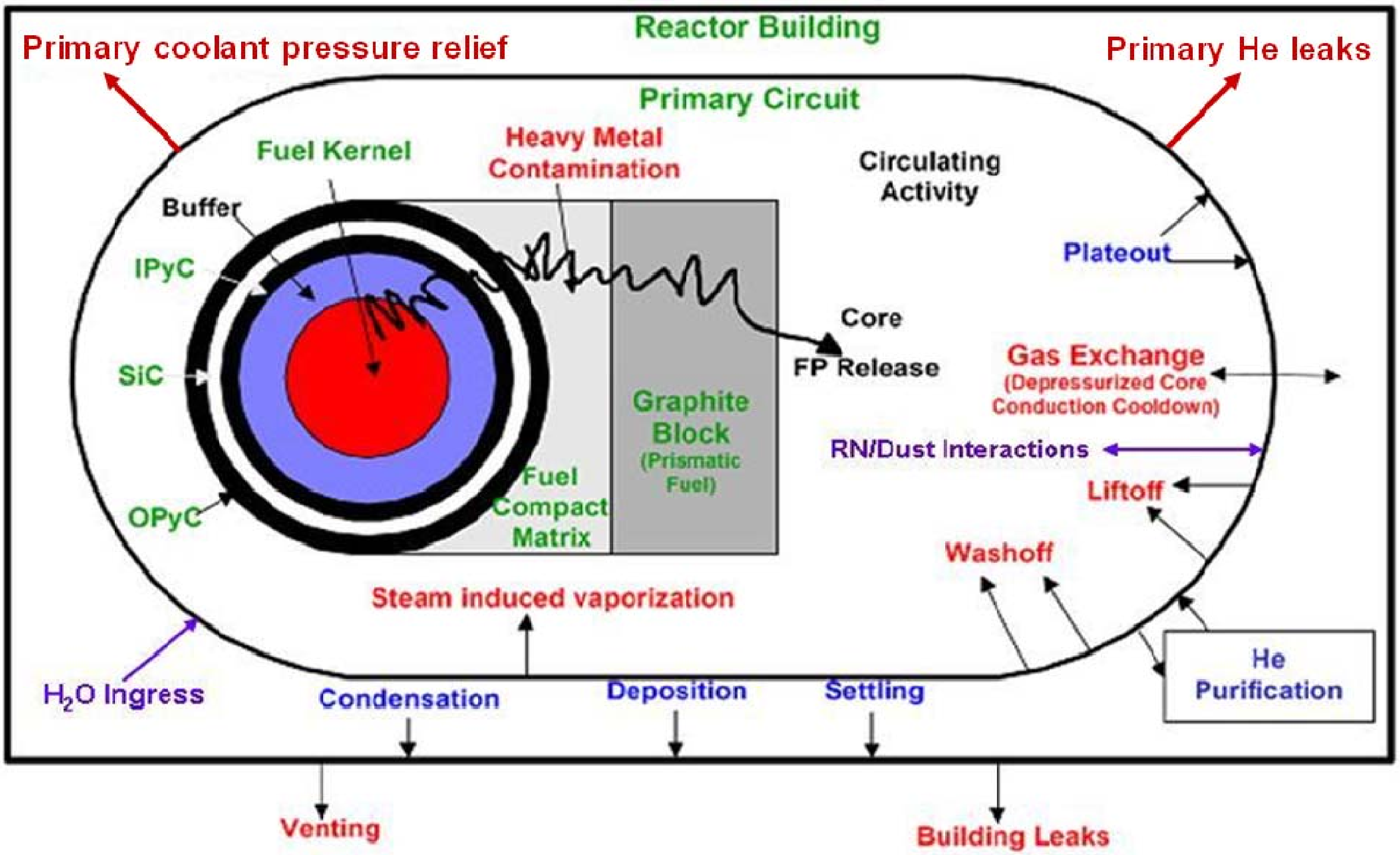
He Pressure Boundary (Primary Circuit) (App. A)



Typical HTGR Plant Layout (App. A)



