High Temperature Gas-cooled Reactor Technology Training Curriculum

Presented by Idaho National Laboratory
July 16-17, 2019

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Day 1 – Tuesday July 16, 2019		
Time	**Topic**	**Presenter**
8:30	High Temperature Gas-cooled Reactor: Introduction • Motivation and Applications for HTGRs • High Level HTGR Design and Safety Approach	Hans Gougar
8:50	High Temperature Gas-cooled Reactor: History • Overview: U.S., World Experience (Experimental, Demo, or Commercial) • Evolution of HTGRs • Lessons Learned	Hans Gougar
9:30	*Break*	

9:45	High Temperature Gas-cooled Reactor: Core Design
	• General Attributes of Modular Prismatic and Pebble Bed HTGRs
	  ○ Physics
	  ○ Neutronics
	  ○ Prismatic and Pebble Fuel
	  ○ Thermal-Fluidics
	  ○ Inherent Safety
	• Plant Systems and Power Conversion
	  ○ Reactivity Control
	  ○ Instrumentation and Control
	  ○ Helium Conditioning
	  ○ Power Conversion
	• Normal Operation and Power Maneuvers
		Hans Gougar

11:30	*Lunch*	
12:30	TRISO Fuel: Design, Manufacturing, and Performance • Background and History • Fabrication and Quality Control • Irradiation Performance • Accident Performance • Fuel Performance Modeling	Paul Demkowicz
1:45	Modular High Temperature Gas-cooled Reactor: Safety Design Approach • HTGR Design Criteria • Inherent and Passive Safety • Prevention vs. Mitigation • Radionuclide Sources/Barriers • Residual Heat Removal • Reactivity Control • Reactor Building	Jim Kinsey

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Time	**Topic**	**Presenter**
2:45	*Break*	
3:00	Modular High Temperature Gas-cooled Reactor: Accident Analysis • Types of Potential Accidents • Reactor Response • Safety Analysis Approach • Codes and Tools • Experimental Validation	Hans Gougar
4:00	Modular High Temperature Gas-cooled Reactor: Accident Analysis (continued) • Licensing Modernization Project • Use of PRA in LMP, ASME/ANS Non-LWR PRA Standard • Methods for Incorporating Passive System Reliability into a PRA	Jim Kinsey
5:00	*Adjourn*	

Day 2 – Wednesday July 17, 2019		
Time	**Topic**	**Presenter**
8:30	TRISO Fuel: Mechanistic Source Term • Radionuclide Barriers • Radionuclide Design Criteria • Computational Tools • Source Term Estimation	Paul Demkowicz
9:30	Modular High Temperature Gas-cooled Reactor: Licensing Experience • Past US HTGRs Licensing Approach • Summary of NGNP Experience	Jim Kinsey
10:00	*Break*	
10:15	Modular High Temperature Gas-cooled Reactor: Licensing Experience (cont.) • NRC Regulatory Approach Assessment (Next Generation Nuclear Plant)	Jim Kinsey
11:00	High Temperature Gas-cooled Reactor: Materials • Nuclear Graphite Components • Structural Alloys for HTGR and VHTR Systems • Component Design (Materials and Applications)	Richard Wright
12:00	*Lunch*	
1:00	Group Discussion and Review	Hans Gougar
2:15	Overview and Concluding Remarks	Hans Gougar

Acronym List

AC	alternating current
AF	attenuation factors
AGR	Advanced Gas Reactor
AHTR	Advanced High-Temperature Reactor
ALARA	low as reasonably achievable
ANL	Argonne National Laboratory
ANS	American Nuclear Society
AOO	anticipated operational occurrences
ART	Advanced Reactor Technologies
ASTM	American Society for Testing and Materials
AVR	Arbeitsgemeinschaft Versuchsreaktor
BDB	beyond design basis
BDBE	beyond design basis events
BWXT	BWX Technologies, Inc.
CDF	core damage frequency
CFD	computational fluid dynamics
Ci	curries
CO	carbon monoxide
CR	control rod
CRP	Coordinated Research Program
CTE	coefficient of thermal expansion
D&D	decontamination and decommissioning
DBA	design basis accident
DBE	design basis events
DID	defense in depth
DLOFC	depressurized loss of forced cooling
DOE	Department of Energy
EAB	exclusion area boundary
ECCS	eliminating emergency core cooling system
EPA	Environmental Protection Act
EPRI	Electric Power Research Institute
EU	European Union
F-C	frequency-consequence

FFC	fuel and fuel cycle
FGMS	Fission Gas Monitoring System
FHM	Fuel Handling Machine
FIMA	fissions per initial metal atom
FP	fission product
FSV	Fort St. Vrain
GA	General Atomics
GE	General Electric
GFR	Gas-cooled Fast Reactor
GIF	Generation IV International Forum
GTAW	gas-tungsten arc welding
HFIR	High Flux Isotope Reactor
HFR	High Flux Reactor
HMC	heavy metal contamination
HPB	helium pressure boundary
HPS	Helium pressure systems
HT&SS	Helium Transfer and Storage System
HTGR	high-temperature gas-cooled reactors
HTR-PM	high-temperature gas-cooled reactor – Pebble Bed Module
HTS	heat transport system
HTTF	High Temperature Test Facility
HTTR	High Temperature Engineering Test Reactor
IAEA	International Atomic Energy Agency
IHX	intermediate heat exchanger
IMGA	Irradiated Microsphere Gamma Analyzer
IPD	Integrated Decision Panel
IPyC	inner pyrolytic carbon
ISO	International Organization for Standardization
IVV-2M	Russian reactor in Zarechny Russia
LBE	licensing basis event
LERF	large early release fraction
LMP	Licensing Modernization Project
LWR	light water reactor
mHTGR	modular high temperature gas-cooled reactor
MHTGR	modular high temperature gas-cooled reactor (Specific to General Atomics Design)

MTR	materials test reactors
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NEUP	Nuclear Energy University Program
NGNP	Next Generation Nuclear Plant
NPR	new productive reactor
NRC	Nuclear Regulatory Commission
NSRST	non-safety-related with special treatment
NST	no special treatment
NSTF	natural circulation shutdown heat removal facility
NUREG	nuclear regulatory guide
OPyC	outer pyrolytic carbon
ORIGEN	
OSU	Oregon State University
PAGs	protective action guides
PARFUME	PARticle FUel ModEl
PB1	Peach Bottom Unit 1
PBMR	Pebble Bed Modular Reactor (Pty) Limited (Africa)
PIRT	Phenomena Identification and Ranking Tables
PLOFC	pressurized loss of forced cooling
PRA	probabilistic risk assessment
PRISM	power reactor inherently safe module
PV	pressure vessel
QA	quality assurance
QAPD	quality assurance program description
QC	quality control
QHO	qualitative health objective
R&D	research and development
R/B	release-rate-to-birth-rate
RB	reactor building
RCCS	reactor cavity cooling system
RIDM	risk-informed integrated decision-making
RIPB	risk-informed and performance based
RIPB	risk-informed, performance-based
RN	radionuclide

RPV	reactor pressure vessel
SAFDL	specified acceptable fuel design limit
SARRDL	system radionuclide release design limits
SC-HTGR	steam cycle high temperature gas-cooled reactor
SCS	shutdown cooling system
SECY	office of the secretary
SG	steam generator
SiC	standard industrial code
SR	safety-related
SRCs	Structural Reliability Classes
SSC	structures, systems, and components
TECDOC	IAEA technical document
TEDE	total effective dose equivalent
TRISO	tristructural isotropic
UK	United Kingdom
VHTR	very high-temperature reactor
VHTRC	very high-temperature reactor critical

High Temperature Gas-cooled Reactor: Introduction

Hans Gougar

High Temperature Gas-cooled Reactor: Introduction

Advanced Reactor Technologies

Idaho National Laboratory

Hans Gougar, PhD

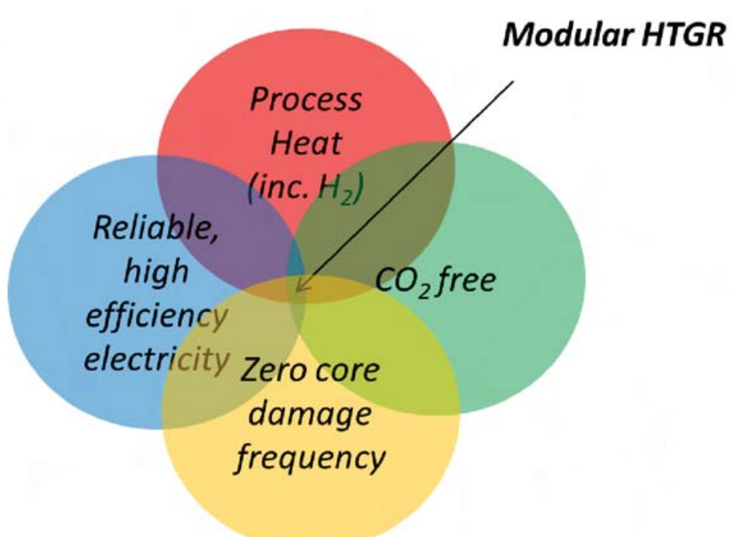
Nuclear Engineer

NRC HTGR Training July 16-17, 2019

ART ADVANCED REACTOR TECHNOLOGIES

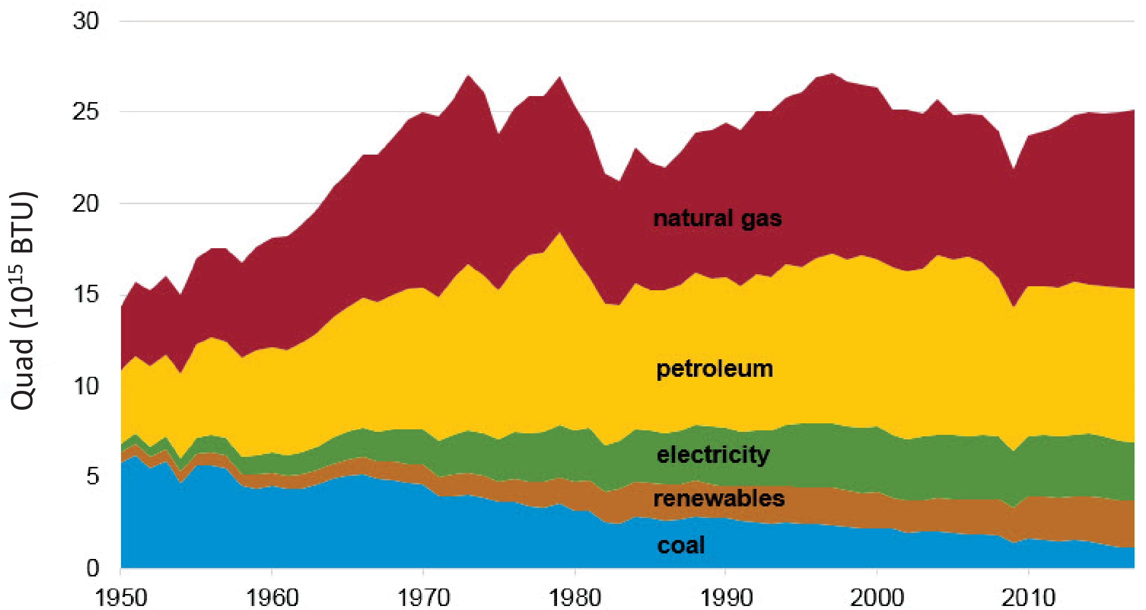
Modular High Temperature Gas-cooled Reactors

- Inherently safe core cannot melt
- High outlet temperature for more efficient electricity production and process heat
- Minimal radiological or dynamic coupling between the reactor and the collocated process heat application
- Environmentally benign, reliable, mature (for a non-LWR)



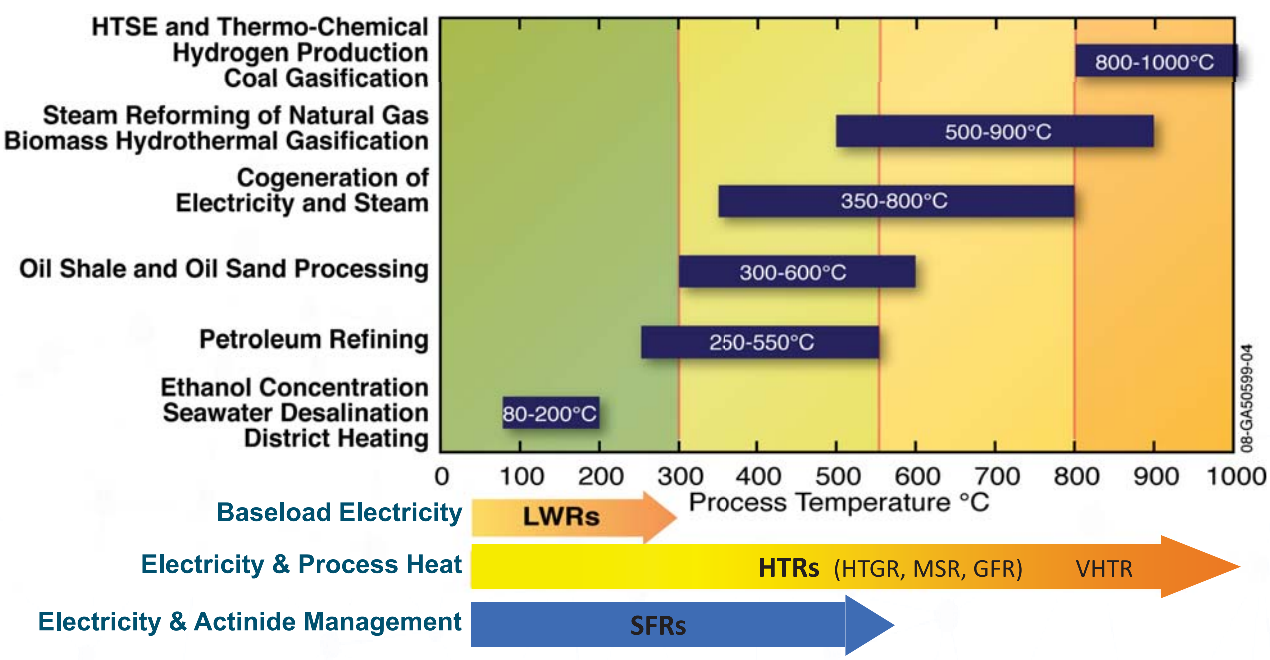
Electricity Fraction of Industrial Energy Use

U.S. industrial sector energy use by source, 1950–2017



Note: Includes energy sources used as feedstocks in manufacturing products.
Electricity is retail purchases. Renewables are mainly biomass.
Source: U.S. Energy Information Administration, *Monthly Energy Review*, Table 2.4, May 2018

Some Process Heat Applications suitable for Nuclear



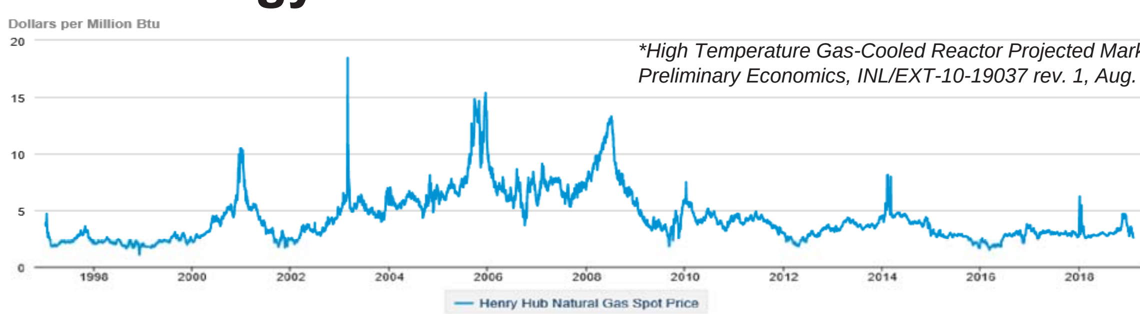
Potential Markets for Modular HTGR Steam

Business Subsector	Target Industry	Required heat input (MWt) between 300°C and 850°C	Number of 150 MWt HTGRs Required
Petroleum and Coal Products	Refineries	13456	399
Primary Metal Manufacturing	Iron and Steel mills	3225	226
Chemical Manufacturing	Basic Chemical Manufacturing (Methanol)	12714	85
	Ethyl Alcohol	3448	23
	Plastics Material and Resin	8780	60
	Alkalies and Chlorine	545	4
	Fertilizer (Ammonia)	2448	16
Food Manufacturing	Wet Corn Milling	2239	15
Mining (exc. oil & gas)	Potash, Soda, Borate	3318	22

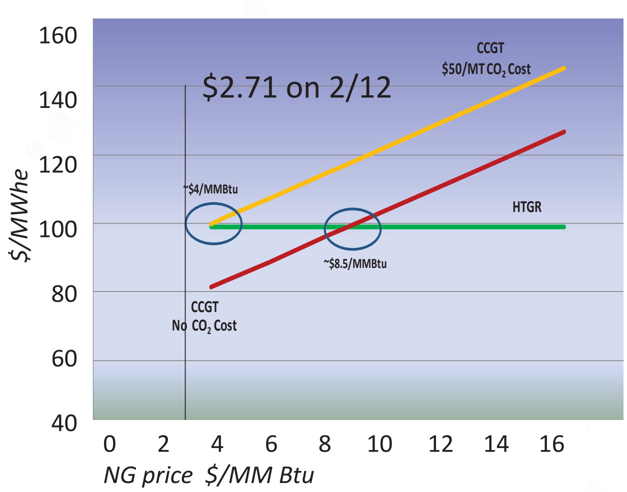
McMillan, C. et al, "Generation and Use of Thermal Energy in the U.S. Industrial Sector and Opportunities to Reduce its Carbon Emissions", NREL/TP-6A50-66763, INL/EXT-16-39680

Cost of Energy – HTGR vs. Natural Gas

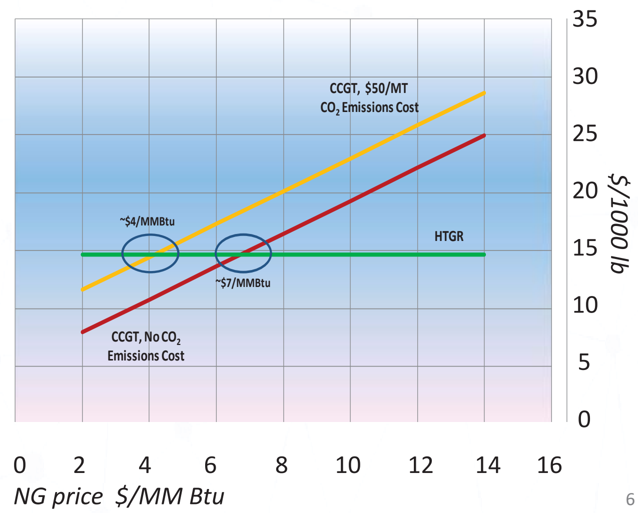
**High Temperature Gas-Cooled Reactor Projected Markets and Preliminary Economics, INL/EXT-10-19037 rev. 1, Aug. 2011.*



HTR and CCGT NG Electricity Production Price vs. Price of Natural Gas with and w/o carbon tax



HTR and CCNG Steam Production Price vs. Price of Natural Gas with and w/o carbon tax



Relatively Mature Technology

Proof of Performance

FORT ST. VRAIN – 842 MWt
(U.S.A.)
1976 - 1989

THTR – 750 MWt
(FRG)
1986 - 1989

…but the fuels, materials, and methods must be qualified for today's market and regulatory environment

Engineering Reactors

HTTR – 30 MWt
(Japan)
1999 - present

HTR-10 – 10 MWt
(China)
2000 - present

DRAGON – 20 MWt
(U.K.)
1964 – 1975

AVR – 46 MWt
(FRG)
1967 - 1988

PEACH BOTTOM 1 – 115 MWt
(U.S.A.)
1967 - 1974

Current Industrial Interest in TRISO-fueled power

Larger (200-625MWt) Plants for the Grid and Heat Users

Kairos Power

framatome

X energy

Microreactors (5-50 MWt) Units for Off-grid, Military Power

GENERAL ATOMICS

STARCORE NUCLEAR

BWXT
BWX Technologies, Inc.

HolosGen™

U BATTERY
Local Modular Energy

US ULTRA SAFE NUCLEAR

High Level Safety Design Objectives

- Meet regulatory dose limits at the Exclusion Area Boundary (EAB)
  - 25 rem Total Effective Dose Equivalent (TEDE) for duration of the release from 10 CFR 50.34 (10 CFR 52.79) at EAB for design basis accidents
  - EAB is typically estimated to be approximately 400 meters from the plant for a modular HTGR; supports co-location with industrial facilities
- Meet safety goals for cumulative individual risk for normal and off-normal operation
- Meet the EPA Protective Action Guides (PAGs) at the EAB as a design goal
  - 1 rem TEDE for sheltering
  - Design basis and beyond design basis events are considered
  - Realistically evaluated at the EAB
  - Emergency planning and protection

High Level Safety Design Approach

- Design using materials with properties that retain integrity at high temperature and are chemically stable
  - Helium coolant – neutronically transparent, chemically inert, low heat capacity, single phase
  - Ceramic coated fuel – high temperature capability, high radionuclide retention
  - Graphite moderator – high temperature stability, large heat capacity, long thermal response times
- Design the reactor with inherent and passive safety features
  - Retain radionuclides at their source within the fuel
  - Shape and size of the reactor allows for passive core heat removal from the reactor core through the uninsulated reactor vessel
    - Heat is still removed if the system is depressurized as a result of a breach in the reactor helium pressure boundary
    - Heat is radiated from the reactor vessel to the reactor cavity cooling system (RCCS) panels and rejected passively to the environment
  - Large negative temperature coefficient for intrinsic reactor shutdown
  - No reliance on AC-power to perform necessary safety functions
  - No reliance on operator action and insensitive to incorrect operator actions

Modular HTGR Safety Basis and Approach paper submitted for information to NRC, INL/EXT-11-22708, September 2011, ML14174A774

Comments to Address Issues from NRC Review

- Training slides are organized according to previously agreed-upon agenda topics and are consistent with previous training courses; therefore, not reorganized around specific learning objectives
- NRC ML numbers have been provided in the Suggested Reading lists where they apply

Idaho National Laboratory

High Temperature Gas-cooled Reactor: History

Hans Gougar

High Temperature Gas-cooled Reactor: History

Advanced Reactor Technologies

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Nuclear Engineer

NRC HTGR Training July 16-17, 2019

ART ADVANCED REACTOR TECHNOLOGIES

Overview

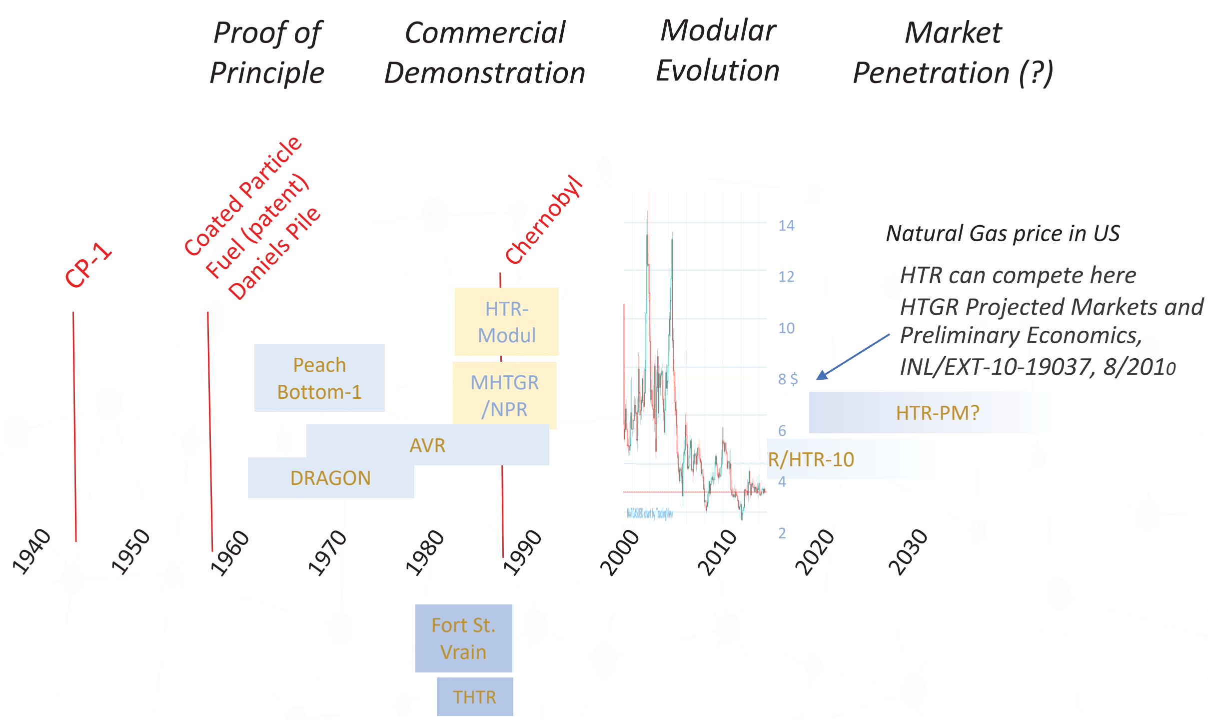
- Early and related concepts
- First Generation US and German plants
- Modular High Temperature Gas-cooled Reactors (mHTGR*)

** In these presentations, **M**HTGR refers to a specific design developed by General Atomics*

Visitor Entrance to THTR300 (European Institute for Climate and Energy website)

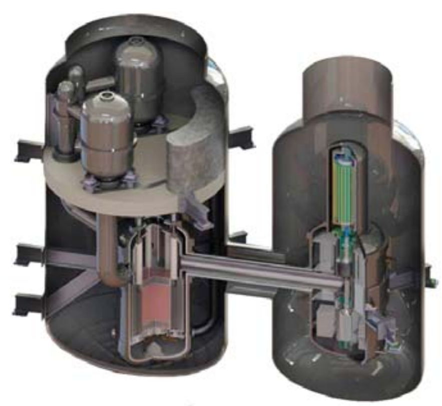
The Training Course delivered to the NRC in 2010 was spread over a few more days and was prepared and delivered by experienced vendors (see Suggested Reading List). You are encouraged to review that course material for specific design details and the view from a vendor perspective.