HTGR Fission Product Retention System (2.3.3)



THE PHENOMENA ILLUSTRATED IN THIS FIGURE ARE MODELED TO DETERMINE MECHANISTIC SOURCE TERMS

Fission Product Transport Models Mechanistically Calculate (1.3.1)

- Transport of radionuclides from their point of origin through the fuel to the circulating helium
- Circulating activity in the primary circuit
- Distribution of condensable radionuclides in the primary circuit (plateout and dust)
- Radionuclide release to and distribution in the reactor building
- Radionuclide release from the reactor building to the environment (source term)

IN ADDITION TO PROVIDING SOURCE TERMS THESE CALCULATIONS PROVIDE RADIONUCLIDE INVENTORIES THROUGHOUT THE FACILITY

Calculated Radionuclide Inventories Throughout the Facility Are Used for Other Purposes

- Shielding and worker dose analyses
- Equipment environmental qualification
- Control room habitability analyses
- Assessments of accident risks in environmental impact statements

For the HTGR, the TRISO-Coated Fuel Particle is the Primary Barrier to Radionuclide Release (2.3.2)

- Low heavy metal contamination and low initially defective fuel particles in as-manufactured fuel (~1E-5)
- Minimal radionuclide release from incremental fuel failure during normal operation (~1E-4)
- Minimal radionuclide release from incremental fuel failure during Licensing Basis Events (~1E-4)
- Radionuclide release during LBEs dominated by exposed heavy metal (contamination and exposed fuel kernels)

Regulatory Foundation (Section 3)

NRC Regulations (3.1)

- Current NRC regulations are oriented to LWRs
- Regulatory adjustment is needed to support HTGR safety review
  - Example:
    - 10CFR52.17 (*Contents of Applications*) calls for an analysis of fission product releases that result from postulated events that include substantial core meltdown.
    - This is a LWR-specific requirement with no direct HTGR analogy. HTGR core does not "melt".
    - Comparable HTGR event entails limited incremental degradation or failure of fuel particle coatings without loss of cooling geometry.
- MST WP Table 3-1 notes applicability of some key NRC regulations.
- A comprehensive "Regulatory Gap Analysis" has been commissioned by the NGNP Project specifically for HTGRs.

SECY Positions on Mechanistic Source Terms (3.2)*

- Credit may be taken for unique aspects of plant design.
- Reactor and fuel performance and fission product behavior must be well understood over a wide range of scenarios (risk-informed/performance based).
- Fission product transport must be adequately modeled mechanistically for all barriers.
- Events analyzed must be bounding and account for design uncertainties.
- Fuel and plant performance must be maintained over the life of the plant.

* SECY-93-092 and SECY-03-0047

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*SECY Positions on Mechanistic Source Terms, cont'd.**

- Scenarios are to be selected from a design-specific PRA.
- Source term calculations must be based on verified analytical tools.
- Source terms for compliance should be 95% confidence level values based on best-estimate calculations.
- Source terms for emergency preparedness should be mean values based on best-estimate calculations.
- Source terms for licensing decisions should reflect scenario-specific timing, form, and magnitude of the release.

*SECY 05-006 – not yet approved by the Commission

Alternative Source Term (AST) (3.3.2)

- Provided for in 10CFR50.67 and Regulatory Guide 1.183
- AST must
  - Be based on major accidents resulting in hazards not exceeded by other credible accidents (substantial core meltdown with subsequent release of appreciable amounts of fission products)
  - Include times and rates of appearance of fission products released into containment, types and quantities of the species released, and chemical forms of iodine released
  - Represent a spectrum of credible severe accident events
  - Have a defensible technical basis supported by sufficient experimental and empirical data, be verified and validated, and be documented in a suitable form that facilitates public review and discourse
  - Be peer reviewed
- Specific aspects of these expectations are not directly applicable to the HTGR (e.g., core meltdown)
- Event specific HTGR MSTs will meet the intent of these expectations

US HTGR Precedents (3.5)

- Peach Bottom Final Hazards Summary Report (operated 1967-1974)
  - Conservative, time dependent source term based on mechanistic release phenomena
- Fort St. Vrain Final Safety Analysis Report (operated 1974-1989)
  - MST was compared to TID-14844 assumptions to demonstrate relative safety of the HTGR
- GASSAR-6 Safety Analysis Report (standard large HTGR plant, docketed 1975)
  - NRC evaluation of MST used conservative release model to circumvent shortcomings in existing fuel performance data
- DOE Modular HTGR PSID (NUREG-1338, draft-1989/final-1995)
  - Identified MST as licensability issue
- Pebble Bed Reactor preapplication submittals addressed HTGR technology issues supporting MST
  - Exelon (2000)
  - PBMR (Pty) Ltd (2006 – 2009)
  - Reviews were not completed

Approach to Mechanistic Source Terms (Section 4)

MST White Paper Contains Information on Radionuclide Transport and Retention in the HTGR (4.4, App.C)

- Radionuclide behavior in the fuel, primary circuit, and reactor building
- Radionuclide behavior models and modeling assumptions
- Sources of data on radionuclide behavior
- Experimental methods for data collection

Classes of Radionuclides of Interest in HTGR Design (4.4)

RN Class	Key Nuclide	Form in Fuel	Principal In-Core Behavior	Principal Ex-Core Behavior
Tritium	H-3	Element (gas)	Permeates intact SiC; sorbs on core graphite	Permeates through heat exchangers
Noble gases	Xe-133	Element (gas)	Retained by PyC/SiC	Removed by HPS
Halogens	I-131	Element (gas)	Retained by PyC/SiC	Deposits on colder metals
Alkali metals	Cs-137	Element	Retained by SiC; some matrix/graphite retention	Deposits on metals/dust
Tellurium group	Te-132	Complex	Retained by PyC/SiC	Deposits on metals/dust
Alkaline earths	Sr-90	Oxide-carbide	High matrix/graphite retention	Deposits on metals/dust
Noble metals	Ag-110m	Element	Permeates intact SiC	Deposits on metals
Lanthanides	La-140	Oxide	High matrix/graphite retention	Deposits on metals/dust
Actinides	Pu-239	Oxide-carbide	Quantitative matrix/graphite retention	Retained in core

Radionuclide Behavior During Normal Operation (4.4.1)

- Most radionuclides reach a steady state concentration and distribution in the primary circuit (long lived isotopes like Cs-137 and Sr-90 are exceptions – inventory builds up over plant life)
- Concentration and distribution are affected by:
  - Radionuclide half-life
  - Initial fuel quality
  - Incremental fuel failure during normal operation
  - Fission product fractional release from fuel kernel
  - Transport of fission products through particle coatings, matrix, and graphite
  - Fission product sorptivity on fuel matrix and graphite materials
  - Fission product sorptivity on primary circuit surfaces (plateout)
  - Helium purification system performance
  - Fission product interaction with dust in primary circuit