Timeline of HTGR Development

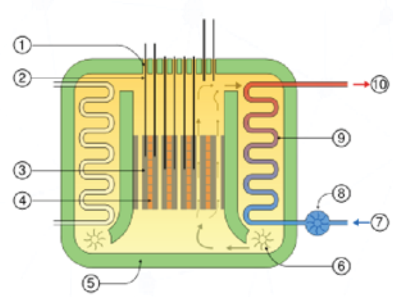


Related Concepts

- British Advanced Gas-cooled Reactor (AGR)
  - CO_2-cooled, 600°C outlet
  - UO_2 rods in SSTL clad
- Very High Temperature Reactor (VHTR)
  - Really hot HTGR (>850-1000°C)
- Advanced High-Temperature Reactor (AHTR) or PB-FHR (Kairos)
  - Molten salt instead of He
- Gas-cooled Fast Reactor (GFR)
  - Fast spectrum (no graphite)
  - UC fuel



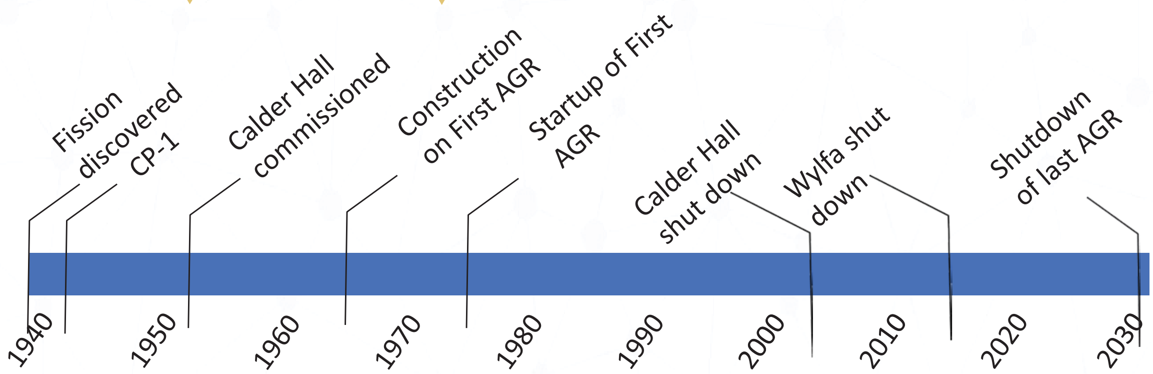
General Atomics EM2 GFR concept



Torness AGR (Scotland)

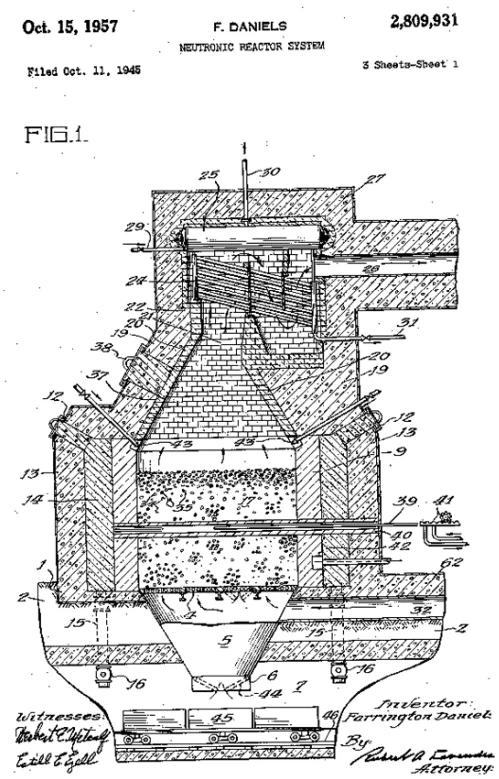
Prologue – Graphite-moderated, Gas-cooled Reactors (US/UK/France)

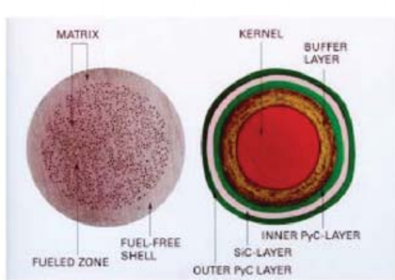
- CP-1 (air-cooled)
- Production/Power Reactors
- CO_2 cooled
  - MAGNOX (UK), UNGG(Fr)
  - AGR (UO_2 pellets in SS, <650°C CO_2, concrete RPV, reasonable performance after a rocky start)



HTR Conceived

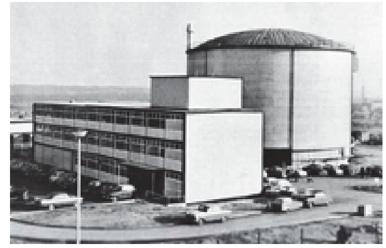
- Daniels Power Pile (1945)
  - F. Daniels (ORNL)
  - Graphite or BeO moderated
  - He cooled, 1350°F/732°C outlet
  - IHX and closed cycle Brayton
  - UC_2 or UO_2 in cladding
- Actual Experimental reactors followed
  - GCRE, ML-1, EGCR
- Final Puzzle Piece…Coated Fuel Particle
  - UKAEA, Battelle idea (~1957)
  - Superior retention of fission products at elevated temperatures (esp. in the TRISO version)

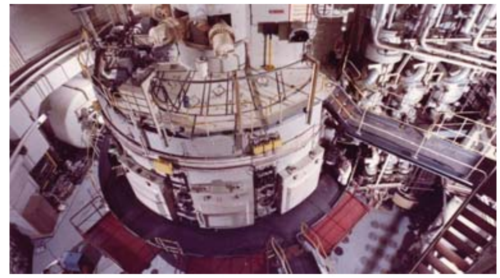


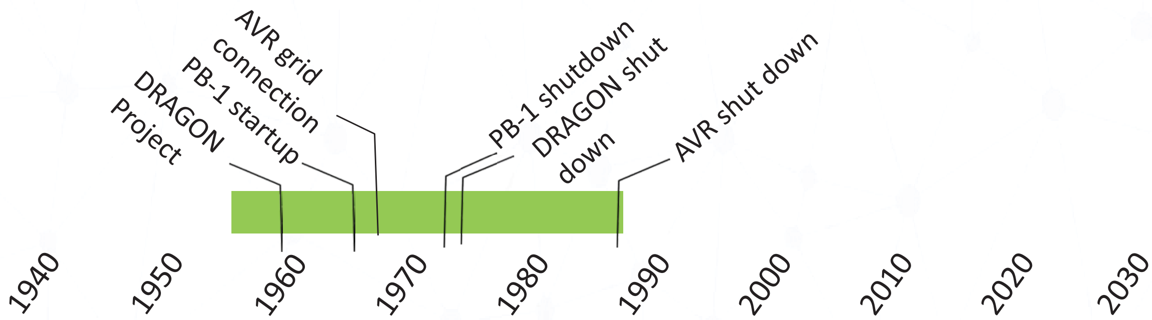


Phase 1 – Proof of Concept – DRAGON

- Built in the UK under a OECD/Euratom sponsorship
- Particle fuel and material testing
- Engineering challenges encountered and resolved
  - Control rod bowing
  - Replacement of inner reflector blocks
  - IHX and pipe corrosion

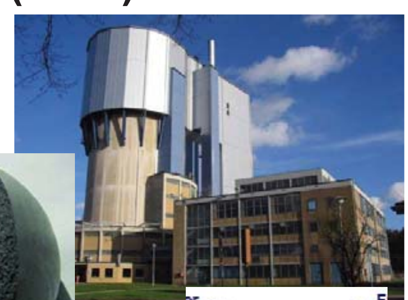


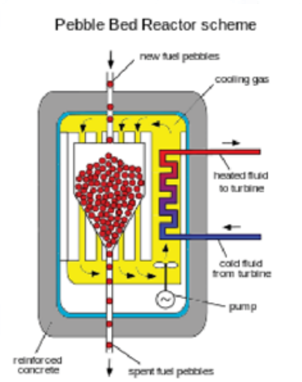


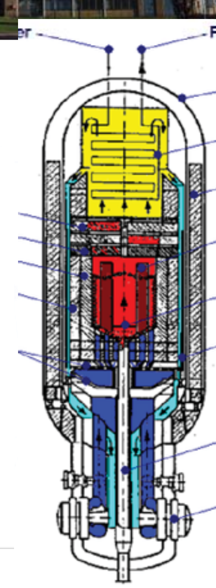


Arbeitsgemeinschaft VersuchsReaktor (AVR) (Germany)

- Pebble Bed reactor conceived by R. Schulten
- Arbeitsgemeinschaft VersuchsReaktor – 40 MWt/15 MWe prototype PBR for testing systems and fuels (BISO/TRISO)
- He-cooled up to 950°C at the outlet
- **Only One** (1!) operator needed for reactor/primary circuit operation
- Shutdown achieved by stopping forced circulation (rods inserted after cooldown)
- Growing pains
  - Leaky shield led to steam generator (SG) contamination
  - 1978 SG leak dumped 27 m^3 of water into the core while shut down (dried out and restarted)
  - Unpredicted high core temperatures





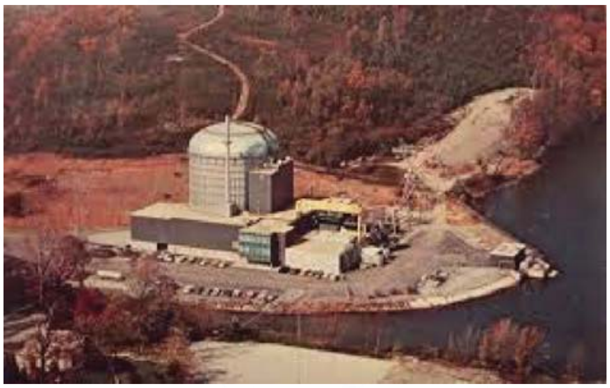


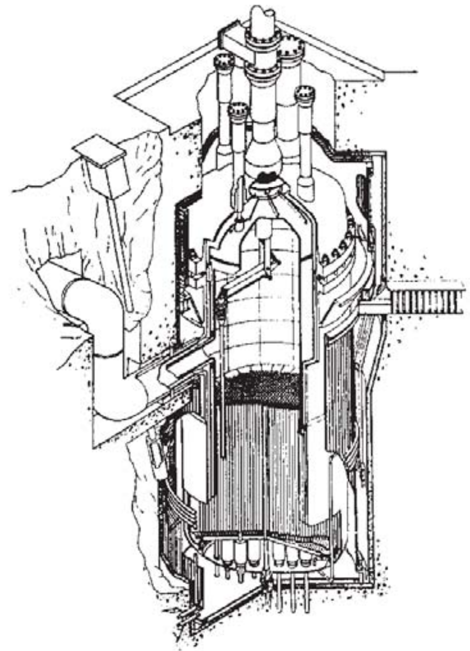
Despite some bad publicity (Moorman, 2008), AVR is considered an HTR success story (Kuppers, 2014).

By Cschirp at the German language Wikipedia, CC BY-SA 3.0, https://commons.wikimedia.org/ w/index.php?curid=11451341

Peach Bottom 1

- 115 MWt/40 MWe designed by General Atomics with support from the AEC and 57 utilities
- Prismatic – BISO coated fuel particles (cfp) in compacts/blocks
- 85% availability, load following, low operator doses
- Growing pains – Some cracking of blocks in the first core





Kingrey, K., "Fuel Summary for Peach Bottom Unit 1 High Temperature Gas-Cooled Reactor Cores 1 and 2", INEEL/EXT-03-00103, April 2003.

Phase 2 – Commercial Demo – Fort St. Vrain

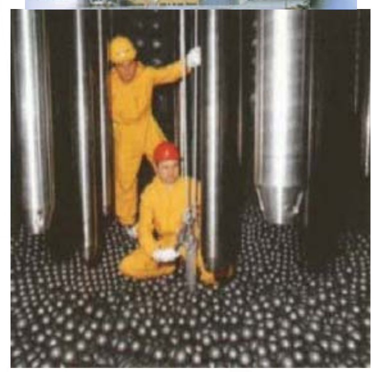
- 842 MWt/330MWe General Atomics design built with support form 57 utilities
- HEU/Th coated fuel particles in compacts/blocks
- Pre-stressed concrete Pressure Vessel (PV)
- Very low worker doses
- Growing pains resulted in low availability
  - Core flexing→ coolant oscillations (restraints recommended)
  - Leaky water-lubed gas circulators led to large ingress event
  - Core thermal fluctuations (Xe)
  - Reserve shutdown malfunction, hot He bypass on CR drives

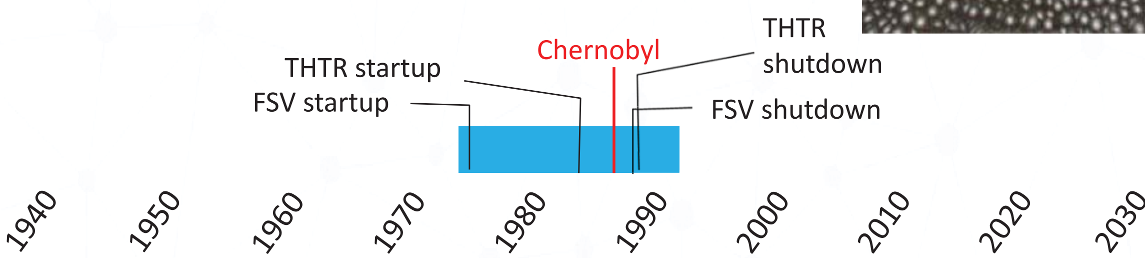
Despite these engineering issues, modern HTGR technology was demonstrated.

Shenoy, History and Evolution - Module 2A -HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Thorium High-temperature Nuclear Reactor

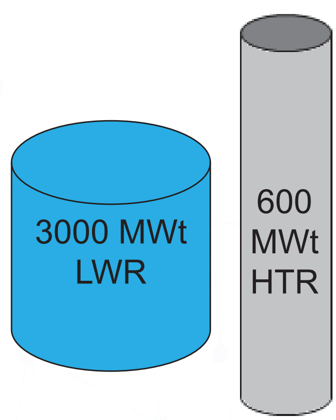
- 750 MWt/300 Mwe German Design
- HEU/Th cfp in pebbles
- Prestressed concrete PV
- Dry cooling
- Growing pains
  - Broken pebbles (shutdown rod insertion)
  - He upflow hindered pebble discharge
  - Bolt heads detached from hot duct assembly

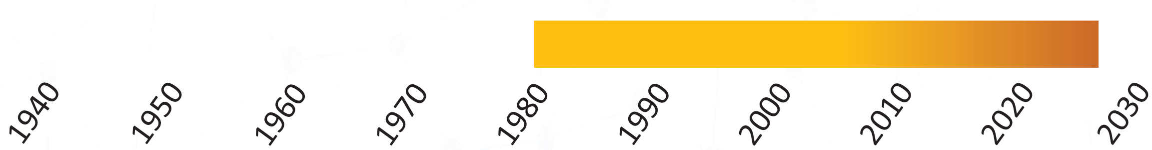




Phase 3 – Small and Modular (mHTGR)

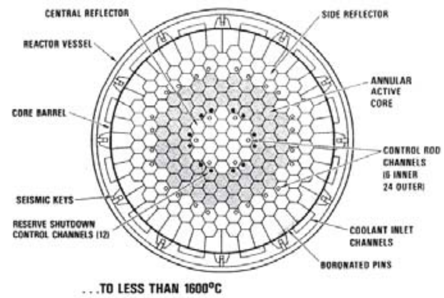
- Larger HTRs were envisioned after FSV and THTR
  - Low power density meant that the vessel would be huge
  - Active decay heat removal required
- Modular
  - Multiple modules with staggered deployment
  - Passive heat removal (high aspect ratio)

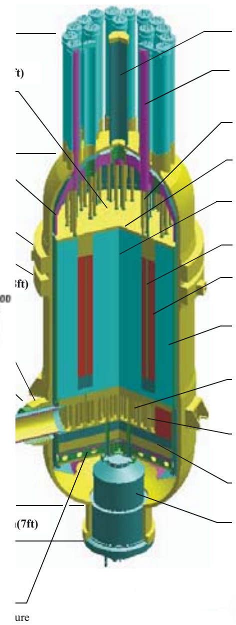


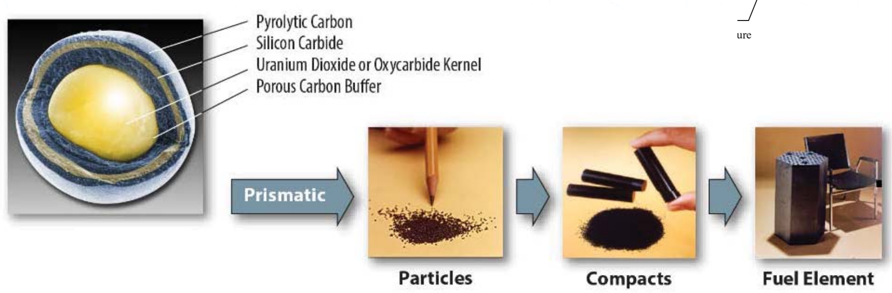


Modular High Temperature Gas-Cooled Reactor (MHTGR)

- General Atomics (GA) design, coalition of industrial interests
- 350 MWt prismatic (annular core) in a steel RPV
- Draft Pre-Application SER issued by NRC in 1989, revised and re-issued in 1995
- The basis for subsequent modular prismatic reactor designs such as the New Production Reactor, GT-MHR, Deep Burn MHR, AREVA SC-HTGR



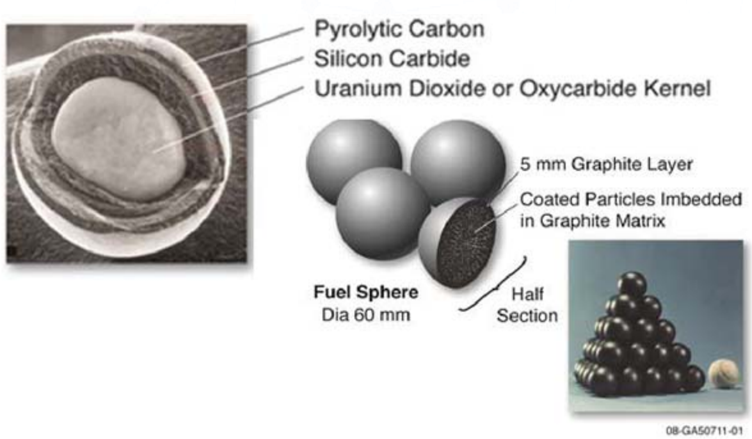


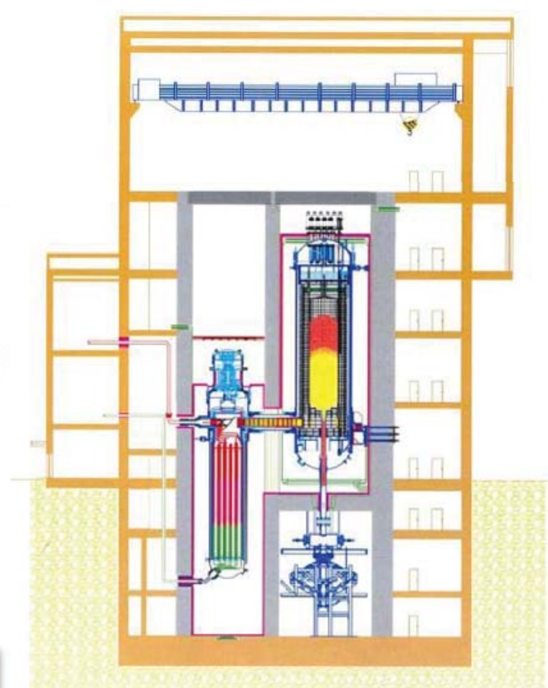


TRISO-coated fuel particles (left) are formed into fuel compacts (center) and inserted into graphite fuel elements (right) for the prismatic reactor

HTR Modul

- KWU/Siemens-Interatom
- 200 MWt pebble bed with online recirculating fuel (high burnup)
- Design submitted to German Licensing Authority in the late 1980's
- The basis for subsequent modular PBR designs like the PBMR and HTR-PM



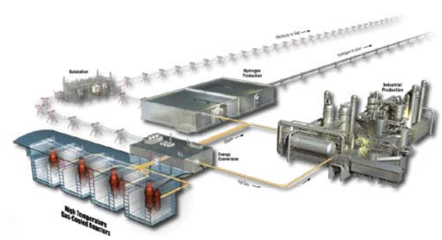


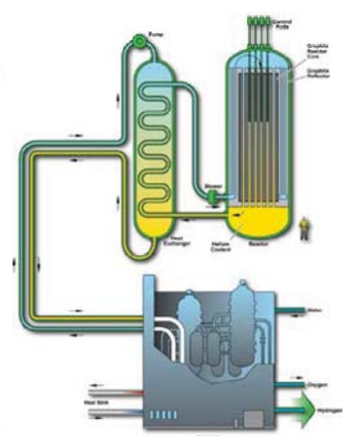
Lessons Learned

- HTR Potential was recognized very early
  - Accident tolerant fuel (TRISO)
  - Process heat applications
  - Modularity
- Problems (engineering) were typical of FOAK efforts – not generally inherent to the technology
  - Poor fuel performance in NPR, MHTGR
- Sensitive to the market, and politics
- NRC draft SER
  - Event selection was ok; TRISO fuel was problematic
- NGNP
  - (AGR) TRISO Fuel is ok; event selection needs work (current License Modernization Project is addressing this issue)

Phase 4 – Energy Security and Flexibility (CO_2-free)

- Government-sponsored R&D
  - US (NGNP/ART) – EPACT 2005, fuel and material qualification, etc.
  - Japan (JAEA) – technology development since the 1980s, HTTR, gas turbine and H_2 technology
  - China (INET) – keeps it simple (200 MWt PBR), 2-unit demo under construction and a '6-pack' looking for a site
  - Generation IV International Forum VHTR
- Industrial Interest
  - NGNP (GA, AREVA/Framatome, Westinghouse/PBMR)
  - X-energy, BWXT(fuel),
  - vSMR – StarCore, U-Battery, UltraSafe Nuclear, HolosGen, BWXT, X-Energy





?