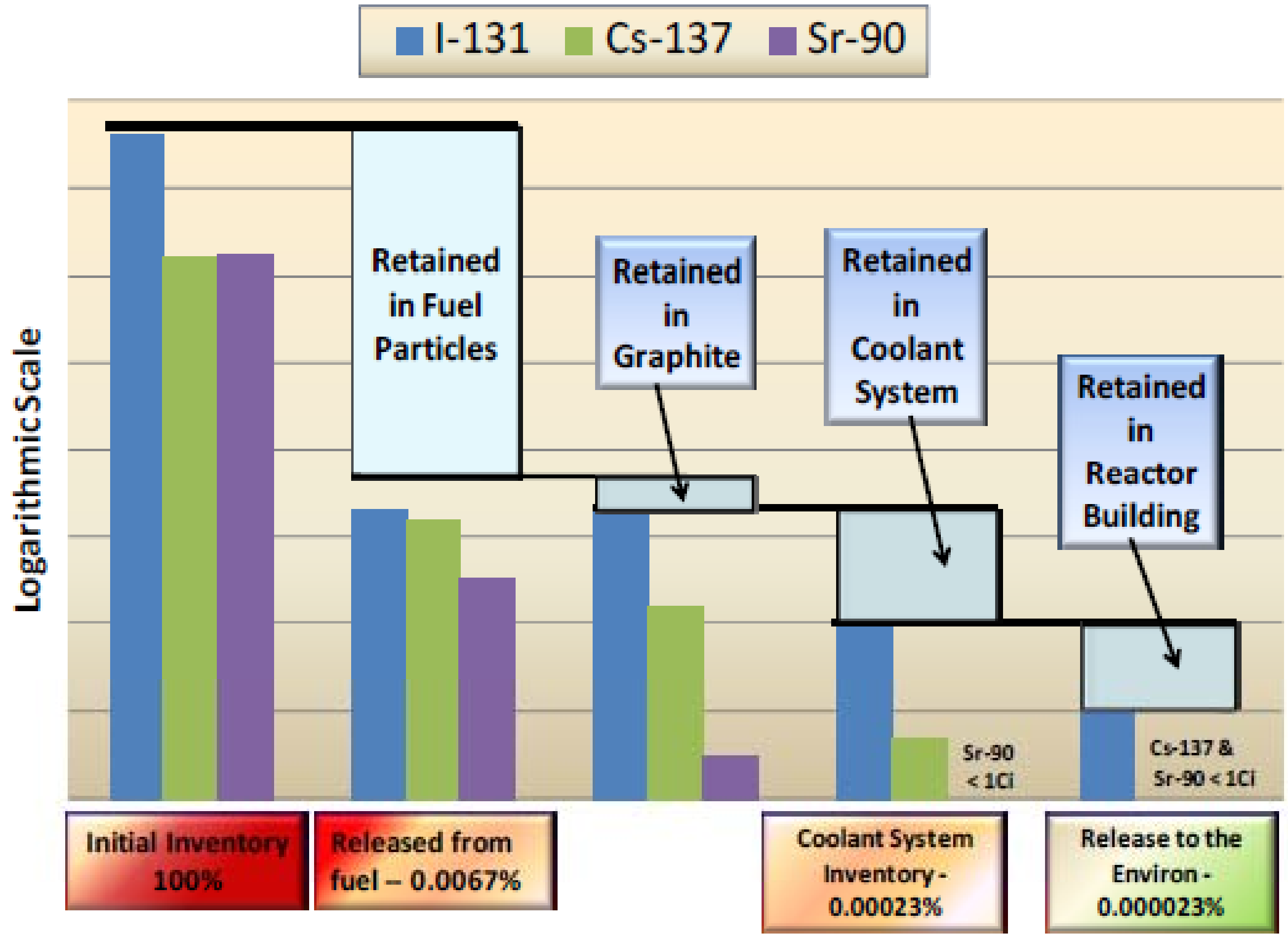
Radionuclide Behavior During Depressurization Events (4.4.2)

- Depressurization events may consist of three phases that can overlap, depending on the size of the postulated opening in the pressure boundary:
    1. Initial depressurization (minutes to hours)
    2. Subsequent core heatup (~100 hours) if forced cooling is lost
    3. Subsequent core cool-down
- Radionuclide release during Phase 1 is affected by:
    - Radionuclide content of the primary circuit (circulating and plated out)
    - Fraction of helium lost
    - Blowdown rate and shear ratio (size and location of pressure boundary failure)
    - Liftoff of plated out radionuclides
    - Release of contaminated dust
    - Fuel time at temperature (for slow blowdown rate)
    - Removal of radionuclides in the reactor cavity and other reactor building volumes
    - Reactor building venting

Radionuclide Behavior During Depressurization Events, cont'd

- Radionuclide release during Phases 2 and 3 (if forced cooling is lost) is affected by:
  - Fuel time at temperature
  - Residual blowdown rate (for slow depressurizations)
  - Heatup of residual helium in primary circuit
  - Gas exchange phenomena across helium pressure boundary and across reactor building
  - Flow dynamics among reactor building cavities
  - Reactor building leak rate
- Based on prior modular HTGR analyses, release during Phase 2 is expected to dominate the magnitude of the source term.

HTGR Functional Containment Performance for a Postulated DLOFC Event (4.4.3)



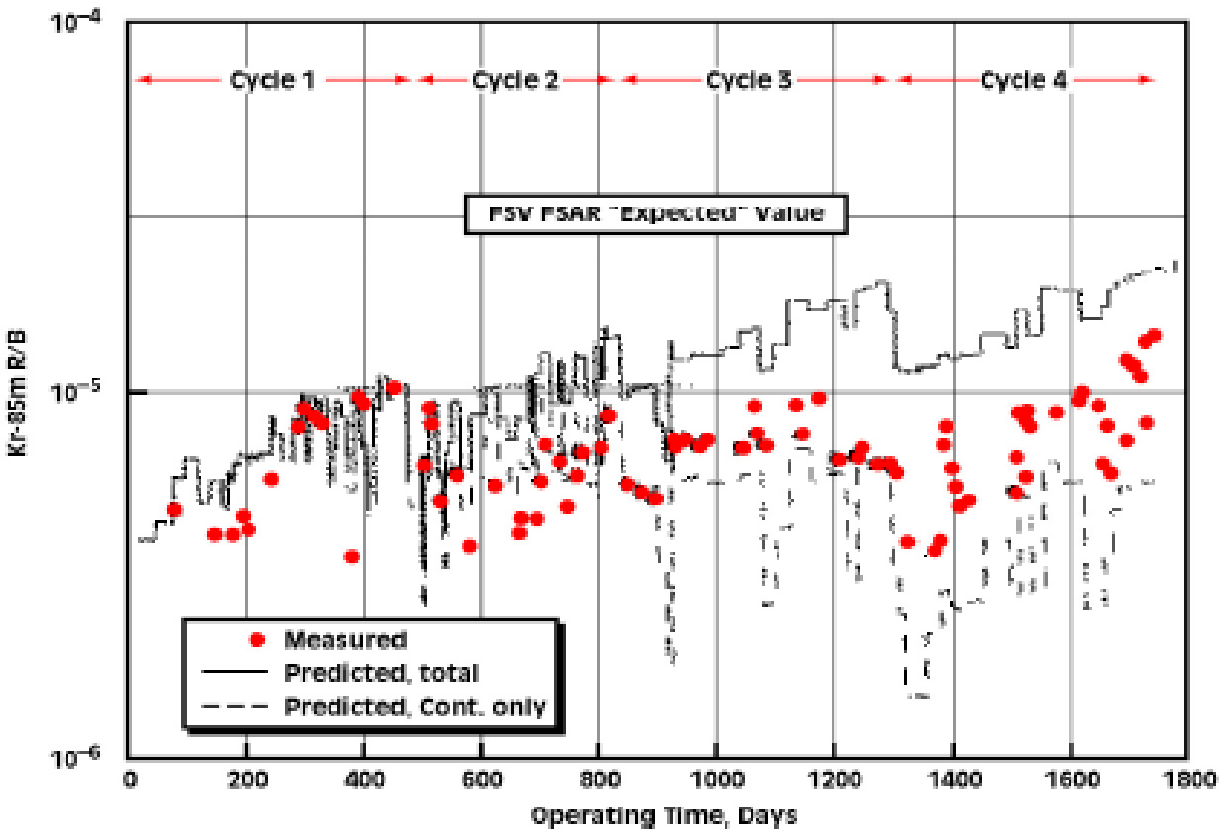
Source Term Determination: Uncertainty Allocation and Management (2.3.3)

A principal objective of the development of the source terms for all licensing basis events is to balance the degree of certainty required to characterize the retention of radionuclides by barriers "downstream" of the fuel with the tightness of the specification on fuel initial quality and the performance of the fuel under normal and accident conditions.