1940 1950 1960 1970 1980 1990 2000 2010 2020 2030

International Efforts

South Africa

- In ~1998 the PBMR company tried to pick up where HTR Modul left off. Ran out of Government support in 2010. Almost $1B spent
- Some very nice test facilities constructed

He Test Facility Pelindaba

Japan

- Steady prismatic HTR technology development since the 1980’s
- Nice 30 MWt engineering-scale reactor (to be connected to a gas turbine and H_2 plant)
- 50, 300, and 600 MWt commercial designs
- Working on gas turbine and H_2 technology

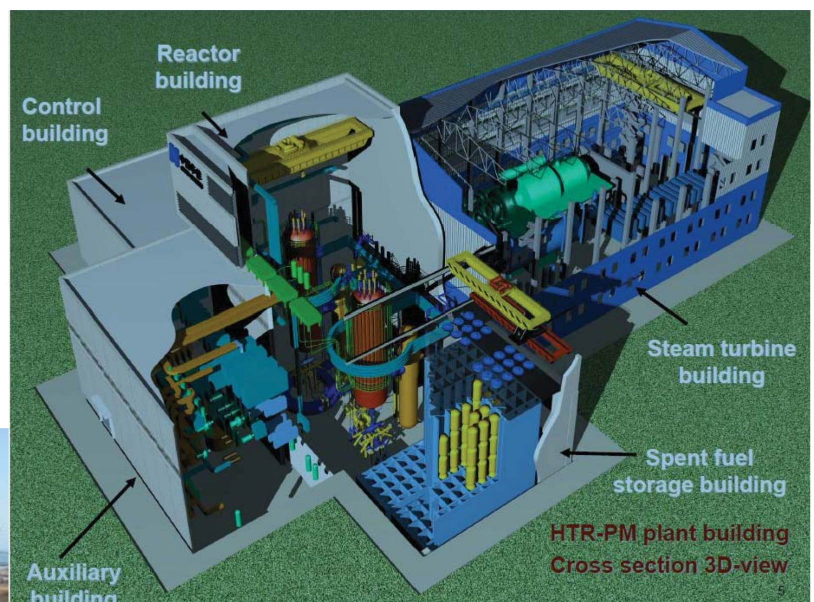
HTTR (Japan: 1999-)

China

- 10 MWt engineering scale reactor
- 2 unit HTR-PM DPP to go critical in 2019
- Impressive engineering test facilities

HTR-10 (China: 2000-)

HTR-PM under Construction in Weihai, China



Overview of HTGR Projects in China, Technical Meeting on Knowledge Preservation for Gas Cooled Reactor Technology and Experimental Facilities, Vienna, Austria, 2018.

Suggested Reading List

- 2010 HTGR Technology Course for the Nuclear Regulatory Commission.
- Bechtel National, Inc., et al. 1986. Preliminary safety information document for the standard MHTGR. HTGR-86-024. Stone and Webster Engineering Corporation.
- Brey, H.L., 2003. The Evolution and Future Development of the High Temperature Gas Cooled Reactor. Proceedings of GENES4/ANP2003, Sep. 15-19, 2003, Kyoto, Japan.
- Daniels, F. 1957. Neutronic Reactor System. United States Patent 2809931.
- Moore, R. A. et al., 1982. HTGR Experience, Programs, and Future Applications. Nucl. Eng. Des. 72, 153.
- IAEA, 1996. High Temperature Gas Cooled Reactor Technology Development. IAEA - TECDOC-988.
- IAEA 2001. Current Status and Future Development of Modular High Temperature Gas Cooled Reactor Technology. IAEA-TECDOC-1198.
- Kadak, A. C., 2016. The Status of the US High-Temperature Gas Reactors. Engineering, Vol. 2 (2016), pp. 119-123.
- Kugeler, K. et al. 2017. The High Temperature Gas-cooled Reactor - Safety considerations of the (V)HTR-Modul. EUR 28712 EN, Joint Research Center.

Suggested Reading List (cont)

- Küppers, C., et al. 2014. The AVR Experimental Reactor – Development, Operation, and Incidents Final Report of the AVR Expert Group. Forzungzentrum Juelich, Germany.
- Massimo,L. "The Physics of High Temperature Reactors", ebook ISBN 9781483280288.
- Melese and Katz, "Thermal and Flow Design of Helium-Cooled Reactors", American Nuclear Society, ISBN 0-89448-027-8, 1984.
- Moorman, R. 2008. A safety re-evaluation of the AVR pebble bed reactor operation and its consequences for future HTR concepts. Jul-4275 (ISSN 0944-2952), Julich Forschungzentrum.
- Moorman, R. 2008. Fission Product Transport and Source Terms in HTRs: Experience from AVR Pebble Bed Reactor. Science and Technology of Nuclear Installations, Volume 2008, Article ID 597491.
- Shenoy, A. (General Atomics) History and Evolution of HTGRs, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.
- Venter, P. (PBMR) Module 6b – Pebble Bed HTGR Nuclear Design, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.
- Vollman, R. (General Atomics) Prismatic HTGR Core Design Description, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.
- Windes, W. et al, "Discussion of Nuclear-Grade Graphite Oxidation in Modular High Temperature Gas-Cooled Reactors, M3AT-17IN160303, Idaho National Laboratory, 2017.
- Zhang, Z., et al. 2016. The Shandong Shidao Bay 200 MWe High-Temperature Gas-Cooled Reactor Pebble-Bed Module (HTR-PM) Demonstration Power Plant: An Engineering and Technological Innovation. Engineering 2 (2016), pp. 112–118.

INL

Idaho National Laboratory

High Temperature Gas-cooled Reactor: Core Design

Hans Gougar

High Temperature Gas-cooled Reactor: Core Design

Advanced Reactor Technologies

Idaho National Laboratory

Hans Gougar, PhD

Nuclear Engineer

Gerhard Strydom

National Technical Director – DOE Advanced Reactor Technologies Gas-Cooled Reactor Campaign

NRC HTGR Training July 16-17, 2019

ADVANCED REACTOR TECHNOLOGIES

HTGR Core Design – Overview

- General Attributes of Modular Prismatic and Pebble Bed HTGRs
  - Common features and physics
  - Neutronics
  - Prismatic and Pebble Fuel
  - Thermal-Fluidics
  - Inherent Safety
- Plant Systems and Power Conversion
  - Reactivity Control
  - Instrumentation and Control
  - Helium Conditioning
  - Power Conversion
- Normal Operation and Power Maneuvers

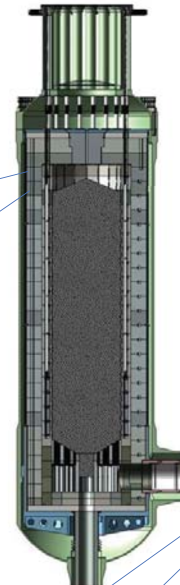
Fort St. Vrain Fuel Blocks (General Atomics)

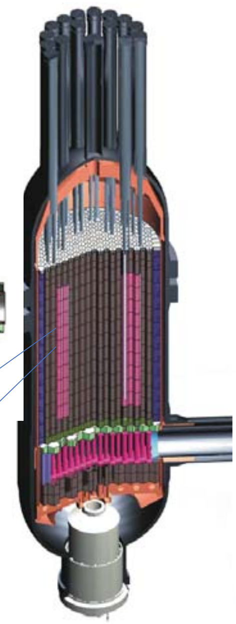
Gun drilling long holes in Ft. St. Vrain fuel elements
Today - drilled with numerically controlled machines

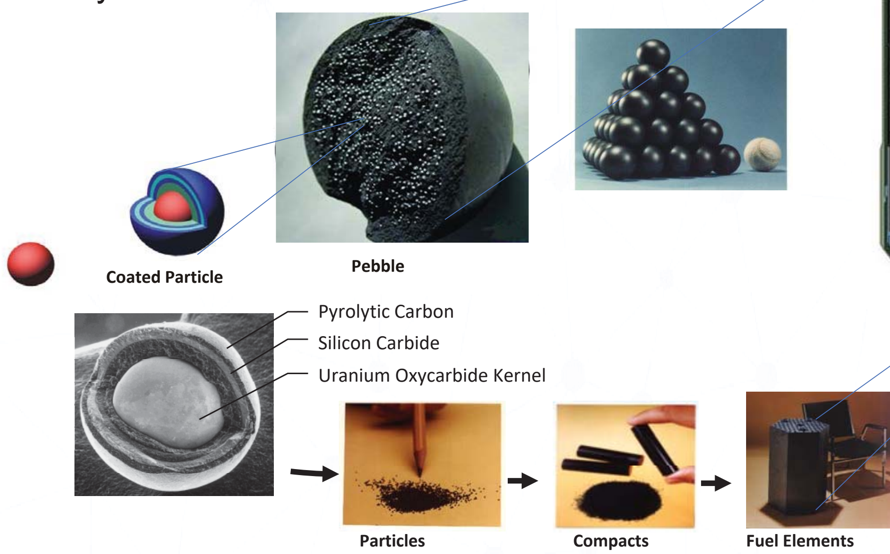
Vollman, R. Prismatic HTGR Core Design Description, Module 5A -HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Modular HTGR(s) from the bottom up

- UO_2 or UCO (ceramic) kernels
- Tristructural isotropic (actually 4 layers around the kernel)
- Pressed into a semi-graphitic matrix and shaped into either ‘compacts’ or pebbles
- Cylindrical or annular cores

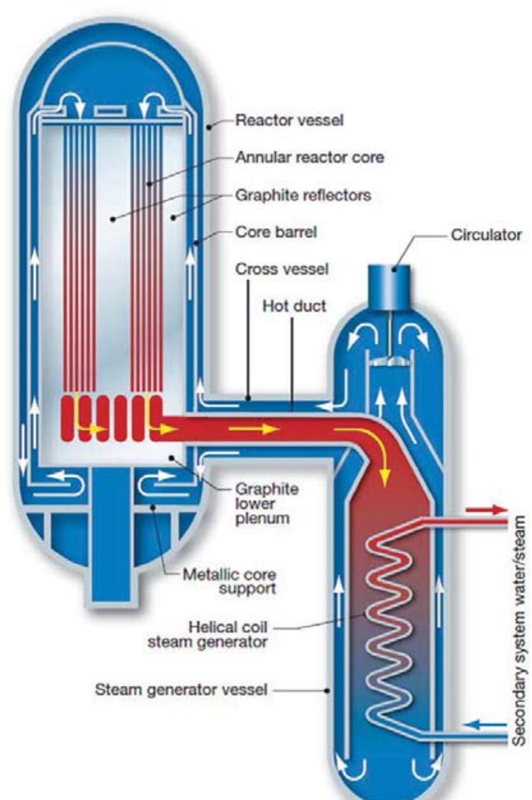






Relevant Attributes of Modular HTGRs

- Graphite-moderated and reflected
- Cooled (usually) by helium (~7 MPa)- Molten salt is being explored (and nitrogen has been proposed)
- Large ΔT_c (>400°C) across the core (top to bottom) compared to 30°C for an Light Water Reactor
- Fuel: TRISO fuel particles in a carbonaceous matrix
- Uninsulated reactor vessel
- Large aspect ratio: heat escapes radially via conduction and radiation if forced cooling is lost. This attribute also limits the power density (~400 MWt for PBRs; ~600MWt for prismatic reactors)
- Slow temperature response during accidents (high heat capacity and low power density)



(1 of 2 steam generators shown)

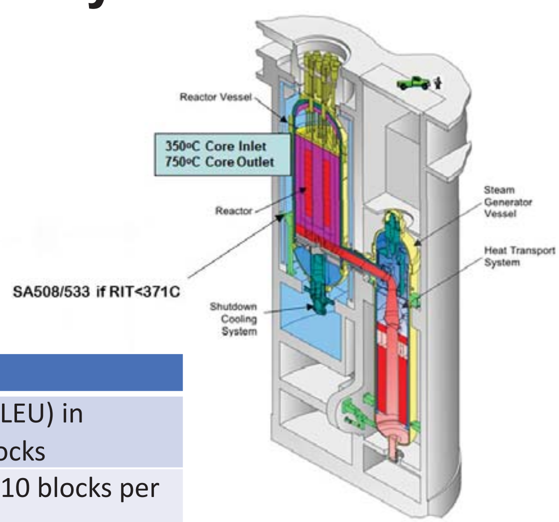
LWRs vs HTGRs in a Nutshell

Item	HTGR	LWR
Moderator	Graphite	Water
Coolant	Helium	Water
Average coolant exit temperature	700-950°C	310°C
Structural material	Graphite	Steel
Fuel clad	SiC and PyC	Zircaloy
Fuel	UO_2, UCO	UO_2
Fuel damage time at temperature	UCO - No failures for at least 150 hrs @ 1800°C*	1260°C
Power density, W/cm^3	4 to 6.5	58-105
Migration Length, cm	57	6

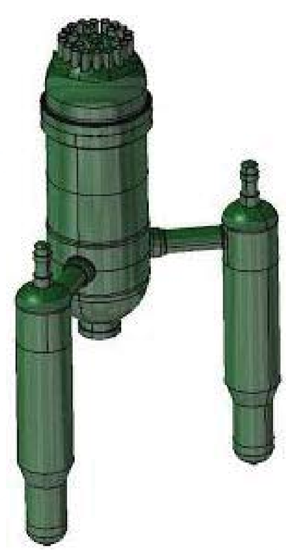
* Not a hard limit; based on statistical failure rates. Typical duration of peak fuel temperature is less than 100 hrs for a Loss of Forced Cooling event

Shenoy, A.. (General Atomics) History and Evolution of HTGRs, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Common Primary Loop Features – Framatome Steam Cycle-HTGR

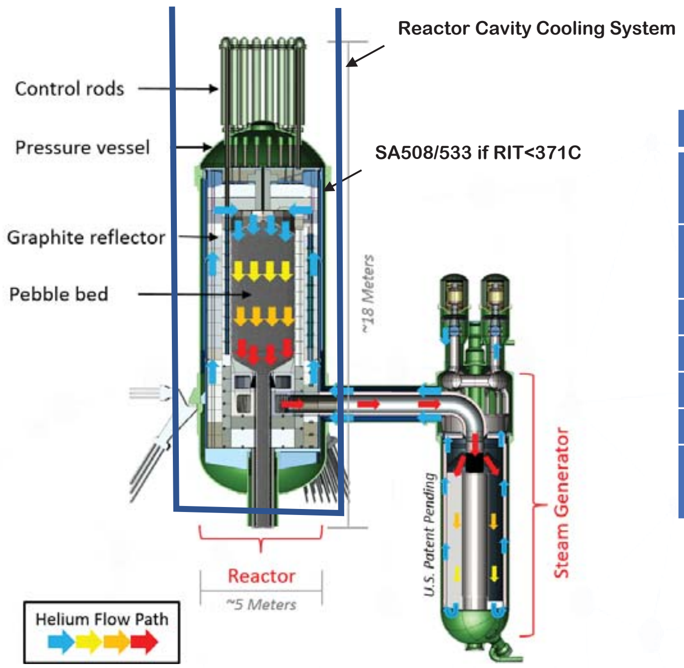


Parameter	
Fuel	TRISO (<20% LEU) in Compacts/Blocks
Core Geometry	102 columns,10 blocks per column
Reactor Power	625 MWt
Reactor Outlet Temperature	750°C
Reactor Inlet Temperature	325°C
Primary	He at 6 MPa
Secondary (x2)	Steam @ 16.7 MPa, 566°C



Framatome 625 MWt Prismatic SC-HTGR *(framatome.com) – Heat Transport System (HTS) supports process heat applications*

Common Primary Loop Features



Parameter	
Fuel	TRISO (~15% LEU) in Pebbles
Core Geometry	218 Pebbles in a Cylindrical Bed
Reactor Power	200 MWt
Reactor Outlet Temperature	750^{o}C
Reactor Inlet Temperature	260^{o}C
Primary	He at 6 Mpa
Secondary (×2)	Steam at 16.5 MPa, 565^{o}C

Xe-100 200 MWt Pebble Bed HTGR
(x-energy.com)

Reactor (Vessel) Cavity Cooling System

- Active or passive heat removal via absorption of thermal radiation (shine) emitted from a hot uninsulated reactor pressure vessel
- Ultimately rejects heat to the atmosphere
- Air-cooled, water-cooled, or hybrid configurations

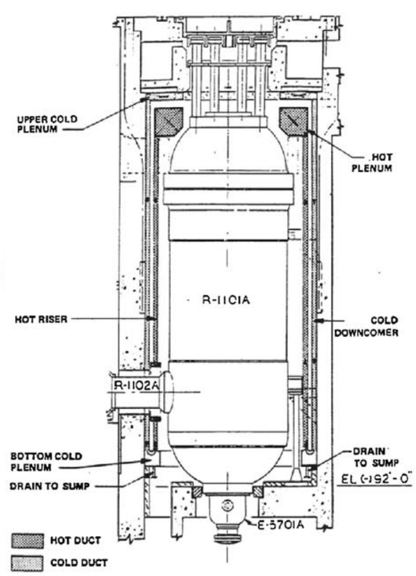
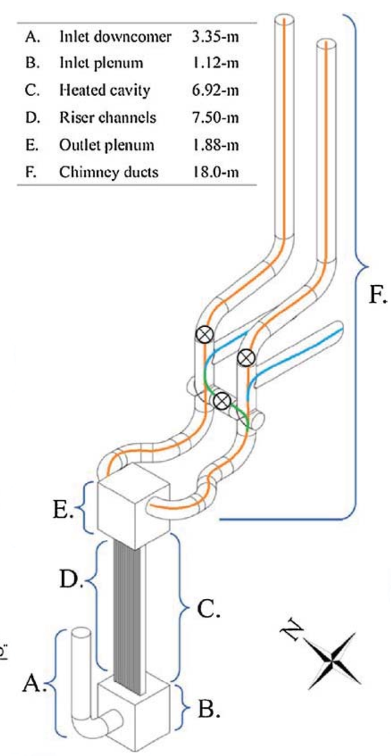


FIGURE 5.5-6
RCCS COOLING PANEL CONFIGURATION – ELEVATION
HIGH TEMPERATURE GAS-COOLED REACTOR PRELIMINARY SAFETY INFORMATION DOCUMENT HTGR-86-024



Lisowski, D.D. et al, Experimental Observations of Natural Circulation Flow in the NSTF, Nuclear Engineering and Design 306, (2016) 124-132.

Physics of HTGRs

Graphite dominates

- **Neutronics**
  - Core looks very homogeneous and diffusive, longer mean free path
  - Slightly harder spectrum than LWRs (more negative temperature feedback)
  - Good Pu-burner but MA buildup is high
- **Thermal-fluidics and Accident Behavior**
  - Graphite acts as a thermal buffer – absorbs heat during reactivity insertions and conducts (or radiates) it away
  - Time constant is much longer than neutronics
- **Mechanical**
  - Holds the core together and ‘creeps’ to relieve stress
- **Fission Product Retention in fuel element (block)**
  - Holds much of what little FP escape from the TRISO fuel
- **Spent fuel**
  - Large volume, low heat production, geochemically stable

Core composition HTR-PM

	v/o	m/o
Carbon	60.6	96.0
Helium	39.0	0.2
UO_2	0.4	3.8

Massimo,L. “The Physics of High Temperature Reactors”, ebook ISBN 9781483280288

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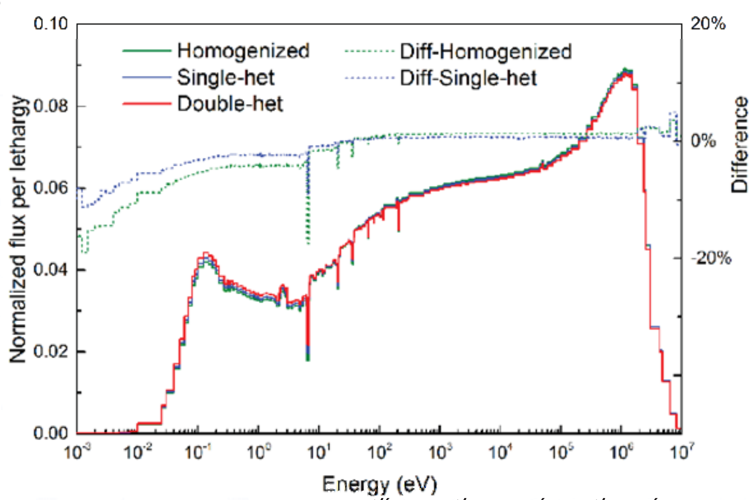


Fu & Hou, Jason & Ivanov, Kostadin. (2016). Effect Of Double Heterogeneity Treatment On Neutronics Modeling of HTGR Unit Cell.

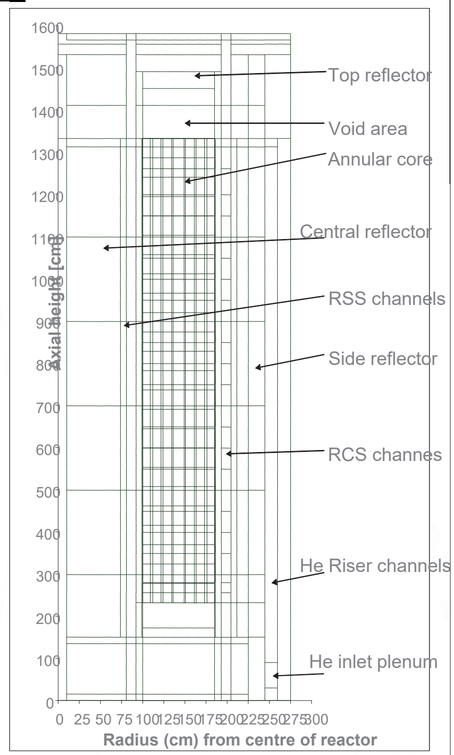
Core Neutronics

0.0253eV capture cross section of C-12

JENDL-4.0	JENDL-3.3	ENDF/B-VII.0	JEFF-3.1
3.85 mb	3.53 mb	3.36 mb	3.36 mb

Goto, et al, Impact of revised thermal neutron capture cross section of carbon stored in JENDL-4.0 on HTTR criticality cross section, J of NS&T, Jan. 2012.

- So much graphite…
  - Criticality benchmark evaluations (Bess, 2014) frequently overpredicted k_{eff} by several hundred pcm until new measurements dropped σ_c by ~0.3 mb. (under-prediction resulted)
  - Relatively large uncertainties in neutronic calculations (e.g., XS input uncertainties lead to ~600 pcm keff uncertainty (1 std.dev)
  - Fortunately, safety parameters (e.g. rod worth, power peaking) are largely insensitive (e.g. <1.5% variation in local block power) to these XS uncertainties (Strydom, 2018)
- Large temperature and burnup variation along z
  - Need to discretize the core along z
  - Must couple (at least loosely) to thermal-fluidics
- Large mean free path (mfp)
  - Neutronic coupling between blocks or pebble bed 'zones' – single assembly lattice calculations do not capture the leakage effects



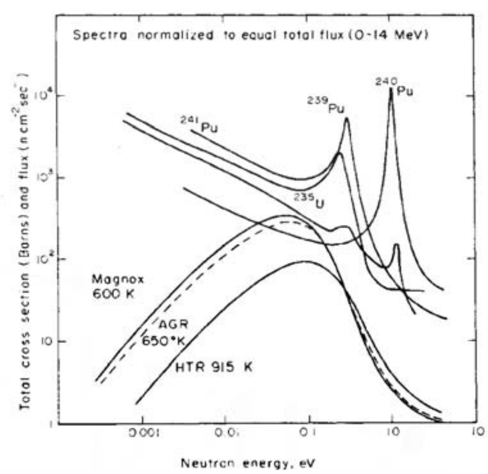
Typical R-Z discretization of the PBMR-400 Core

John D. Bess, Leland M. Montierth, Oliver Köberl and Luka Snoj (2014) Benchmark Evaluation of HTR-PROTEUS Pebble Bed Experimental Program, Nuclear Science and Engineering, 178:3, 387-400,DOI: 10.13182/NSE14-13

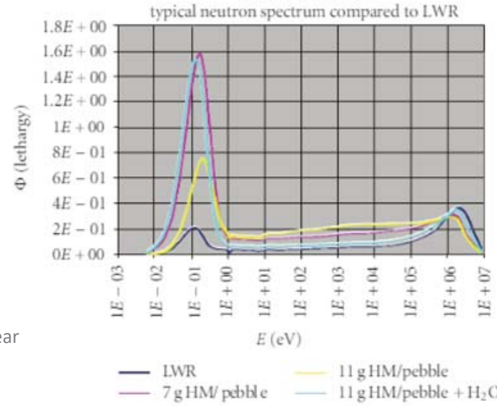
G. Strydom, P. Rouxelin (2018). IAEA CRP on HTGR UAM: Propagation of Phase I cross section uncertainties to Phase II neutronics steady state using SCALE/SAMPLER and PHISICS/RELAP5-3D. Proc. of HTR2018, Warsaw, Poland.

Graphite vs. H_20 as Moderators

	H2O	Graphite
Average Thermal Energy (eV)	0.17	0.22
Enrichment %	3-5	8-16
Moderating Ratio ($\xi\Sigma_s/\Sigma_a$)	62	216
# scatters to thermal	~18	~114
Mean free path (cm)	0.3	3.9
Migration Length (cm)	57	6



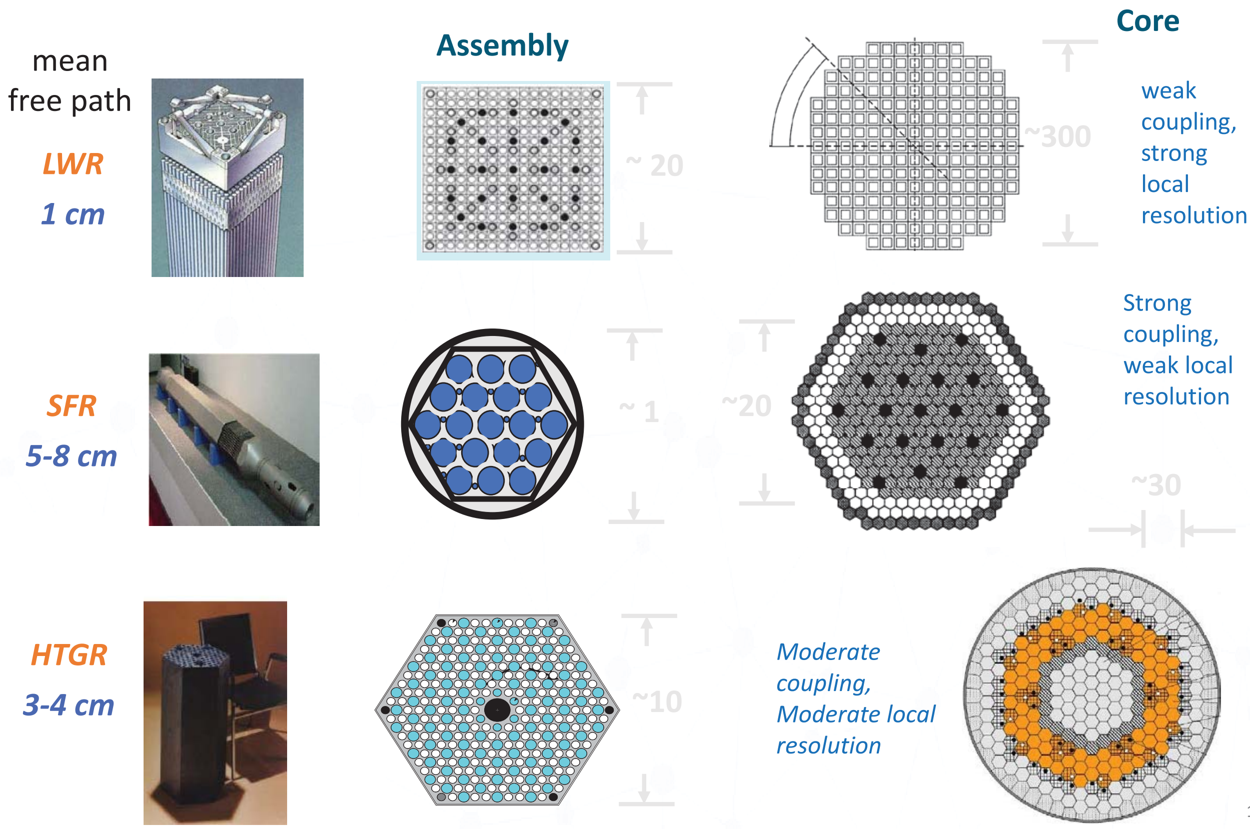
HTR-pebble bed typical neutron spectrum compared to LWR



- Greater buildup of minor actinides
- Stronger negative fuel temperature feedback
  - HTGR: -7 pcm/K
  - PWR: -1 to -4 pcm/K

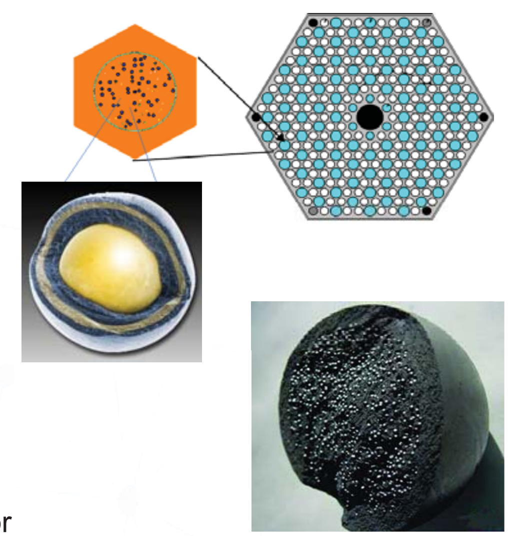
Bomboni, Eleonora and Cerullo, Nicola and Lomonaco, Guglielmo and Romanello, Vincenzo. (2008). A Critical Review of the Recent Improvements in Minimizing Nuclear Waste by Innovative Gas-Cooled Reactors. Science and Technology of Nuclear Installations. 10.1155/2008/265430.

Relative Size (mfp) of Fuel and Core



Cross-Generation Considerations

- 3 or 4 levels of heterogeneity
- More scattering in the resonance region
- Long migration area
- Reflectors (and control rods in them)
- Uncertainties in nuclear data
- Good agreement can be obtained by using:
  - More groups (8-26)
  - A supercell method for capturing leakage and generating cross sections for the control rod regions in the reflector
  - 'SuperHomogenization' or discontinuity factors for harmonizing transport and diffusion reactor rates
  - Discretize in the axial dimension



H. Gougar, A. Ougouag, W. Yoon, "Multiscale Analysis of Pebble Bed Reactors," Proceedings of 5th International Topical Meeting on High Temperature Reactor Technology, (HTR 2010), Prague, 2010.
Laboure, V., Ortensi, J., an Hummel, A., :HTTR 3-D Cross-Section Generation with Serpent and MAMMOTH, INL/EXT-18-51317, September 2018.

HTGR Core Design – Overview

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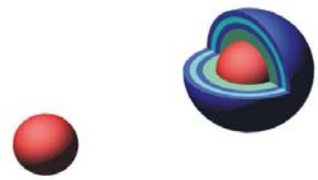
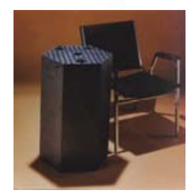
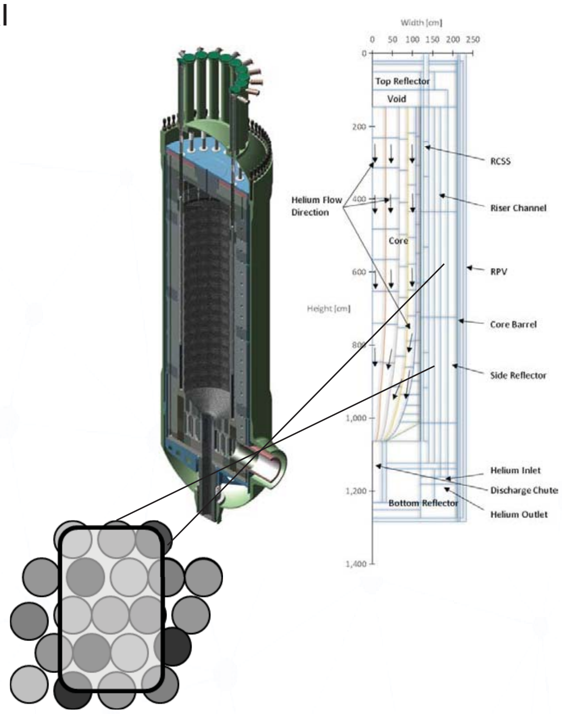
Fuel Elements in HTGRs





Pebble Bed Fuel Considerations

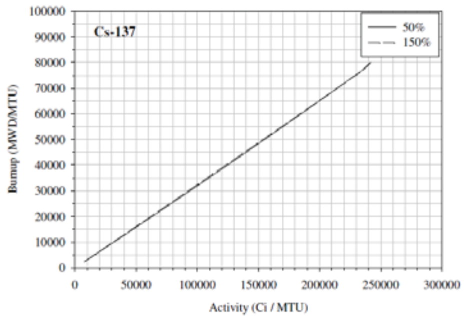
- Lack of 'natural' assemblies; cross sections are computed for somewhat arbitrarily chosen 'spectral zones' to account for variations in temperature and composition
- Fuel movement and reshuffling
  - Loaded from the top (unless it's cooled with molten salt)
  - Pebbles roughly follow axial flowlines; radial motion toward a discharge chute. Burnup is solved along these
  - Partially burnt pebbles sent back to the top (requires online burnup measurement)
  - If the power and fuel pebble design are kept constant, eventually the core reaches an equilibrium burnup profile
  - Online fueling allows for a very low excess reactivity
  - Analysis of the 'Running-in' Period (which can be a few years) poses a challenging design problem

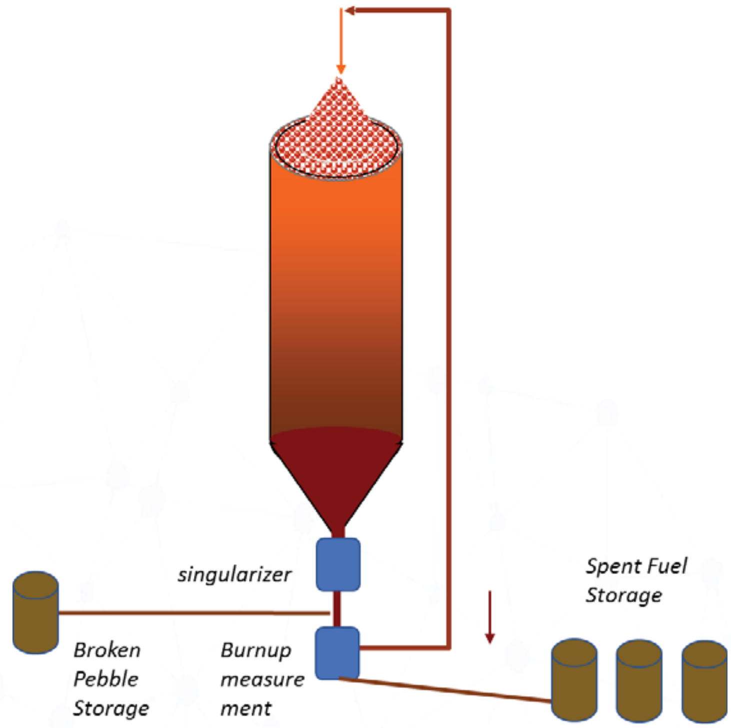


A.M. Ougouag, H.D. Gougar, R.S. Sen, "Identification of Spectral Zone Boundaries in Pebble Bed Reactors," Proceedings of 9th International Topical Meeting on High Temperature Reactor Technology, (HTR 2018), Warsaw, 2018.

Pebble Bed Reactor (PBR) Fuel Handling

- Pneumatic transfer
- Burnup Measurement
- Spent Fuel Storage

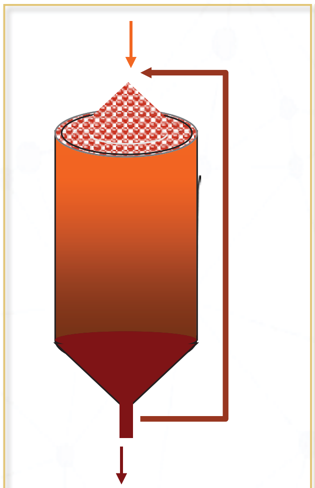




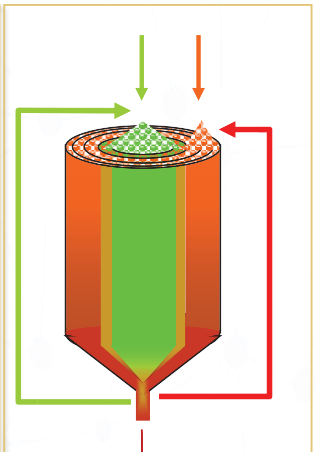
Su, Bingjing and Zhao, Zhongxiong and Chen, Jianwei and I. Hawari, Ayman. (2006). Assessment of on-line burnup monitoring of pebble bed reactor fuel by passive neutron counting. Progress in Nuclear Energy - PROG NUCL ENERGY. 48. 686-702. 10.1016/j.pnucene.2006.06.013.

PBR Fuel Zoning possibilities

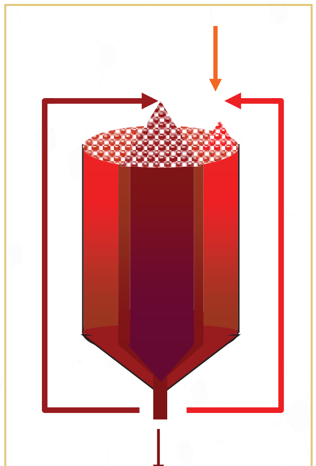
- Pebble flow is largely axial and incompressible
- Mixing between ‘streamlines’ is minimal, allowing (for most design and analysis purposes) the Bateman equation to be solved along the flow lines
- Flow is subjected to drag forces along reflector walls (variable residence time)
- Cylindrical or annular cores, multiple pebble types, and different loading patterns are possible (cylindrical vessels with a single pebble type are the most common)



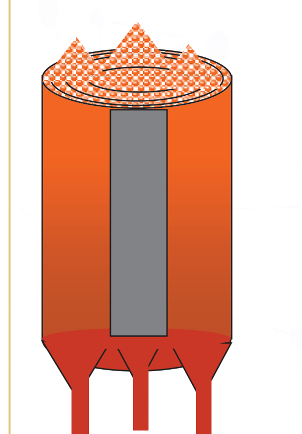
Single pebble type
Cylindrical core (Xe-100)



Fuel and graphite pebbles
Cylindrical core (early PBMR concept)



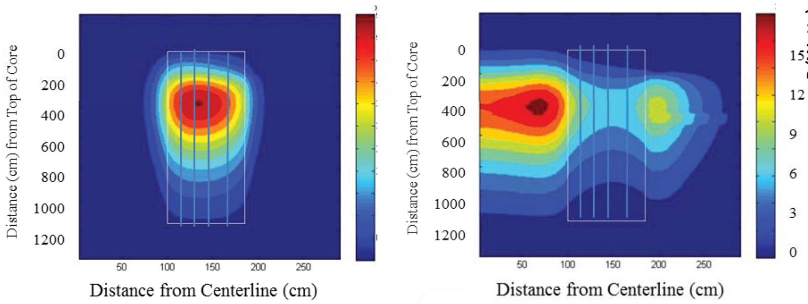
Single pebble type
with burnup zoning (THTR)



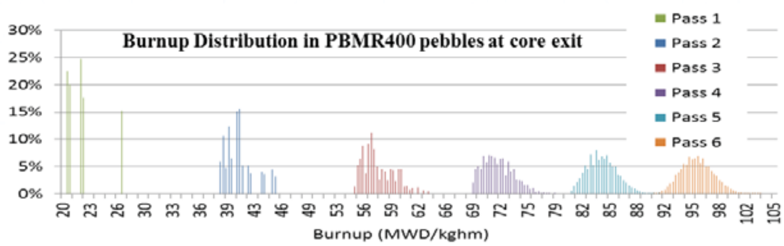
Single pebble type
Annular core (PBMR-400)

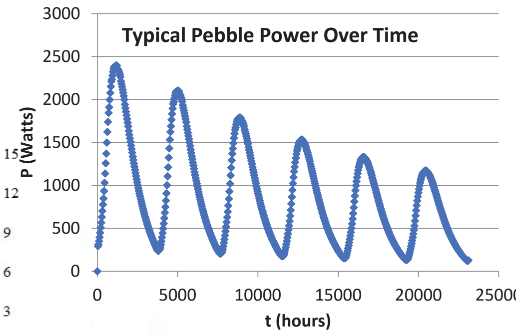
Resulting Profiles

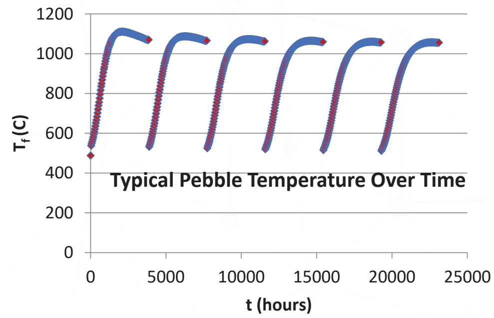
Fast and thermal flux profiles in the PBMR-400 equilibrium core (6-pass core with inner reflector)



Spectral variations leads to a burnup distribution in pebbles leaving the core



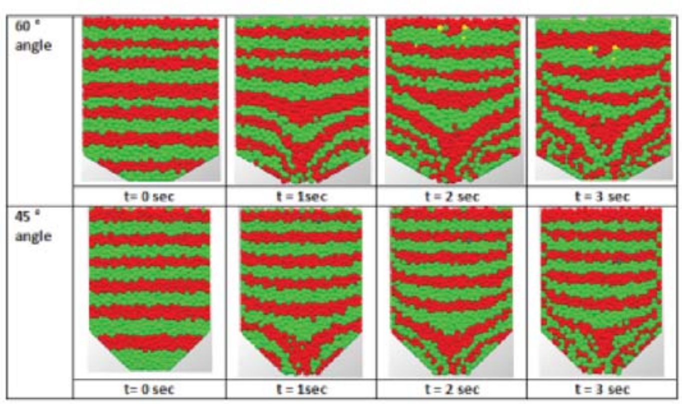




Gougar, H.D., Ougouag, A. M., Terry, W. K., and Ivanov, K. I., "Automated Design and Optimization of Pebble Bed Reactor Cores, Nuclear Science and Engineering, Oct. 2008.

PBR Fuel Flow Modeling

- Inter-pebble and pebble-wall friction and the geometry of the vessel lead to non-uniform radial flow patterns
- Flow lines were originally determined experimentally; now DEM codes are used (PEBBLES, LIGGGHTS- LAMMPS, PFC-3D)
- Earthquakes can be modeled



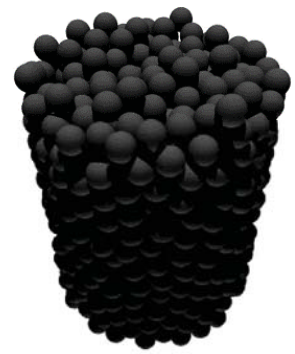
Cogliati, J., "PEBBLES: A Computer Code for Modeling Packing, Flow, and Recirculation of Pebbles in a Pebble Bed Reactor," Proceedings of 5th International Topical Meeting on High Temperature Reactor Technology, (HTR 2010), Prague, 2010.

C. H. Rycroft, G. S. Grest, J. W. Landry, and M. Z. Bazant, Analysis of Granular Flow in a Pebble-Bed Nuclear Reactor, Phys. Rev. E 74, 021306 (2006).
PFC3D – Itasca Consulting Group.

More on Pebble Motion

- Earthquakes cause pebble bed to settle
- A settling induced reactivity insertion and subsequent power transient ~~requires~~ some computational horsepower to simulate
- Fortunately this does not appear to be much of a safety issue – temperature feedback shuts down the reactor with a relatively mild heatup
- Block shifting may interfere with control rod motion
- The real hazards from earthquakes are the stress put on pipes and other components

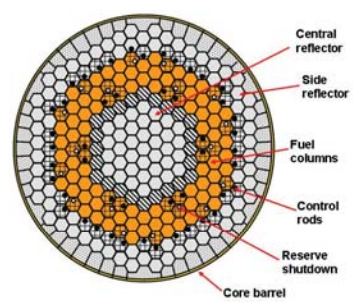
Solid Volume Fraction of randomly-packed spheres = ~0.59-0.64

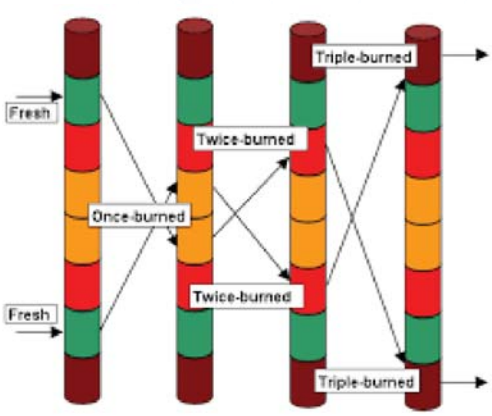


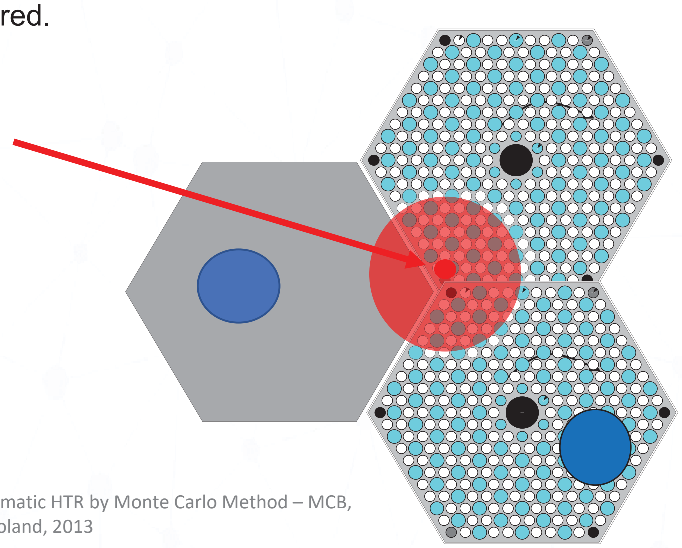
Ougouag and Cogliati. "Earthquakes and Pebble Bed Reactors: Time-dependent Densification". Joint International Topical Meeting on Mathematics and Computation and Supercomputing in Nuclear Applications (M&C + SNA 2007) Monterey, California, April 15-19, 2007

Prismatic Fuel Considerations

- Compacts in blocks with engineered coolant channels – more heterogeneous than PBRs – batch-loaded
- Burnable poison pins are used to flatten the power and hold down reactivity over the cycle
- Shutdown rods are inserted into the fuel blocks – normally out (holes become streaming pathways)
- Fuel reshuffling can be 3D, but generally not (uneven swelling of blocks?). Axial shuffling generally preferred.