Various Fission Product Transport Codes are Used for Source Term Determination (4.5)

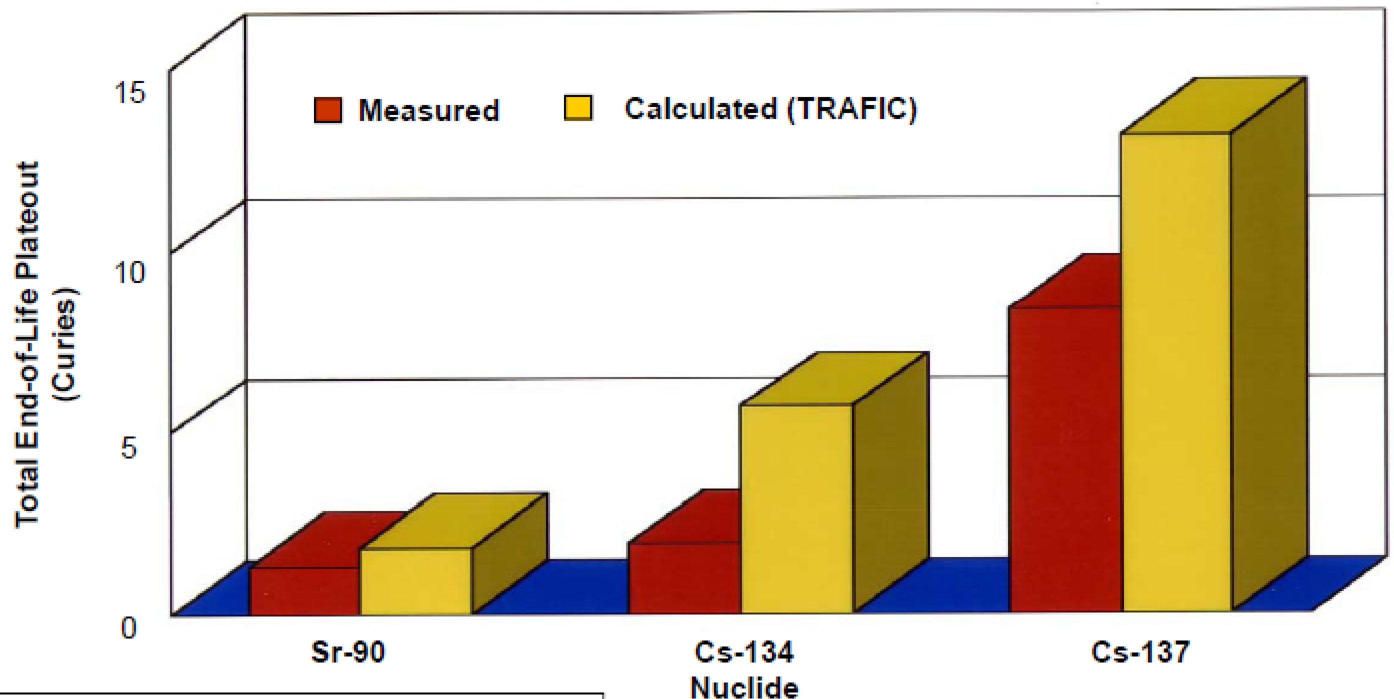
- Prismatic HTGR (GA)
  - SURVEY/POKE (Fuel Temperatures, Particle Failure Rates, Fission Gas Release – Normal Operation [NO])
  - TRAFIC-FD (Metallic Fission Product Release – NO)
  - PADLOC (Condensable Fission Product Plateout Distributions)
  - POLO (Liftoff and Reentrainment – Depressurization Events; Radionuclide Behavior in Reactor Building)
  - SORS (Fuel Performance and Fission Product Release – Core Heatup Events)
  - OXIDE (Fuel Performance for Moisture Ingress Events)
  - TRITGO (Overall plant tritium mass balance)
- Pebble Bed HTGR (PBMR (Pty), Limited)
  - NOBLEG (Fission Gas Release – NO)
  - FIPREX/GETTER (Metallic Fission Product Release – NO and LBEs)
  - DAMD (Plateout Distributions and Dust Behavior – NO)
  - ASTEC (Radionuclide Behavior in Reactor Building)