Cetnar, J. et al, Assessment of Pu and MA utilisation in deep burn Prismatic HTR by Monte Carlo Method – MCB, Project PUMA, AGH-University of Science and Technology, Krakow, Poland, 2013

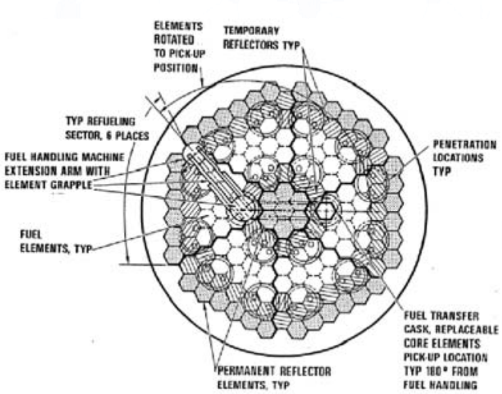
Tolerances in General Atomic's Neutronic Codes (C-E)/E)

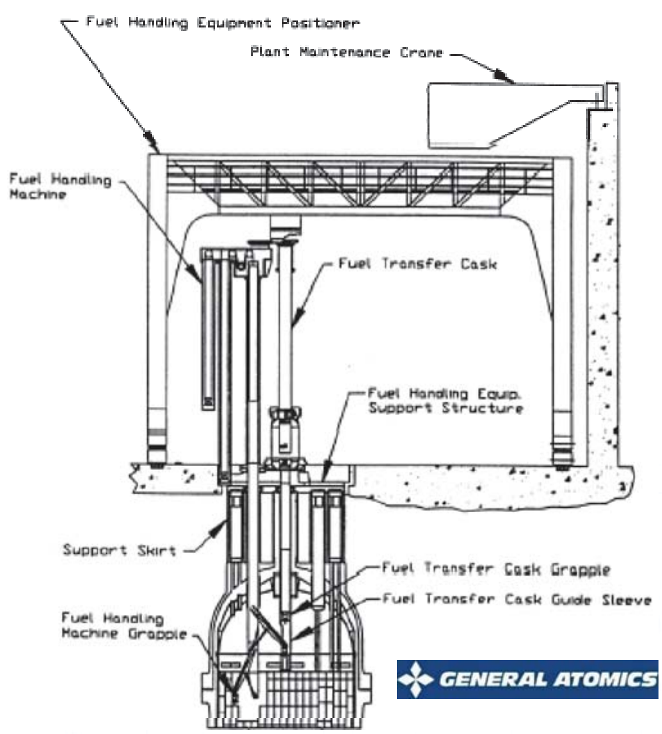
Facility	Temp. Defect	C. R. Worth	Power Distr.	K_{eff}	Water Ingress	Decay Heat
HEU-CORES						
Peach Bottom Critical	±14%	-11%	±10%	±0.7%	DA	-
Peach Bottom	-11% to +4%	-6% to +10%	±10%	±0.7%	-	DA
HTGR Critical	+6%	+4% to 13%	-	-0.1% to +1.0%	-	-
Fort St. Vrain	-9% to +12%	±10%	±15%	±0.5%	-	DA
HTLTR	±8%	-	-	-	-	-
KAHTER	-	DA	DA	-0.3% to +6%	±13%	-
DRAGON	DA	-11%	DA	-	-	DA
HEU/LEU CORES						
AVR	-25%	-5% to +15%	-	±11%	-	DA
LEU CORES						
HITREX-2	-	-	±10%	±0.5%	-	
HITREX-2	-	-	±10%	±0.5%	-	

Baxter, A.. (General Atomics) Module 5b - Prismatic Nuclear Design, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Prismatic Fuel Handling – MHTGR

Fort St. Vrain Fuel Handling Machine (FHM)



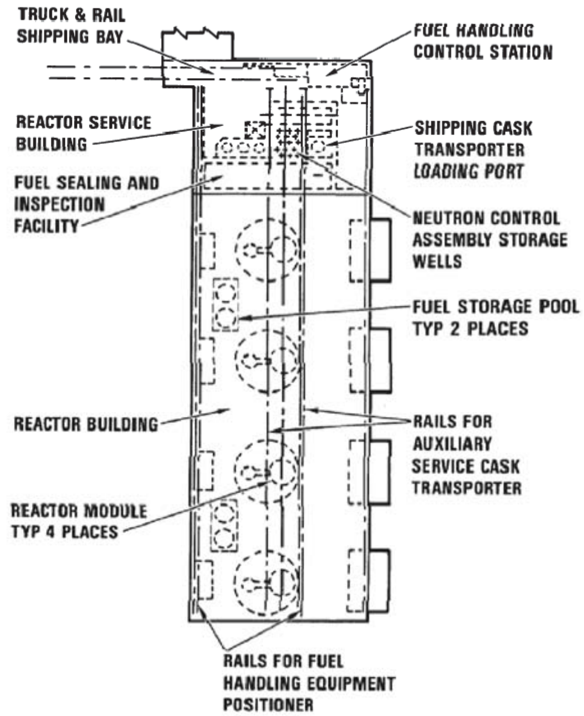


GENERAL ATOMICS

Vollman, R. (General Atomics) Prismatic HTGR Core Design Description, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Prismatic Fuel Handling – MHTGR (cont.)

Fuel Loading Deck of the
Fort. St. Vrain Core



Layout of a 4-module MHTGR

HTGR Core Design – Overview

- General Attributes of Modular Prismatic and Pebble Bed HTGRs
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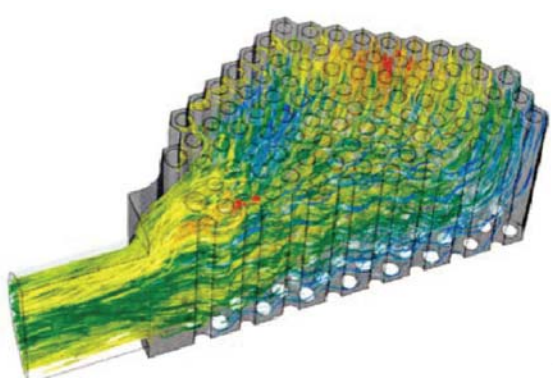
  - Power Conversion
- Normal Operation and Power Maneuvers

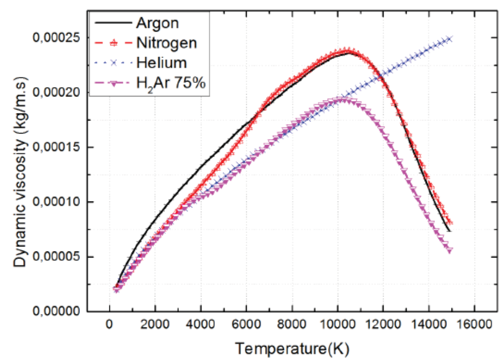
Coolant flow in Lower Plenum



Petti, D. et al (2019). Current Status of VHTR Technology Development.

Thermal-Fluidics

- Downward flow
  - Inlet coolant directed upward along the inside of the RPV to keep it and the Control Rod structures cool
  - Flow reverses during loss-of-force cooling (LOFC)
  - Complex mixing structure at core outlet to prevent thermal 'hot-striping' and stress on downstream components
- So much carbonaceous material…
  - Thermal transients are relatively slow
  - Heat transfer via conduction/radiation after a loss of force flow
- Helium
  - Neutronically transparent and chemically inert
  - Viscosity increases with temperature (potential stagnation in hot channels)



$$\mu(T) = 3.953\text{E}{-7}T^{0.687}\,\text{Ns}/m^2$$

Abderrahmane, Aissa, Mohamed, Abdelouahab, Noureddine, Abdelkader, El Ganaoui, Mohammed , Pateyron, Bernard. (2013). Ranz and Marshall correlations limits on heat flow between a sphere and its surrounding gas at high temperature. Thermal Science. 10.2298/TSCI120912090A.

Melese and Katz, "Thermal and Flow Design of Helium-Cooled Reactors", American Nuclear Society, ISBN 0-89448-027-8, 1984.

Temperature Feedback

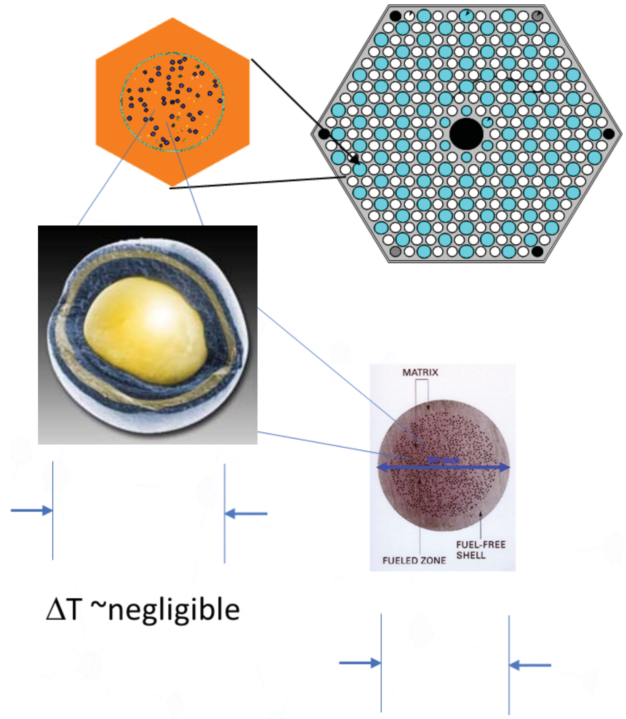
- The core will shut itself down in the event of a loss of coolant
- Enables load following with He mass flow control

Temperature Coefficients	Unit	Under Operating Conditions
Fuel (Doppler coefficient of mainly ^{238}U)	$\Delta\rho/°C$	-4.4×10^{-5}
Moderator	$\Delta\rho/°C$	-1.0×10^{-5}
Reflector regions (all together)	$\Delta\rho/°C$	$+1.8 \times 10^{-5}$
Total	**$\Delta\rho/°C$**	**-3.6×10^{-5}**

Venter, P. (PBMR) Module 6b – Pebble Bed HTGR Nuclear Design, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Heat Deposition

- Kernels are small, but still larger than the recoil distance of fission products ⇒ most of the fission heat is deposited in the kernel, but…
- This heat dissipates easily into the surrounding matrix, so for all but the most extreme (beyond esign basis) reactivity spikes, the particles are largely in thermal equilibrium with the surrounding matrix, even during transients
- This allows one to define the 'fuel temperature' as the compact or fueled region of the pebble

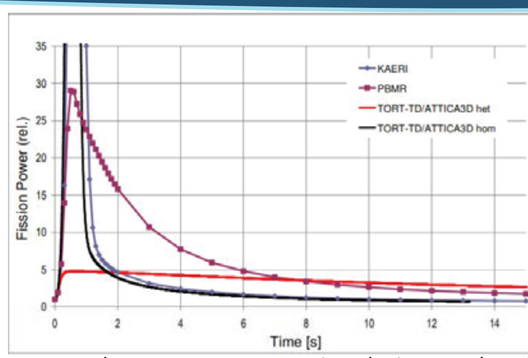


ΔT ~negligible

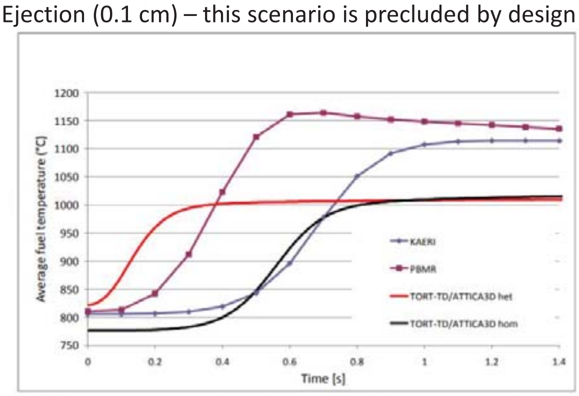
ΔT ~60-80C

Explicit Particle Heat Deposition Models

- Some codes have been developed with a 'subgrid' model of heat deposition only in the kernel and transient heat conduction out of the particles and into the matrix
- Results show very different fuel temperature and power trajectories between 'smeared' and explicit models for large (and in some cases unphysical) transients
- The smeared fuel models are generally much more conservative – kernel-limited heat deposition leads to faster Doppler turnaround



Power and temperature excursion during Total Rod Ejection (0.1 cm) – this scenario is precluded by design



Lapins, Janis and Seubert, A and Buck, Michael and Bader, Jo and Laurien, E. (2011). Tort-td/AtticA3D: A Coupled Neutron Transport and Thermal Hydraulics Code System for 3-D Transient Analysis of Gas Cooled High Temperature Reactors. 10.13140/2.1.3526.3369.

Ortensi, J., Boer, B, and Ougouag, A,. Thermo-mechanical Analysis of Coated Particle Fuel Experience a Fast Control Rod Ejection, Proceedings of the 5th International Topical Meeting on High temperature Reactor Technology (HTR2010), Prague, October 2010.

Hu, Jianwei and Uddin, R., 3D Thermal Modeling of TRISO Fuel Coupled with Neutronic Simulation, LA-UR-10-00442, Los Alamos national Lab, 1 January 2010.

Core Thermal-Fluidics: Prismatic

- To first order, heat transfer during power operation can be captured with 1-D pipe flow models and 2-D heat conduction

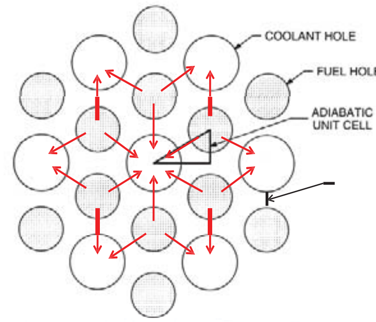
$$Nu_{FD} = 0.023\ \mathrm{Re}_b^{0.8}\ \mathrm{Pr}_b^{1/3}(\mu_b/\mu_w)^{0.14}$$

Seider-Tate Correlation for single phase flow in a circular channel

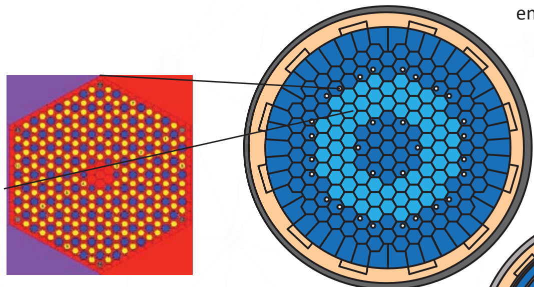
$$Nu_{FD} = 0.021\mathrm{Re}_b^{0.8}\mathrm{Pr}_b^{0.4}(T_w/T_b)^{-0.50}$$

$$Nu = Nu_{FD}[1 + (z/D)^{-0.70}]$$

McEligot Correlation for fully developed flow and entrance effects

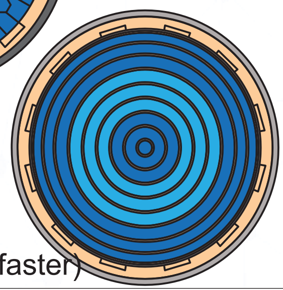


Seker, V.. (2007). Multiphysics methods development for high temperature gas reactor analysis. ETD Collection for Purdue University.

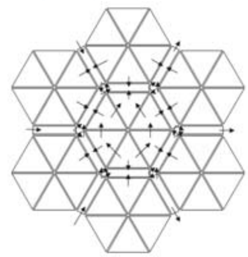


Block-wise resolution

Homogenization used in RELAP5



Ring-wise resolution (faster)

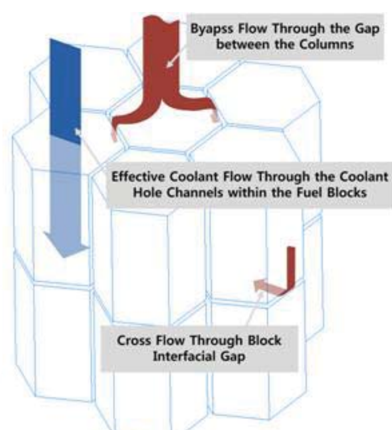


Homogenization/ Network Model used in AGREE/GASNET

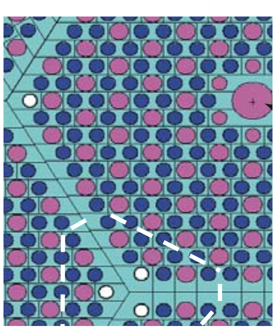
Triangle-wise resolution

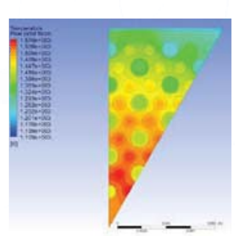
Core Thermal-Fluidics: Prismatic (cont.)

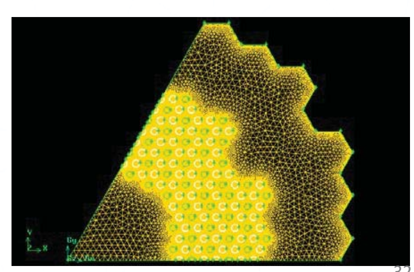
- Dimensional changes in graphite lead to alternate coolant pathways (bypass flow) – significantly altering the temperature profile in the core and reflector. Bypass flows can be modeled as extra channels in network codes
- Little momentum upon loss of pumping power, coolant quickly slows (relaminarization) and is then driven by buoyancy. If there are significant bypass gaps, radiation across the gaps becomes a dominant heat transfer mechanism
- Transient analysis are still performed with the simple, homogenized block (or subblock) models. Coarse mesh CFD methods may be an adequate compromise (PRONGHORN?)



Richard W. Johnson, Hiroyuki Sato, and Richard R. Schultz. CFD Analysis of Core Bypass Phenomena. United States: N. p., 2009. Web. doi:10.2172/974775.







Core Thermal-Fluidics: Pebble Bed (cont.)

Convective heat transfer in a packed bed

$$h = \frac{Nuk_{He}}{D_p}$$

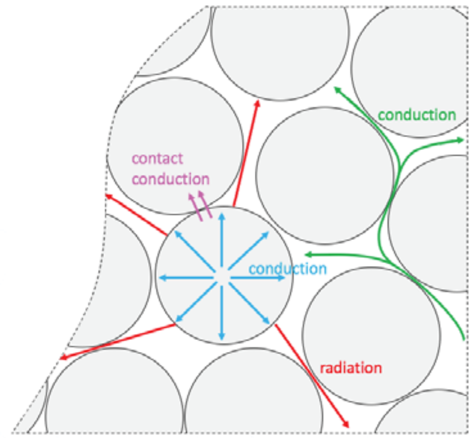
$$Nu = f_e Nu_s$$

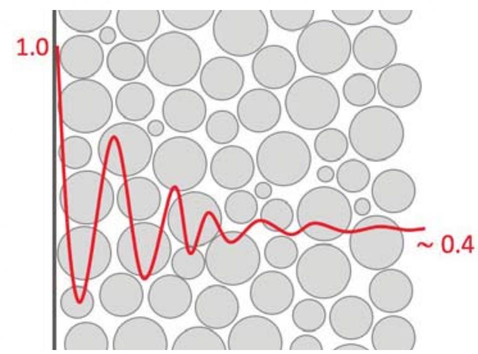
$$Nu_s = 2 + \sqrt{Nu_l^2 + Nu_t^2} \qquad \varepsilon \sim 0.40$$

$$f_e = 1 + 1.5(1-\varepsilon)$$

$$Nu_l = 0.664\left(\frac{\mathrm{Re}}{\varepsilon}\right)^{1/2} \mathrm{Pr}^{1/3}$$ Laminar component

$$Nu_t = \frac{0.037\left(\frac{\mathrm{Re}}{\varepsilon}\right)^{0.8} \mathrm{Pr}}{1 + 2.443\left(\frac{\mathrm{Re}}{\varepsilon}\right)^{-0.1}\left(\mathrm{Pr}^{2/3} - 1\right)}$$ Turbulent component





Other correlations have been developed to capture variable porosity, wall effects, radiation and conduction under low flow conditions

CFD models of local geometries have been executed and avoid many of these empirical assumptions

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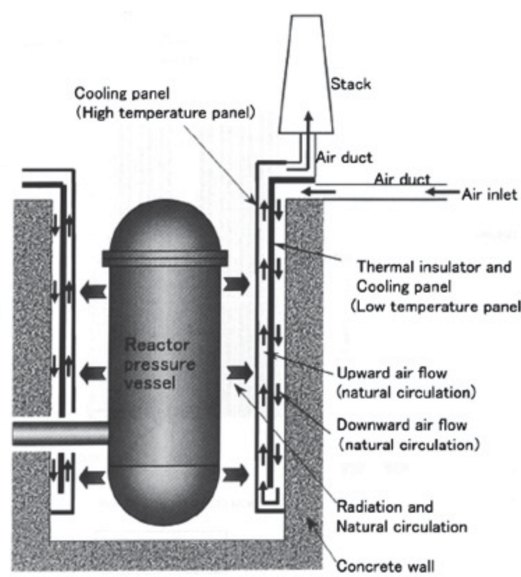
Pressure Vessel in the Reactor Cavity

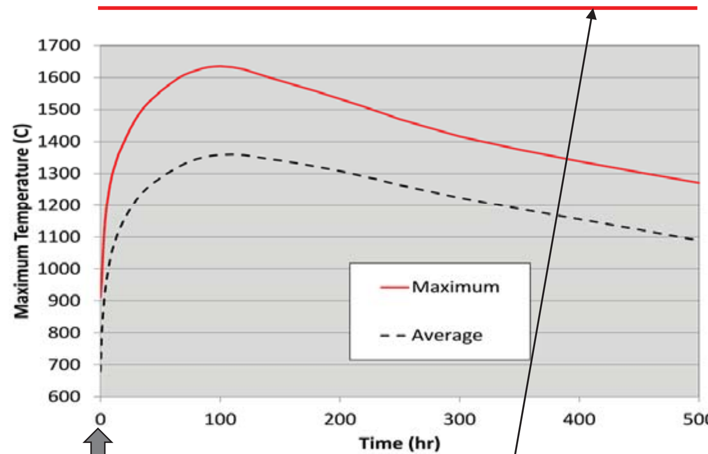


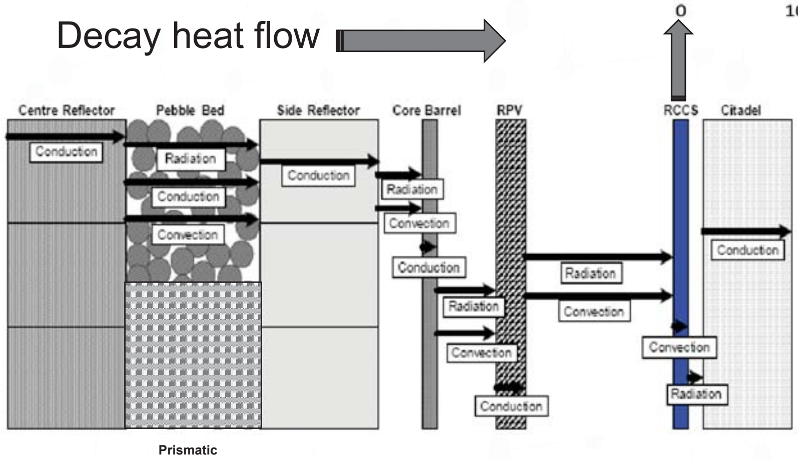
Kuniyoshi Takamatsu, Tatsuya Matsumoto, Koji Morita, New reactor cavity cooling system (RCCS) with passive safety features: A comparative methodology between a real RCCS and a scaled-down heat-removal test facility, *Annals of Nuclear Energy*, Volume 96, 2016.

Inherent Safety

'Grace period' (no operator intervention) measured in tens or hundreds of hours

Core temperatures during a DLOFC
AREVA Technical Document 12-9251926-001,
Summary Report-SC-HTGRE Demonstration
Reactor



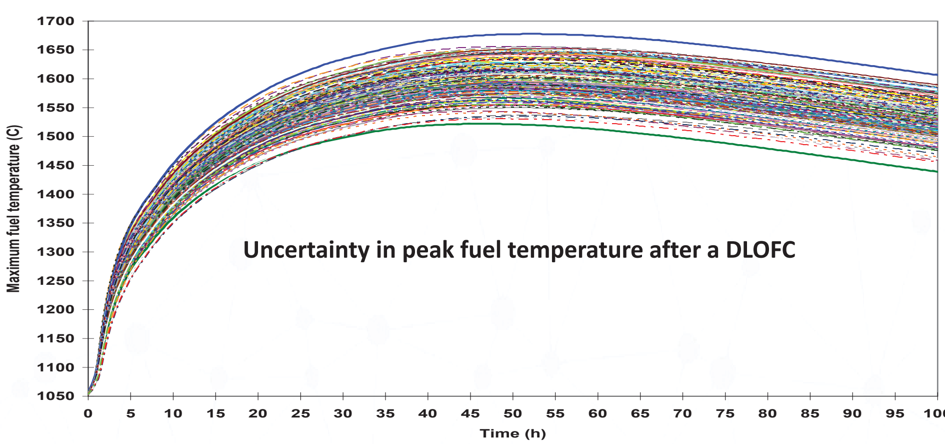


1800C – No appreciable UCO particle failures observed in AGR heating test at this temperature although accelerated diffusion of certain FP (Sr, Cs, Eu) is observed.

Courtesy of F. Reitsma, IAEA

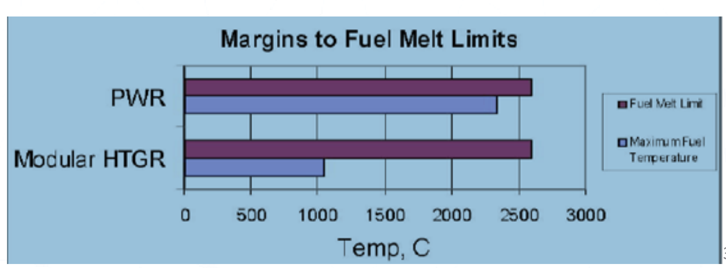
Large Margins to Particle Failure Temperature

1800°C beyond 150 h

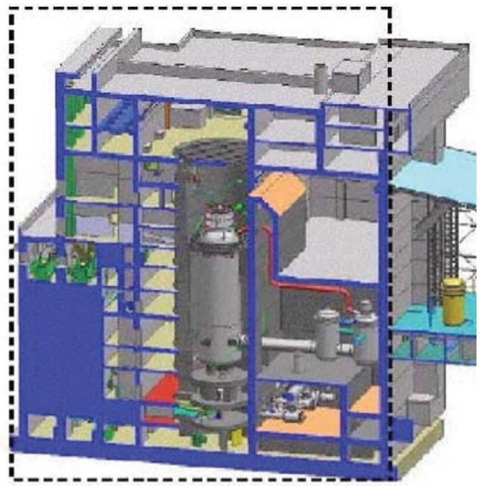


Uncertainty Analysis of Peak Fuel Temperature during a DLOFC in the PBMR400, G. Strydom, INL. Different trajectories obtained by varying input parameters over expected ranges (graphite conductivity, etc.)

Margins are large;
grace periods are long.



Radiological Release Sequence and the Vented Reactor Building Concept



Cutaway diagram of the PBMR-400 Demonstration Plant (PBMR (Pty) Co. Ltd)

Buildup: During operation small amounts FP diffuse out of the fuel/graphite (limited by He Purification System)

- Some (e.g. Ag) adsorb onto cooler surfaces in the primary loop
- Others (Eu, Cs, Sr) remain as 'circulating inventory'

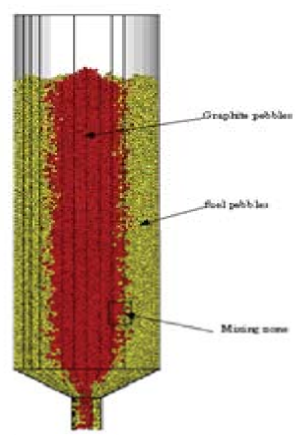
Puff: After a significant break, circulating inventory is released and vented from the building

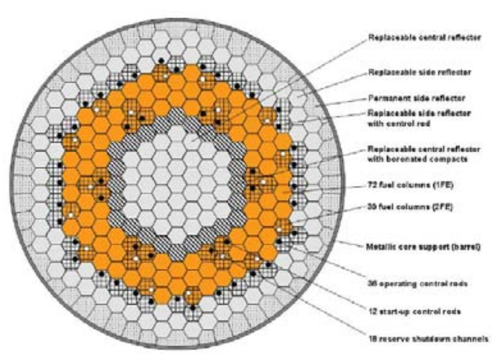
Cook (heatup/cooldown): After depressurization, the vents are closed. FP-driven heatup of the core drives additional releases from the fuel, some of which will eventually make its way out of the building.

D. A. Petti, R. R. Hobbins, P. Lowry and H. Gougar (2013) Representative Source Terms and the Influence of Reactor Attributes on Functional Containment in Modular High-Temperature Gas-Cooled Reactors, Nuclear Technology, 184:2, 181-197, DOI: 10.13182/ NT184-181

Core Analysis Summary

- Big graphite cores pose an interesting challenge for core modelers, especially for transient analysis
- Fortunately,
  - Safety parameters (fuel failure temperatures and fission product release rates) are not overly sensitive to neutronics parameters
  - Grace periods are long (many hours or days rather than minutes)
  - No coolant phase change
- High fidelity tools (Monte Carlo transport and CFD) are useful mainly for quantifying uncertainties; they are not essential for routine core design yet, but we're moving in that direction
- Still, some features of modular HTGRs pose challenges to traditional LWR methods (moving fuel, burnable poisons, spectral leakage). Modern tools are better suited to tackling these features in a rigorous way





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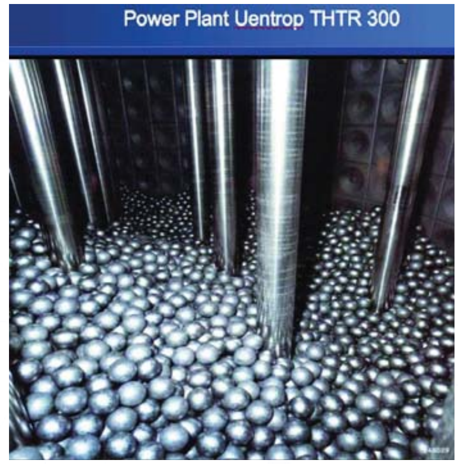
THTR featured a Shutdown CR system in which the rods were forced into the pebble bed. It was designed to be used only intermittently but unintended scrams were frequent. Broken pebbles were a result.

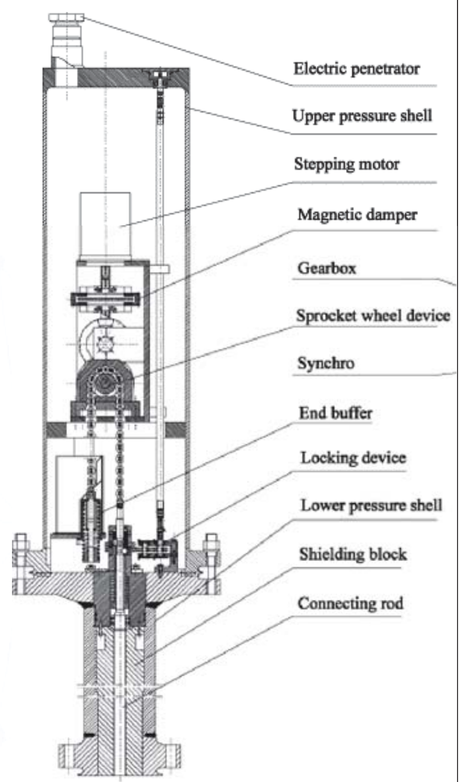


Daoud, H., Serries, F., & Schollmeyer, H. (1989). Operating experience with the THTR core control rods. Germany: INFORUM Verl. (available through IAEA INIS)

Reactivity Control Requirements

mHTGR-DC 26, NRC Reg guide 1.232

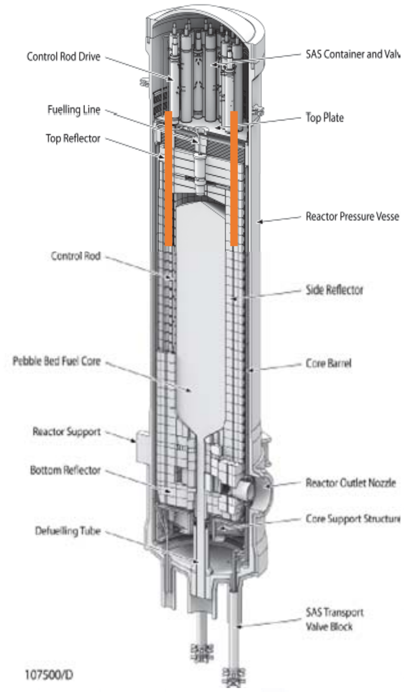
- A means of **inserting negative reactivity at a sufficient rate** and amount to assure… radionuclide release limits and He pressure design limits are not exceeded and safe shutdown is achieved…
- A means which is **independent and diverse** from the other(s), shall be capable of controlling the rate of reactivity..
- A means of inserting negative reactivity at a sufficient rate and amount to assure, … that the **capability to cool the core is maintained and a means of shutting down the reactor** and maintaining, at a minimum, a safe shutdown condition…
- A means for h**olding the reactor shutdown under conditions which allow for interventions** such as fuel loading, inspection and repair shall be provided.



Wu Yuanqiang, Diao Xingzhong, Zhou Huizhong, Huang Zhiyong, Design and tests for the HTR-10 control rod system, Nuclear Engineering and Design, Volume 218, Issues 1-3, 2002.

Reactivity Control

- Typical: Two independent rod banks
- Articulated rods suspended from drives by chains to be lowered into the radial reflector
- Bypass flow cools the rods
- May be partially inserted during power operation to provide Xe restart/load follow capability
- Some load following can be achieved with He flow control
- Prismatic – Shutdown rods can inserted into fuel blocks
- PBR – Small absorber spheres have been proposed for past designs (not in X-energy XE-100)



Both AVR and HTR-10 can be shut down without rods – circulators are stopped to affect a core heatup and Doppler shutdown.

Shutdown Cooling System (SCS)

SCS Protection System

Following detection of:

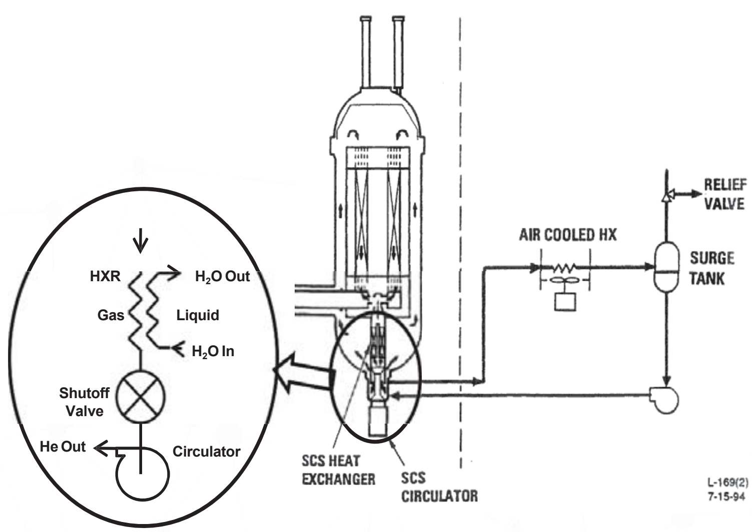
- Heat Exchanger Leaks
- Circulator Overspeed
- Low Cooling Water Flow
- Loss Of Net Positive Suction Head
- High Heat Exchanger temperatures

Actions:

- Shutoff Valve actuated
- Circulator shutdown

Components List

- He Circulator
- He Shutoff Valve
- Gas to Liquid Heat Exchanger
- Control System
- Shutdown Water Cooling System
- Service Equipment



Single Shutdown Cooling System Loop per Reactor Module

OAK RIDGE NATIONAL LABORATORY
MANAGED BY UT-BATTELLE FOR THE DEPARTMENT OF ENERGY

ORNL/TM-2012/107

HTGR Measurements and Instrumentation Systems

May 2012

Prepared by
S. J. Ball
D. E. Holcomb
S. M. Cetiner

UT-BATTELLE

ORNL-27 (4-00)

Helium Conditioning

- Removes chemical and radionuclide impurities from helium coolant
- Pressurizes, depressurizes, and controls the primary helium coolant inventory in conjunction with Helium Transfer and Storage System (HT&SS)
- Provides purified helium for purges and buffers
- Maintains primary coolant system at a slightly subatmospheric during refueling/maintenance
- Purifies helium pumped to storage
- Removes H_2O from primary circuit following water ingress event

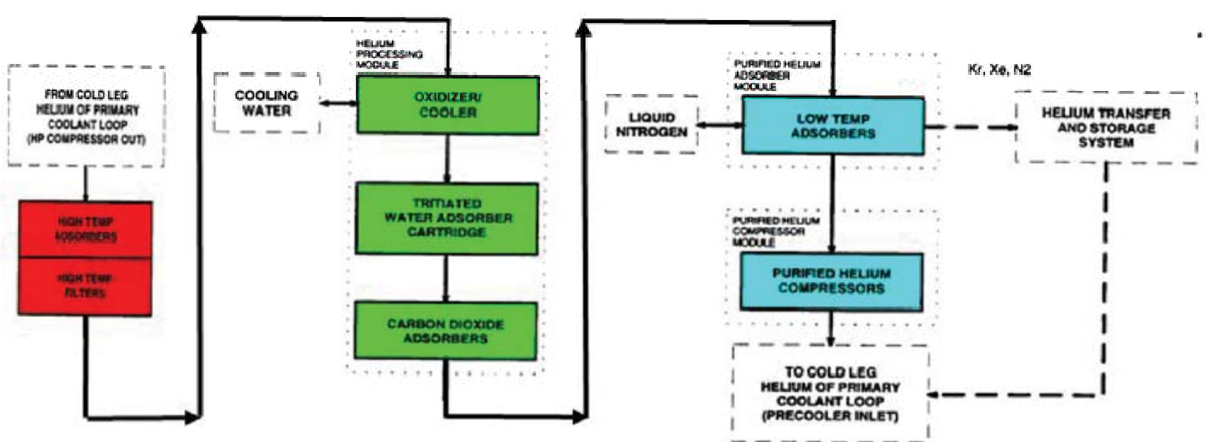
Hanson, D. (General Atomics) – Module 10c - Helium Inventory and Purification, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Helium Purification System Requirements (General Atomics)

- Each reactor module shall have an independent helium purification system
- Shall remove H_2O, CO, CO_2, H_2, N_2, O_2, H_2S, CH_4, and higher molecular weight hydrocarbons
- Shall allow depressurization of the reactor module (and/or adjacent module) within 24 hours after shutdown
- Shall include one regeneration train for two HPS
- Shall be sized to process a slipstream of the primary coolant, typically on the order of 1% of the primary loop volume flow rate

Hanson, D. (General Atomics) – Module 10c - Helium Inventory and Purification, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

HPS Train (General Atomics)



Contaminants Removed

Red
Non-condensable Radionuclides and Particulates

Green
H_2O, including tritiated H_TO and CO_2

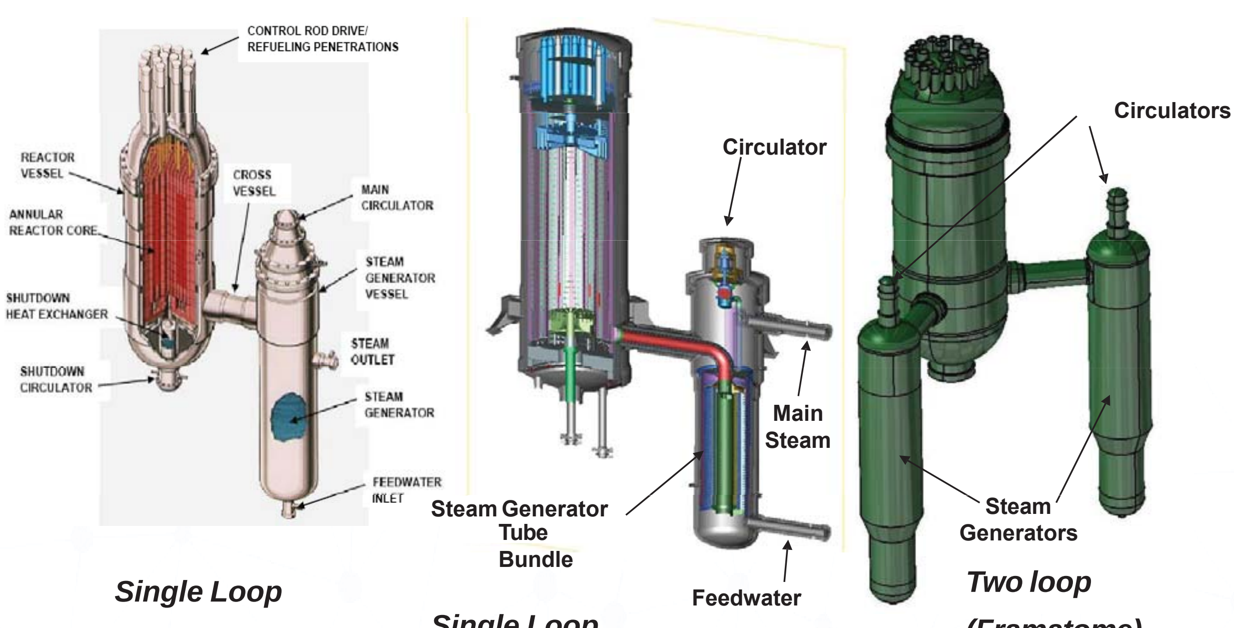
Blue
Kr, Xe, N_2

Lessons Learned from Fort St. Vrain (General Atomics)

- HPS and Helium Transport and Storage System (HT&SS) performed well in seven steam-cycle HTGRs
- Specific lessons from FSV (and AVR)
  - HPS overwhelmed by large H_2O ingresses; long times required for dry out of primary coolant circuit
  - Single transfer compressor required taking plant offline for compressor maintenance
- Components performed well except for Ti Getter Beds in FSV
  - FSV used Ti getter beds instead CuO oxidizers/driers for the removal of hydrogen and tritium
  - No operational consequences because H2 and H-3 sorbed onto core structures
- Design recommendations for future HTGRs:
  - Provide suitable drains for removal of standing water
  - Provide backup He transfer compressor
  - Use CuO oxidizer beds/driers for H2 and H-3 removal

Hanson, D. (General Atomics) – Module 10c - Helium Inventory and Purification, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Power Conversion



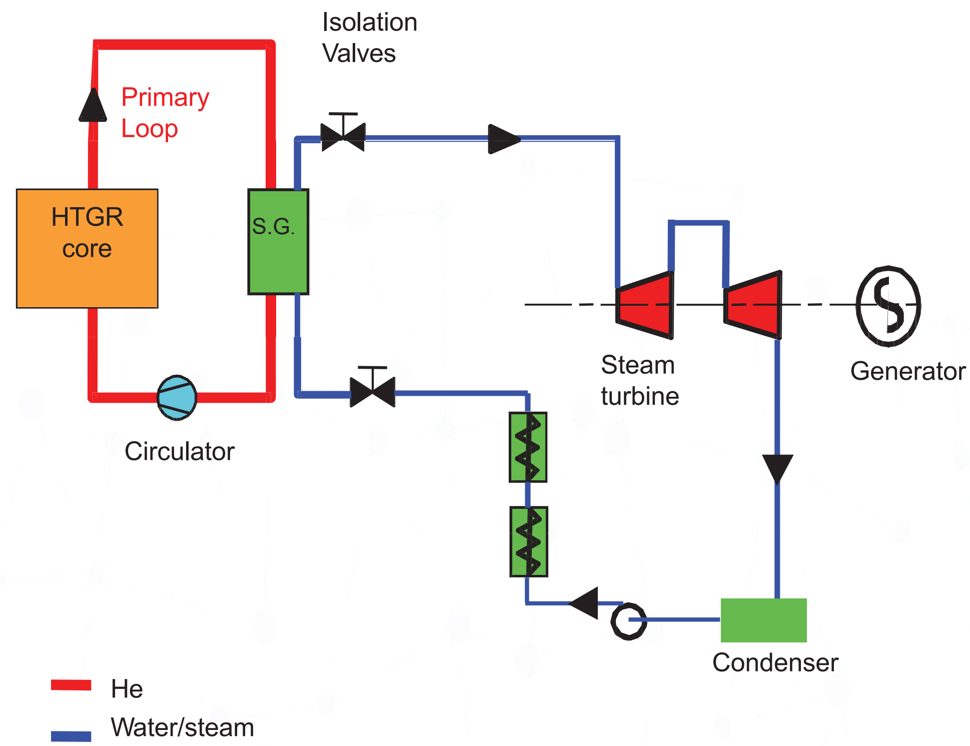
Single Loop (MHTGR)

Single Loop (PBMR-CG)

Two loop (Framatome)

Lommers, L. (Framatome) – Module 10b - Steam Cycle Power Conversion System, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

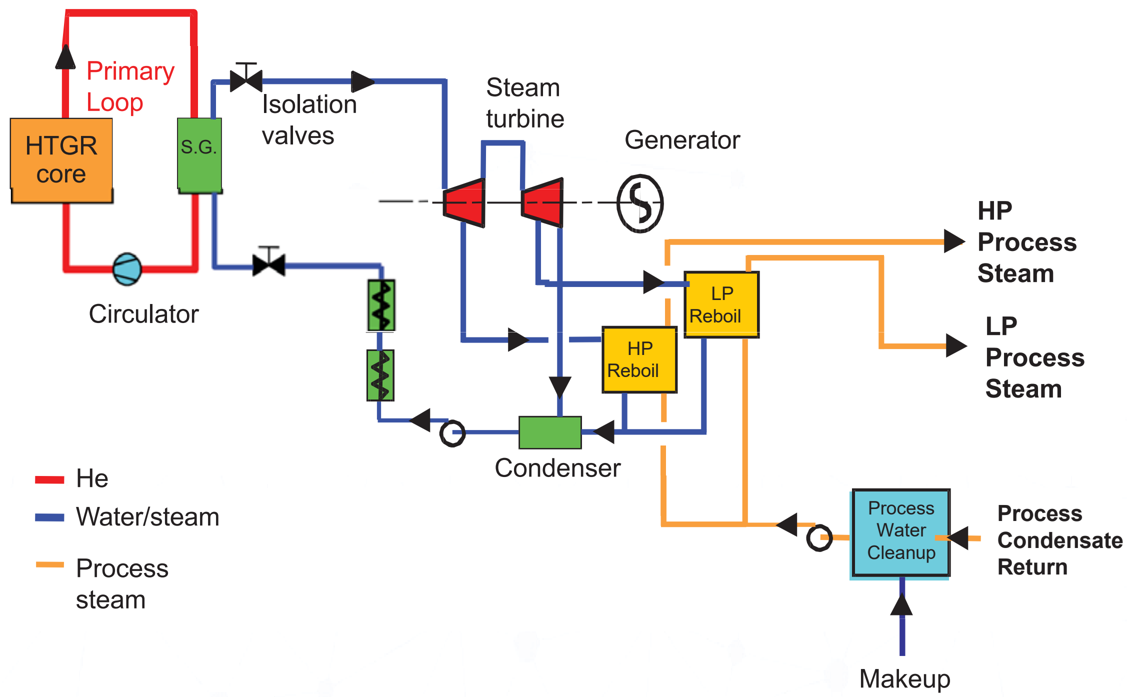
Conventional Rankine Cycle



Steam and Process Heat Considerations

- Process steam pressure/temperature
- Process steam quantity
- Operating flexibility
  - Response to varying user steam demands
  - Flexibility for varying steam vs. electricity production
- Operational interaction between steam supply units and process users
- Process steam contamination concerns
- Feedwater quality control
- Process steam reliability concerns
  - Availability
  - Service interruption

Steam and Process Heat



Other Considerations

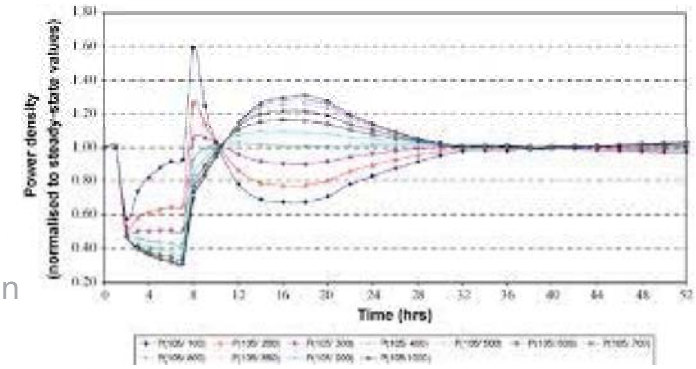
- Steam cycle and process heat components would use established fossil-driven technology
  - Coupling to an HTGR remains an issue
- Helium Circulators
  - Good experience from United Kingdom reactors
  - Magnetic bearings, submerged motors
  - Size is within vendor range
- Steam generators
  - Experience in HTGRs is more benign that PWRs (no shell-side CRUD)
  - HTGRs more robust
  - Problems with HTR-PM design delayed schedule
- Other Rankine cycle components
  - Well-within vendor experience base
- Reboiler (for Process Heat)
  - Used in fossil-drive process heat
  - New to HTGRs – will be customized

Anderson, N. and Sabharwall, P. (2014). RELAP5-3D transient modelling for NGNP integrated plant. Int. J. of Nuclear Energy Science and Technology. 8. 213 - 237. 10.1504/IJNEST.2014.063015.