Comparisons of Calculated and Measured Fission Product Release: Fort St. Vrain Kr-85m R/B (4.6)



(Calculations using SURVEY code at GA)

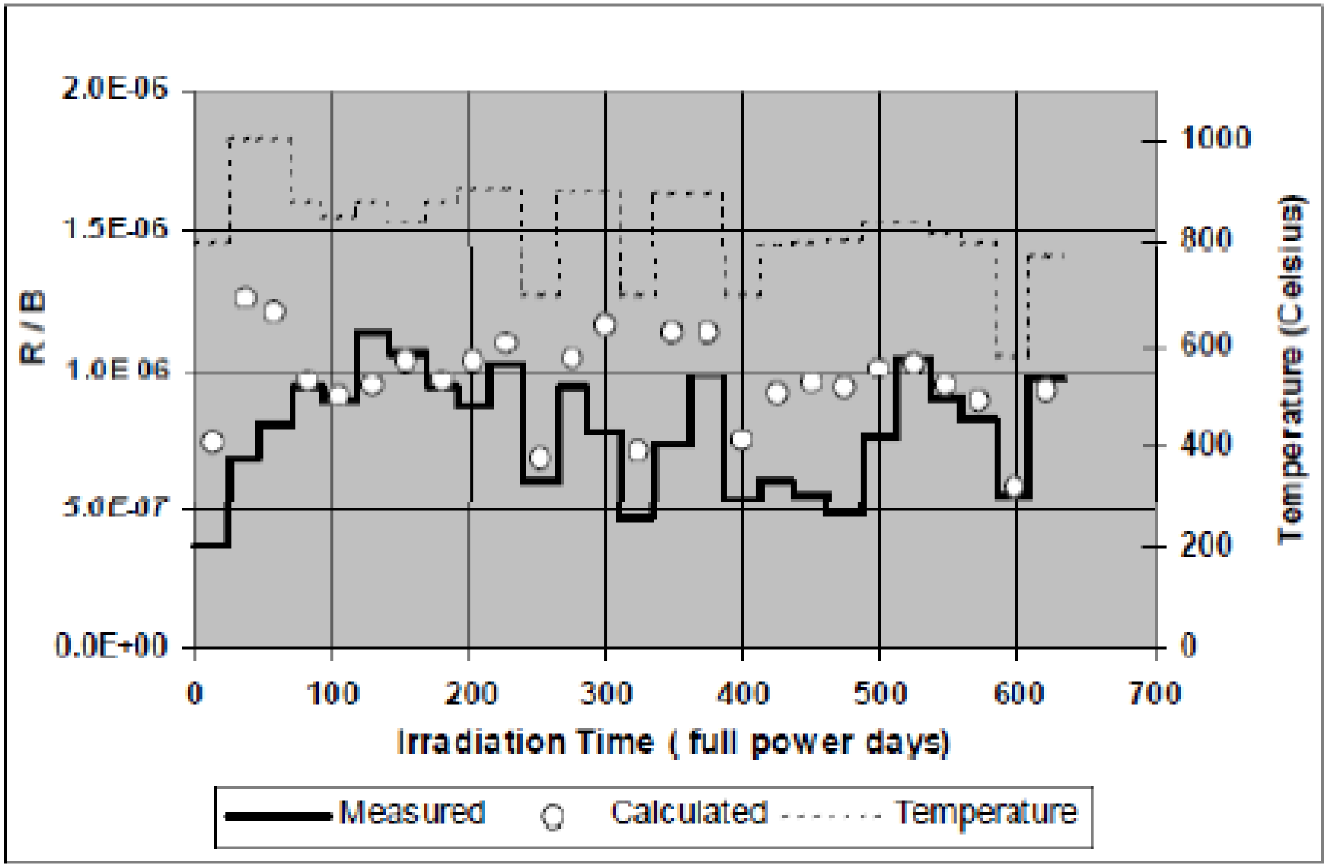
Comparisons of Calculated and Measured Fission Product Release: FSV Metallic Fission Products (4.6)