HTGR Core Design – Overview

- General Attributes of Modular Prismatic and Pebble Bed HTGRs
  - Common Features and Physics
  - Neutronics
  - Prismatic and Pebble Fuel
  - Thermal-Fluidics
  - Inherent Safety
- Plant Systems and Power Conversion
  - Reactivity Control
  - Instrumentation and Control
  - Helium Conditioning
  - Power Conversion
- Normal Operation and Power Maneuvers

100-50-100% Load Follow Trajectory



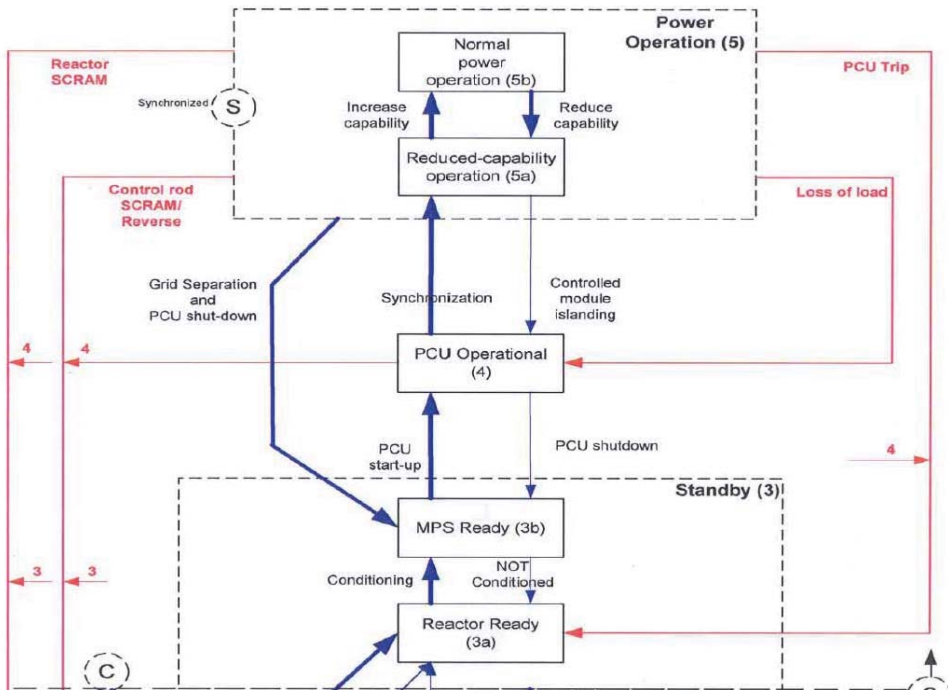
Strydom, G. (2019). Xenon-induced axial power oscillations in the 400 MW pebble bed modular reactor. Thesis (M.Sc. (Reactor Science))--North-West University, Potchefstroom Campus, 2008.

Power Operation/Load Follow

- Various maintenance, shutdown, standby and operational states are usually defined (PBMR example shown)
- Transitions between various modes/states can be complex (next slide)
- Convective heat transfer dominates during steady-state
- Flexible load-follow capability via helium mass flow rate control allows 100-40-100 e.g. power maneuvering to follow demand (PBMR limited to 1%/min)
- Load follow range mostly limited by excess fuel (+) and control rod (-) reactivity available to counter xenon swings

- Power Operation (Mode 5)
  - 100% MCR Load
  - 40% MCR load
- PCU Operational (Controlled Island Operation (Mode 4)
- Standby (Mode 3)
  - Main Power System ready
  - Reactor ready
- Shutdown (Mode 2)
  - Partial (control rods inserted only)
  - Intermediate (control rods and shutdown rods inserted)
  - Full (all rods and small absorber spheres inserted)
- Fueled Maintenance (Mode 1)
  - Helium Pressure Boundary closed
  - Open Power Conversion Unit
- Defueled Maintenance (Mode 0)

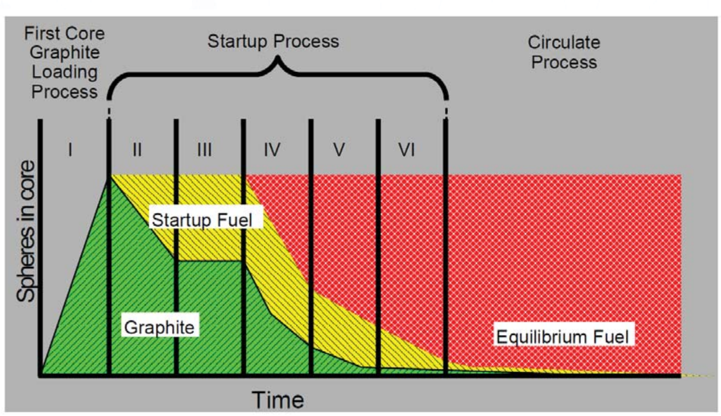
Operating Modes and States (PBMR example)



H.W. Chi, 2006. Presentation on PBMR Safety and Design Familiarization.

Transition from Startup to Equilibrium Core

- Core is initially filled with graphite spheres, and first critically is reached with mixture of graphite and fuel spheres
- Core "running-in" phase is an optimization problem with multiple constraints:
  - peak fuel temperature <1130°C
  - maximum power <4.5 kW/sphere
  - minimize fuel costs - limit fuel types to two enrichments
  - minimize time-to-full-power (revenue $ vs. time)
- Example – "revenue $ vs. time" (above) leads to discharging low-enriched start-up fuel out of the core as quickly as possible, but fuel (and fuel $) is wasted



H. Chi, 2006: Presentation on PBMR Safety and Design Familiarization

Summary

- HTGRs occupy a special niche in the nuclear power world: really high temperatures for process heat, but still passively safe
- (A few) HTGRs have been around awhile – a modular version is about to start up in China
- The low power density, coated particle fuel, and graphite effectively eliminate the possibility of a meltdown. Process heat user can set up operations next door
- The physics are dominated by the graphite
- Neutronics can be challenging, but approximate methods work reasonably well if margins are quantified and care is taken with cross section generation. High fidelity neutronics are showing promise for reducing uncertainties
- Thermal-fluidics can also be approximated with low order models, but higher fidelity models are desired. Full-core CFD is still out of reach for all but a few reference calculations
- Helium conditioning was demonstrated on Fort St. Vrain
- Steam cycle power conversions systems can exploit extensive technology developed for the fossil fuel industry; some specific components will need to be designed

Suggested Reading List

- A.M. Ougouag, H.D. Gougar, R.S. Sen, "Identification of Spectral Zone Boundaries in Pebble Bed Reactors," Proceedings of 9th International Topical Meeting on High Temperature Reactor Technology, (HTR 2018), Warsaw, 2018.
- Abderrahmane, Aissa, Mohamed, Abdelouahab, Noureddine, Abdelkader, El Ganaoui, Mohammed , Pateyron, Bernard. (2013). Ranz and Marshall correlations limits on heat flow between a sphere and its surrounding gas at high temperature. Thermal Science. 10.2298/TSCI120912090A.
- Allelein, H.J., et al. 2016. First results for fluid dynamics, neutronics and fission product behavior in HTR applying the HTR code package (HCP) prototype. Nuclear Engineering and Design, vol. 306, pp. 145–153.
- Anderson, N. and Sabharwall, P. (2014). RELAP5-3D transient modelling for NGNP integrated plant. Int. J. of Nuclear Energy Science and Technology. 8. 213 - 237. 10.1504/IJNEST.2014.063015.
- HTGR Technology Course for the Nuclear Regulatory Commission, 2010.
- J Ball, Sydney & Holcomb, David & Cetiner, Sacit. (2012). HTGR Measurements and Instrumentation Systems. 10.2172/1040751.
- Bechtel National, Inc., et al. 1986. Preliminary safety information document for the standard MHTGR. HTGR-86-024. Stone and Webster Engineering Corporation.
- Beeny, B. & Vierow, K., 2015. Gas-cooled reactor thermal hydraulic analyses with MELCOR. Progress in Nuclear Energy, Volume 85, pp. 404-414.
- Bomboni, Eleonora and Cerullo, Nicola and Lomonaco, Guglielmo and Romanello, Vincenzo. (2008). A Critical Review of the Recent Improvements in Minimizing Nuclear Waste by Innovative Gas-Cooled Reactors. Science and Technology of Nuclear Installations. 10.1155/2008/265430.
- Bostelmann, F. et al., 2018. Assessment of SCALE Capabilities for High Temperature Reactor Modeling and Simulation. Transactions of the American Nuclear Society, Vol. 119, Orlando, Florida, November 11–15, 2018.
- Brey, H.L., 2003. The Evolution and Future Development of the High Temperature Gas Cooled Reactor. Proceedings of GENES4/ANP2003, Sep. 15-19, 2003, Kyoto, Japan.
- By Cschirp at the German language Wikipedia, CC BY-SA 3.0, https://commons.wikimedia.org/w/index.php?curid=11451341.
- C. H. Rycroft, G. S. Grest, J. W. Landry, and M. Z. Bazant, Analysis of Granular Flow in a Pebble-Bed Nuclear Reactor, Phys. Rev. E **74**, 021306 (2006). PFC3D – Itasca Consulting Group.
- Cetnar, J. et al, Assessment of Pu and MA utilisation in deep burn Prismatic HTR by Monte Carlo Method – MCB, Project PUMA, AGH-University of Science and Technology, Krakow, Poland, 2013.

Suggested Reading List (cont)

- Cogliati, J., "PEBBLES: A Computer Code for Modeling Packing, Flow, and Recirculation of Pebbles in a Pebble Bed Reactor," Proceedings of 5th International Topical Meeting on High Temperature Reactor Technology, (HTR 2010), Prague, 2010.
- Connolly, K.J., Rahnema, F., Tsvetkov, P.V., 2015. Prismatic VHTR neutronic benchmark problems. Nuclear Engineering and Design 285, pp. 207-240.
- Daoud, H., Serries, F., & Schollmeyer, H. (1989). Operating experience with the THTR core control rods. Germany: INFORUM Verl. (available through IAEA INIS)
- D. A. Petti, R. R. Hobbins, P. Lowry and H. Gougar (2013) Representative Source Terms and the Influence of Reactor Attributes on Functional Containment in Modular High-Temperature Gas-Cooled Reactors, Nuclear Technology, 184:2, 181-197, DOI: 10.13182/ NT184-181
- Gougar, H.D., Ougouag, A. M., Terry, W. K., and Ivanov, K. I., "Automated Design and Optimization of Pebble Bed Reactor Cores, Nuclear Science and Engineering, Oct. 2008.
- H. Chi, 2006: Presentation on PBMR Safety and Design Familiarization.
- H. Gougar, A. Ougouag, J. Ortensi, C. Rabiti, "Suitability of Energy Group Structures Commonly Used in Pebble Bed Reactor Core Diffusion Analysis as Indicated by Agreement with Transport Theory for Selected Spectral Indices," Proceedings of the 9th International Topical Meeting on HTR Technology (HTR 2018), Warsaw, October 2018.
- H. Gougar, A. Ougouag, W. Yoon, "Multiscale Analysis of Pebble Bed Reactors," Proceedings of 5th International Topical Meeting on High Temperature Reactor Technology, (HTR 2010), Prague, 2010.
- H.W. Chi, 2006. Presentation on PBMR Safety and Design Familiarization.
- Hanson, D. (General Atomics) – Modle 10c - Helium Inventory and Purification, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.
- Henry, K., et al. 2018. Modular High Temperature Gas Reactor Core Modeling with RELAP5-3D/PHISICS – Optimization Schemes for Load Following. Proc. of HTR 2018, Warsaw, Poland, October 8-10, 2018.
- Hu, Jianwei and Uddin, R., 3D Thermal Modeling of TRISO Fuel Coupled with Neutronic Simulation, LA-UR-10-00442, Los Alamos national Lab, 1 January 2010.
- Ilas, G., et al. 2011. Validation of SCALE for High Temperature Gas-Cooled Reactor Analysis. NUREG/CR-7107, ORNL/TM-2011/161, Oak Ridge National Laboratory.
- John D. Bess, Leland M. Montierth, Oliver Köberl and Luka Snoj (2014) Benchmark Evaluation of HTR-PROTEUS Pebble Bed Experimental Program, Nuclear Science and Engineering, 178:3, 387-400,DOI: 10.13182/NSE14-13
- Kadak, A. C., 2016. The Status of the US High-Temperature Gas Reactors. Engineering, Vol. 2 (2016), pp. 119-123.
- Kingrey, K., "Fuel Summary for Peach Bottom Unit 1 High Temperature Gas-Cooled Reactor Cores 1 and 2", INEEL/EXT-03-00103, April 2003.

Suggested Reading List (cont)

- Kingrey, K., "Fuel Summary for Peach Bottom Unit 1 High Temperature Gas-Cooled Reactor Cores 1 and 2", INEEL/EXT-03-00103, April 2003.
- Kugeler, K. et al. 2017. The High Temperature Gas-cooled Reactor - Safety considerations of the (V)HTR-Modul. EUR 28712 EN, Joint Research Center.
- Kuniyoshi Takamatsu, Tatsuya Matsumoto, Koji Morita,New reactor cavity cooling system (RCCS) with passive safety features: A comparative methodology between a real RCCS and a scaled-down heat-removal test facility, *Annals of Nuclear Energy*, Volume 96, 2016.
- Küppers, C., et al. 2014. The AVR Experimental Reactor – Development, Operation, and Incidents Final Report of the AVR Expert Group. Forzungzentrum Juelich, Germany.
- Laboure, V., Ortensi, J., an Hummel, A., :HTTR 3-D Cross-Section Generation with Serpent and MAMMOTH, INL/EXT-18-51317, September 2018.
- Lapins, Janis and Seubert, A and Buck, Michael and Bader, Jo and Laurien, E. (2011). Tort-td/AtticA3D: A Coupled Neutron Transport and Thermal Hydraulics Code System for 3-D Transient Analysis of Gas Cooled High Temperature Reactors. 10.13140/2.1.3526.3369.
- Lisowski, D.D. et al, Experimental Observations of Natural Circulation Flow in the NSTF, Nuclear Engineering and Design 306, (2016) 124-132.
- Lommers, L. (Framatome) – Module 10b - Steam Cycle Power Conversion System, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.
- Massimo,L. "The Physics of High Temperature Reactors", ebook ISBN 9781483280288.
- Melese and Katz, "Thermal and Flow Design of Helium-Cooled Reactors", American Nuclear Society, ISBN 0-89448-027-8, 1984.
- Moorman, R. 2011. Phenomenology of Graphite Burning in Air Ingress Accidents of HTRs. Science and Technology of Nuclear Installations, Volume 2011, Article ID 589747.
- Novak, April and Zou, Ling and Peterson, John and C Martineau, R and Slaybaugh, R. (2018). Pronghorn: A Porous Media Thermal-Hydraulics Core Simulator and its Validation with the SANA Experiments.
- Ortensi, J. et al. 2018. Benchmark Analysis of the HTR-10 with the MAMMOTH Reactor Physics Application. INL/EXT-18-45453, Idaho National Laboratory.
- Ortensi, J., Boer, B, and Ougouag, A,. Thermo-mechanical Analysis of Coated Particle Fuel Experience a Fast Control Rod Ejection, Proceedings of the 5th International Topical Meeting on High temperature Reactor Technology (HTR2010), Prague, October 2010.
- Ougouag and Cogliati. "Earthquakes and Pebble Bed Reactors: Time-dependent Densification". Joint International Topical Meeting on Mathematics and Computation and Supercomputing in Nuclear Applications (M&C + SNA 2007) Monterey, California, April 15-19, 2007.
- Petti, D., et al. 2017. Advanced Demonstration and Test Reactor Options Study. INL/EXT-16-37867, Rev. 3. Idaho National Laboratory.
- Richard W. Johnson, Hiroyuki Sato, and Richard R. Schultz. CFD Analysis of Core Bypass Phenomena. United States: N. p., 2009. Web. doi:10.2172/974775.
- Rouxelin, P. 2019. Reactor physics uncertainty and sensitivity analysis of prismatic HTGRs. PhD dissertation, North Carolina State University.

Suggested Reading List (cont)

- Seker, V. 2007. Multiphysics Methods Development for High Temperature Gas Cooled Reactor. PhD Dissertation, Purdue University.
- She, D. et al. 2019. PANGU code for pebble-bed HTGR reactor physics and fuel cycle simulations. Annals of Nuclear Energy 126 pp. 48–58.
- Shenoy, A. (General Atomics) History and Evolution of HTGRs, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.
- Strydom, G., et al. 2016. Comparison of the PHPHISICS/RELAP5-3D Ring and Block Model Results for Phase I of the OECD/NEA MHTGR-350 Benchmark, INL/JOU-14-33637.
- Strydom, G. (2019). Xenon-induced axial power oscillations in the 400 MW pebble bed modular reactor. Thesis (M.Sc. (Reactor Science))--North-West University, Potchefstroom Campus, 2008.
- ISICS/RELAP5-3D ring and block model results for phase I of the OECD/NEA MHTGR-350 benchmark. Nuclear Technology 193(1).
- Su, Bingjing and Zhao, Zhongxiong and Chen, Jianwei and I. Hawari, Ayman. (2006). Assessment of on-line burnup monitoring of pebble bed reactor fuel by passive neutron counting. Progress in Nuclear Energy - PROG NUCL ENERGY. 48. 686-702. 10.1016/j.pnucene.2006.06.013.
- Sun, X. et al. 2018. CFD investigation of bypass flow in HTR-PM. Nuclear Engineering and Design 329, pp. 147-155.
- Tak, N, et al, CAPP/GAMMA+ code system for coupled neutronics and thermo-fluid simulation of a prismatic VHTR core, Annals of Nuclear Energy 92 (2016)
- Terry, W.K., et al. 2004. Preliminary Assessment of Existing Experimental Data for Validation of Reactor Physics Codes and Data for NGNP Design and Analysis. ANL-05/05. Argonne National Laboratory.
- Venter, P. (PBMR) Module 6b – Pebble Bed HTGR Nuclear Design, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.
- Vollman, R. (General Atomics) Prismatic HTGR Core Design Description, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.
- Wang, Lidong & Guo, Jiong & Li, Fu & Hou, Jason & Ivanov, Kostadin. (2016). Effect Of Double Heterogeneity Treatment On Neutronics Modeling of HTGR Unit Cell.
- Windes, W. et al, "Discussion of Nuclear-Grade Graphite Oxidation in Modular High Temperature Gas-Cooled Reactors", INL/EXT-14-31720, Idaho National Laboratory, 2017.
- Wu Yuanqiang, Diao Xingzhong, Zhou Huizhong, Huang Zhiyong, Design and tests for the HTR-10 control rod system, Nuclear Engineering and Design, Volume 218, Issues 1-3, 2002.
- Zhang, Z., et al. 2016. The Shandong Shidao Bay 200 MWe High-Temperature Gas-Cooled Reactor Pebble-Bed Module (HTR-PM) Demonstration Power Plant: An Engineering and Technological Innovation. Engineering 2 (2016), pp. 112–118.

Idaho National Laboratory

TRISO Fuel: Design, Manufacturing, and Performance

Paul Demkowicz

TRISO Fuel:
Design, Manufacturing, and Performance

Advanced Reactor Technologies

Idaho National Laboratory

Paul Demkowicz, Ph.D.
AGR Program Director

NRC HTGR Training July 16-17, 2019

Idaho National Laboratory

ART ADVANCED REACTOR TECHNOLOGIES

Course Module Objective

- Review TRISO fuel design, fabrication, and performance, with a focus on recent results and developments in the last ~15 years

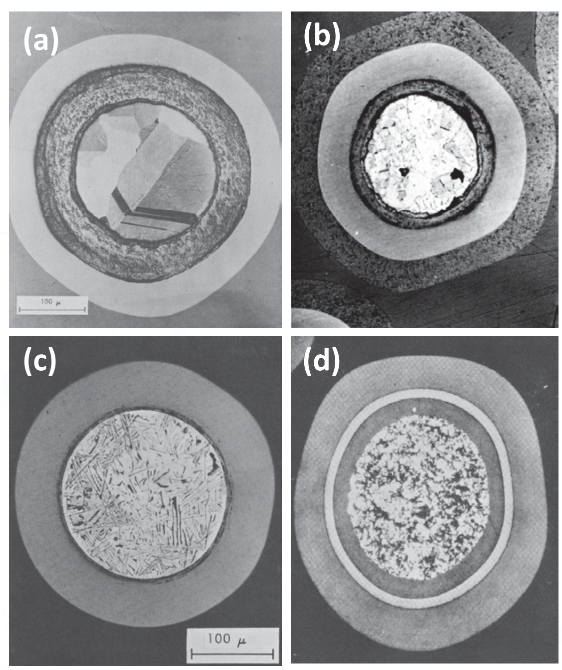
The Training Course delivered to the NRC in 2010 included several modules discussing TRISO fuel (Modules 7a, 7b, and 8). You are encouraged to review that course material for additional details on fuel fabrication and performance history.

Outline

- TRISO fuel background and history
- Fuel fabrication and quality control
- Fuel irradiation performance
- Fuel accident performance
- Fuel performance and fission product transport modeling

Coated Particle Fuel: Early History

- First developed in late 1950s to support Dragon reactor in UK
- Originated as single pyrocarbon layer to protect carbide kernels during fabrication
- Quickly evolved in 1960s into more sophisticated coating designs to provide fission product retention
- First demonstration reactors:
  - Dragon
  - Peach Bottom Unit 1
  - Arbeitsgemeinschaft Versuchsreaktor (AVR)

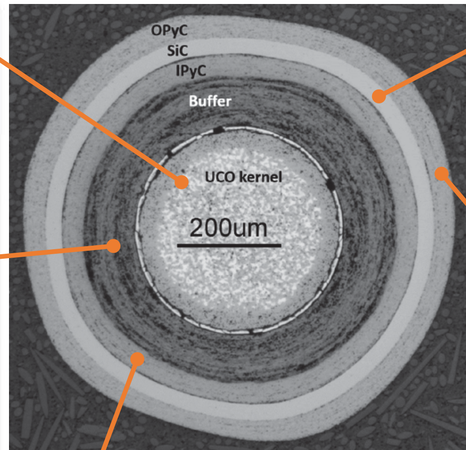


(a) Early example of a BISO (bistructural isotropic) particle. (b) Particle with "Triplex" structure (porous buffer layer followed by laminar and columnar pyrocarbon layers). (c) Carbide particle with single PyC coating layer used in Peach Bottom first core. (d) Fertile $(Th,U)C_2$ particle used in Dragon first charge, consisting of PyC-SiC-PyC structure.