	Sr-90	Cs-134	Cs-137
Measured	1.3	1.9	8.4
Calculated	1.8	5.7	13.2

Calculations using TRAFIC Code at GA

Comparisons of Calculated and Measured Fission Product Release: Kr-88 R/B for Test Element C1K6 (4.6)



[Calculations using NOBLEG code at PBMR (Pty), Ltd]

Technology Development for Source Term Validation (Section 5)

Fission Product Behavior in HTGRs has been Studied Internationally for Several Decades (App. C)

- Significant data on radionuclide transport and retention have been assembled.
- The basic approach for modeling these phenomena is well established.
- However, the phenomena are complex.
- Uncertainties are, in some cases, larger than desired.
- Hence, fission product transport knowledge gaps exist.

Fission Product Transport Knowledge "Gaps" (5.1)

- Fission product release from intact and failed fuel particles under accident conditions
  - Reactor service conditions
  - Effects of moisture ingress
- Fission product transport and sorptivity in core materials (matrix and graphite)
  - Effective diffusion coefficients for some radionuclides
  - Reduced uncertainty in sorptivity
- Fission product plateout, liftoff, and washoff behavior
  - Data at more representative partial pressures
  - Data on more representative materials
- Radionuclide behavior in the presence of dust
  - Data on dust quantities and form
  - HTTR and HTR-10 data acquisition pending
  - Data also available from AVR and FSV
- Radionuclide behavior in the reactor building
  - LWR containment data generally not applicable
  - Suitable model needed – Sandia looking at MELCOR

The Fission Product Transport Portion of the AGR Fuel Plan Has Four Major Components (5.2.2)

- Irradiation and postirradiation accident testing of fuel containing a fraction of particles designed to fail (provides a known fission product source)
  - Fission product release and transport from particles
  - Normal operation and accident conditions

- Single effects tests in an out-of-pile loop
  - Fission product deposition and reentrainment (plateout and liftoff) on primary surfaces
  - Normal operation and accident conditions

- Improve and benchmark existing fission product transport models

- Validate transport models and codes with integrated irradiation and accident testing and, if determined by the project to be necessary, integral in-pile loop experiments

NGNP/AGR Program Plan for Fission Product Transport (5.2.2)

- AGR-3/4
  - Designed to fail (DTF) fuel particles
  - Concentric ring design to provide 1-D geometry to facilitate derivation of effective diffusivities in fuel matrix and graphite
  - Data on fission gas release from failed particles, fission metal diffusion in kernels, fission gas and metal diffusion in coatings, and fission product retentiveness of graphite matrix under normal and accident (postirradiation heatup) conditions
- AGR-8
  - DTF fuel particles
  - Piggyback design for different irradiation temperatures
  - Temperature cycling (TBD)
  - Temperature, fluence, burnup conditions enveloping NGNP
- Additional tests to be developed to assess moisture ingress effects

Other Source Term Validation Tests (Planned or Under Consideration) (5.2.2)

- Laboratory Experiments and Out-of-Pile Test Loop
  - Separate effects
  - Fission product behavior in primary coolant loop and reactor building
    - Plateout
    - Liftoff/Washoff
    - Dust effects

- Integral Loop Demonstration, if needed (under discussion within project)
  - Integrated irradiation and accident behavior data
  - Simulated primary coolant loop and reactor building
    - Plateout
    - Liftoff/Washoff
    - Dust effects

Issues for Resolution (Section 6)

NRC Agreement is Sought That: (6)

- The definition of the mechanistic source term is acceptable.
- The approach to calculation of mechanistic source terms, which includes a radionuclide retention concept ("functional containment") that includes the multiple barriers discussed in the white paper is acceptable, subject to validation of design methods and supporting data.
- The approach of the planned fission product transport tests of the NGNP/AGR Fuel Development and Qualification Program, supplemented by existing irradiation and accident testing data, is acceptable for validation of the fission product transport models that support determination of mechanistic source terms.

Q & A