P.A. Demkowicz et al., Coated particle fuel: Historical perspectives and current progress, J. Nucl. Mater. 515 (2019) 434-450

Modern TRISO Fuel

- **Kernel (350-500 µm)**
  - UO_2 or UCO
  - Retention of fission products
- **Buffer (~100 µm)**
  - ~50% dense pyrolytic carbon
  - Provides space for fission gas and CO(g) accumulation
  - Accommodates fission recoils

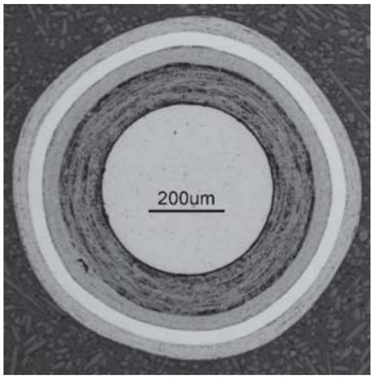


- **IPyC (~40 µm)**
  - Protects kernel from chloride during SiC deposition
  - Surface for SiC deposition
  - Contributes to fission gas retention
- **SiC (~35 µm)**
  - Main structural layer
  - Primary coating layer for retaining non-gaseous fission products
- **OPyC (~40 µm)**
  - Contributes to fission gas retention
  - Surface for bonding to matrix
  - Protects SiC layer during handling

TRISO Fuel Kernel Types

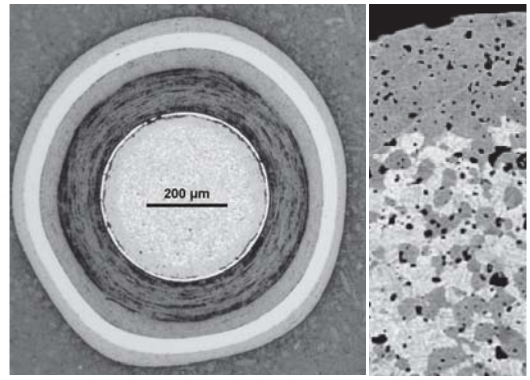
- Kernels are mechanically decoupled from the outer coating layers, giving great flexibility in kernel types
- HTGRs can use many fuel types
  - Fissile: UC_2, PuO_2, $(Th,U)C_2$, $(Th,U)O_2$, UO_2, UCO
  - Fertile: ThC_2, ThO_2, UO_2, UCO
- LEU UO_2 is most widely used fuel type
  - Used in AVR (Germany), HTTR (Japan), HTR-10 and **HTR-PM** (China)
  - Extensive irradiation and heating test database from German HTGR Program
  - Reference fuel type for PBMR
- UCO offers improved fuel performance at higher fuel burnup
  - UCO selected as reference fuel design by X-energy
  - Several countries involved in the Generation IV International Forum (GIF) Very High Temperature Reactor (VHTR) Fuel and Fuel Cycle (FFC) Project Management Board are pursuing R&D on UCO fuel fabrication based on the favorable US program results

UO_2 and UCO TRISO Fuel



UO_2

UCO
(mixture of UO_2 and UC_x)



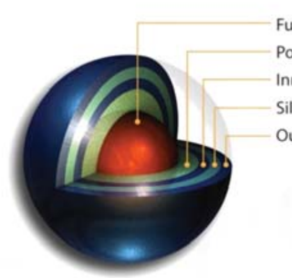
- Different kernel
- Same coatings

- Utilized in modern pebble bed reactor designs (burnup limited to ~11% FIMA)
- Extensive development and testing since the 1970s in many countries
- Good fission product retention in the kernel, but results in formation of CO(g) during irradiation
  - Contributes to internal gas pressure
  - Kernel migration, CO corrosion of SiC

- Mitigates CO(g) formation
- Suited for higher burnup (up to ~20% FIMA and beyond) and larger temperature gradients in prismatic reactors
- Comes at the cost of lower retention of some fission products in the kernel
- Development primarily in the US since the 1970s
- No large-scale, successful performance demonstration through the early 2000s

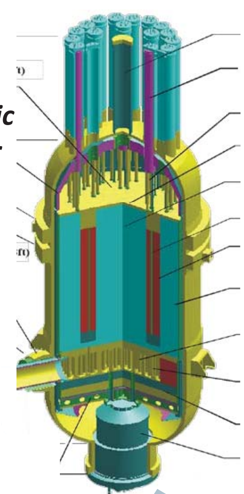
Tristructural Isotropic (TRISO) Coated Particle Fuel

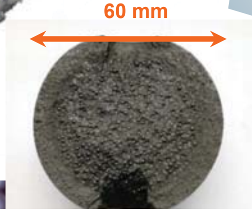
Pebble bed reactor

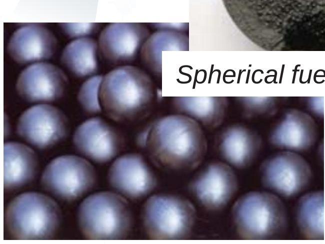


TRISO particle

Prismatic reactor

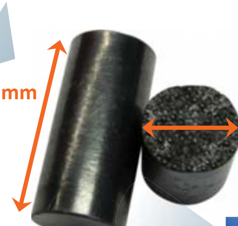




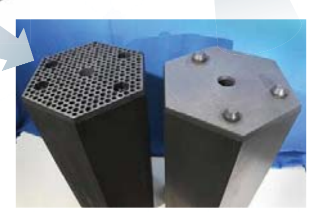


Spherical fuel pebbles

Particle design provides excellent fission product retention in the fuel and is at the heart of the safety basis for high temperature gas reactors



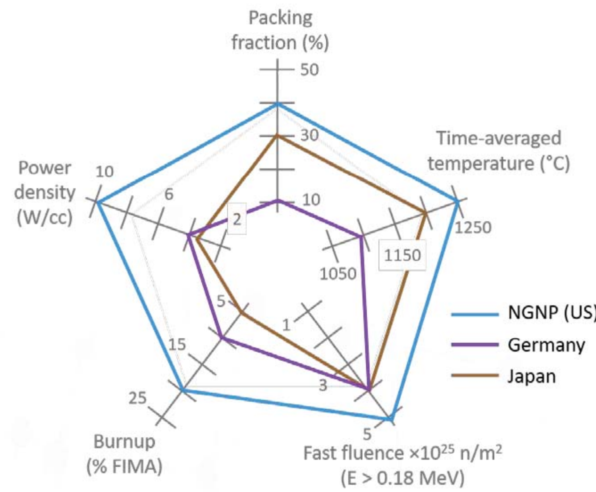
Cylindrical fuel compacts



Prismatic graphite blocks

Emerging Reactor Designs Requiring TRISO Fuel

- Molten-salt-cooled reactors (FHR)
  - Most irradiation conditions are within the fuel performance envelope explored in the US AGR program, with some exceptions, e.g.:
    - Power density may be higher
    - Irradiation temperature may be lower
  - No data on TRISO performance in salt coolant

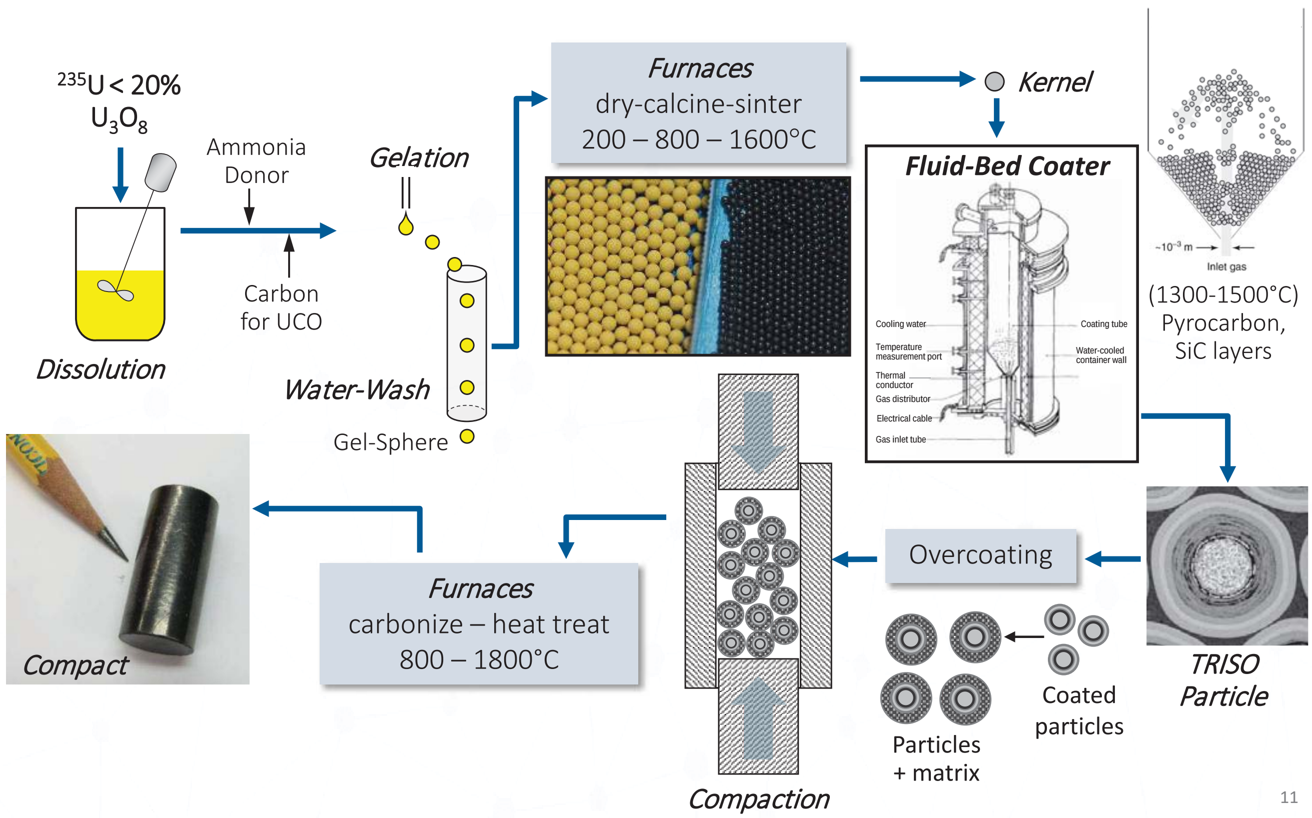


- Microreactors
  - Limited analyses on conceptual designs suggest that irradiation and accident conditions are less severe than larger gas reactor designs

Outline

- TRISO fuel background and history
- Fuel fabrication and quality control
- Fuel irradiation performance
- Fuel accident performance
- Fuel performance and fission product transport modeling

TRISO Fuel Fabrication: Process Overview

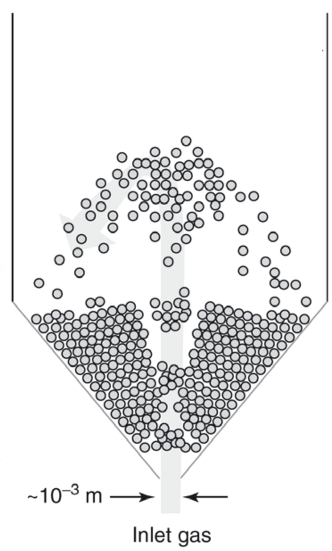


Coating Deposition

- Coatings are deposited onto kernels using a fluidized bed chemical vapor deposition furnace
- Coatings are applied using a continuous process
- Reactant gas mixture and temperature are controlled to obtain desired coating properties
- Coated particles are sorted by size and shape to remove under- and over-sized particles

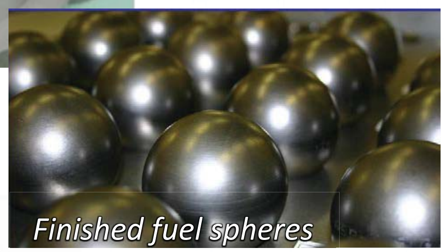
Coater converging section and gas nozzle

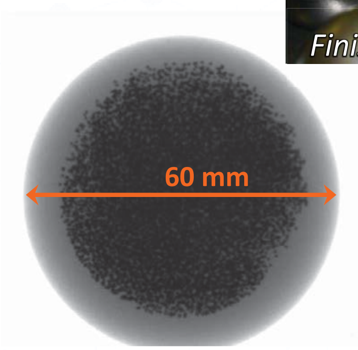
Industrial Scale 150 mm Coater (BWXT)



Fuel Elements

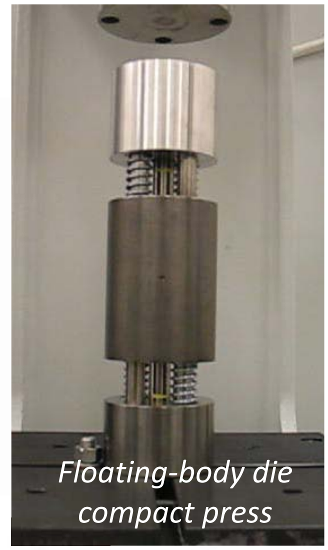
Spherical fuel elements

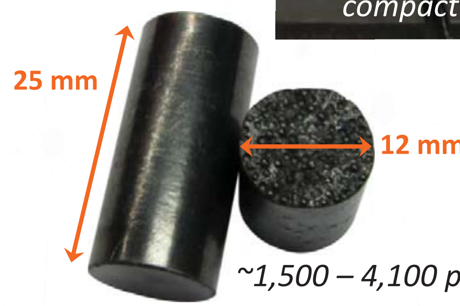




~9,000 – 18,000 particles

Cylindrical fuel elements

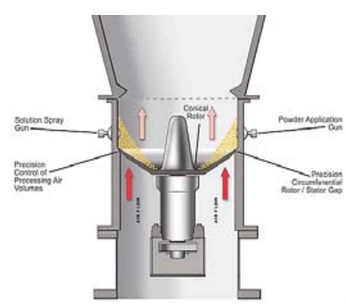




~1,500 – 4,100 particles

US AGR Program Fuel Fabrication Process Improvements

- Reduced human interactions in the process
  - Eliminated tabling with 3D sieving of coated particles
  - Improved matrix production (dry mixing and jet milling)
  - Improved overcoating with automated fluidized bed overcoater
  - Multicavity compacting press with automatic fill
- Kernel fabrication
  - Internal gelation to improve sphericity
  - Method of carbon addition modified to improve distribution of oxide and carbide phases
- Improved chemical vapor deposition process control
  - Argon dilution during SiC coating
  - Coater "chalice" and multiport nozzle to improve process yields (>95%)
  - Mass flow controllers to control gas flows during deposition of each coating layer
  - Improved MTS vaporizer (SiC layer deposition)



TRISO Fuel Quality Control

- Quality Control (QC) is the process used to verify that a product satisfies the design criteria
- QC for coated particle fuel includes:
  - Specifications on source materials, production processes, and process limits
  - Specifications on kernel, coating, and compact properties
  - Specifications on defect populations that may impact performance
- QC measurements of fuel properties are performed using statistical sampling
  - Specifications are met to a 95% minimum confidence level
  - Statistics often force the average fuel quality to be significantly better than the specifications
- IAEA Coordinated Research Program CRP-6
  - Fuel QA/QC round robin experimental study (also included HTGR fuel predictive code benchmarking exercises)

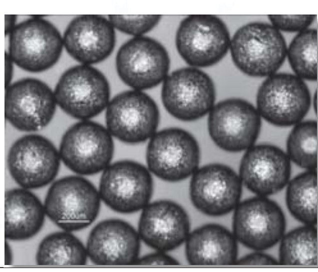
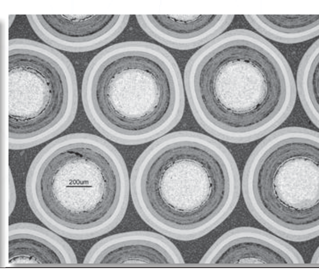
AGR Program Fuel Specifications for QC

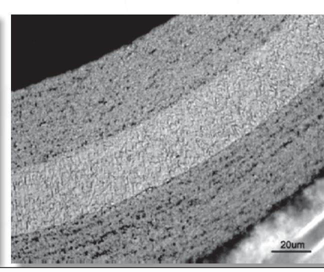
- Specified criteria on both process conditions and fuel properties
- Acceptance stages for kernel batches, kernel composites, particle batches, particle composites, and compacts
- Specified mean values and/or critical limits on the dispersion for variable properties, such as:

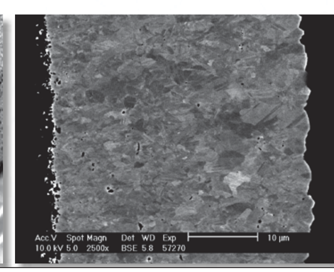
– Kernel diameter – Kernel stoichiometry – Layer thickness – Layer density	– Pyrocarbon anisotropy – Kernel and particle aspect ratio – SiC microstructure	– Compact dimensions – Compact U loading – Dispersed U fraction – Compact impurity content

- Specified maximum defect fractions for attribute properties, such as:

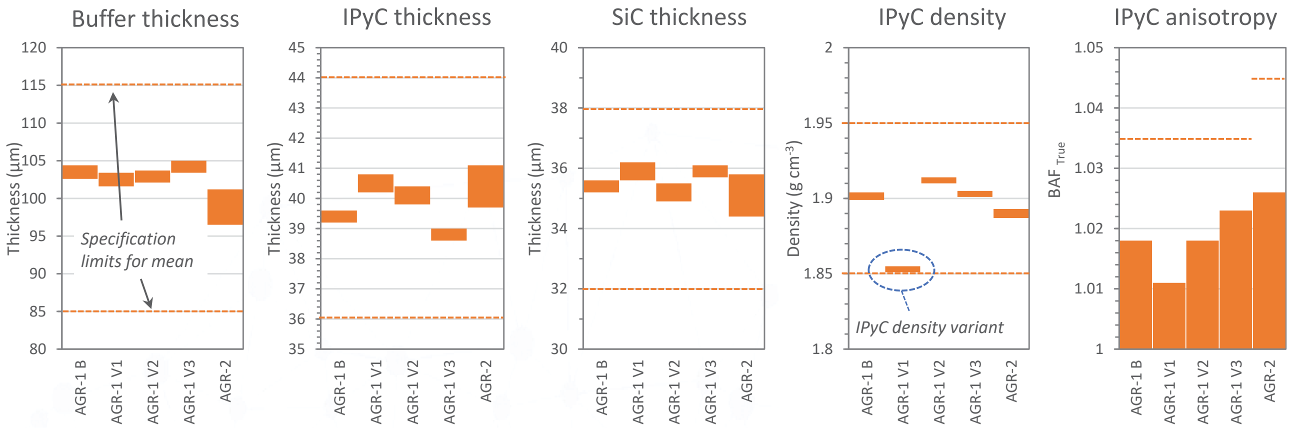
– SiC defects	– IPyC/OPyC defects	– Exposed kernel defects





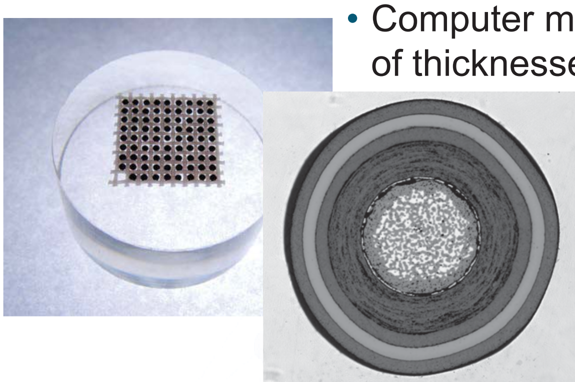
Selected AGR-1 and AGR-2 Fuel Property Means



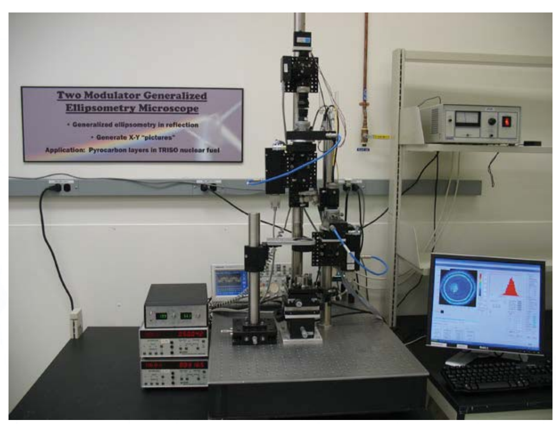
- Mean must be within the specification limits at 95% confidence
- AGR-1 and AGR-2 measured values typically lie well within the specification range
- Note that some specifications were changed for AGR-2, based on computational modeling results on fuel behavior

Improved Measurement Science

- Computer measurements of thicknesses



- Greatly improved PyC anisotropy measurements



- Improved density measurements using better density column fluids

Fuel Fabrication Summary

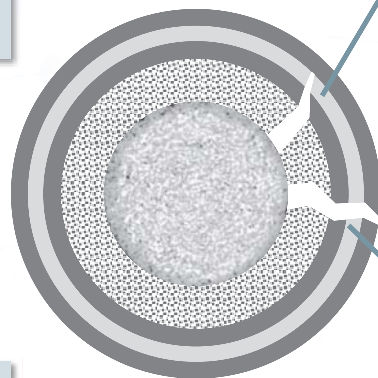
- TRISO fuel fabrication is a process that has matured over the last 50 years
- Statistical sampling is used to verify fuel quality
- Specifications are met to at least a 95% confidence level
- US AGR program has implemented numerous fuel fabrication process and characterization method improvements

Outline

- TRISO fuel background and history
- Fuel fabrication and quality control
- Fuel irradiation performance
- Fuel accident performance
- Fuel performance and fission product transport modeling

TRISO Fuel Performance

- Coating integrity
  - Layers remain intact to retain fission products



- SiC layer failure:
  - Breach in the SiC layer with at least one pyrocarbon layer intact
  - Release most condensable fission products but retain fission gas

- TRISO layer failure:
  - All three dense coating layers breached
  - Release of fission gas and condensable fission products

- Fission product retention
  - Coating integrity
  - Retention in kernel
  - Diffusive transport through layers
  - Matrix retention

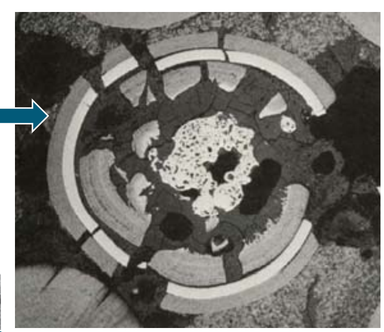
Fuel Failure Mechanisms

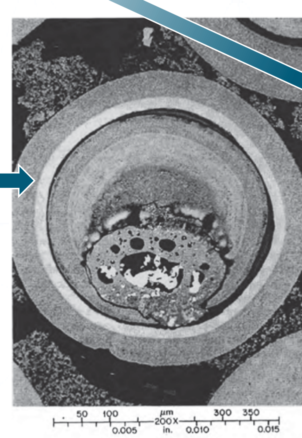
Mechanical

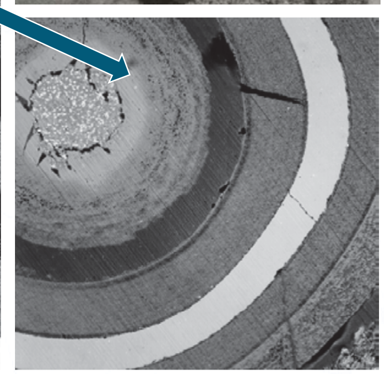
- Pressure vessel failure
- Irradiation-induced PyC failure leading to SiC cracking
- IPyC-SiC partial debonding

Thermochemical

- Kernel migration
- SiC thermal decomposition
- Fission product attack of SiC
- Corrosion of SiC by CO

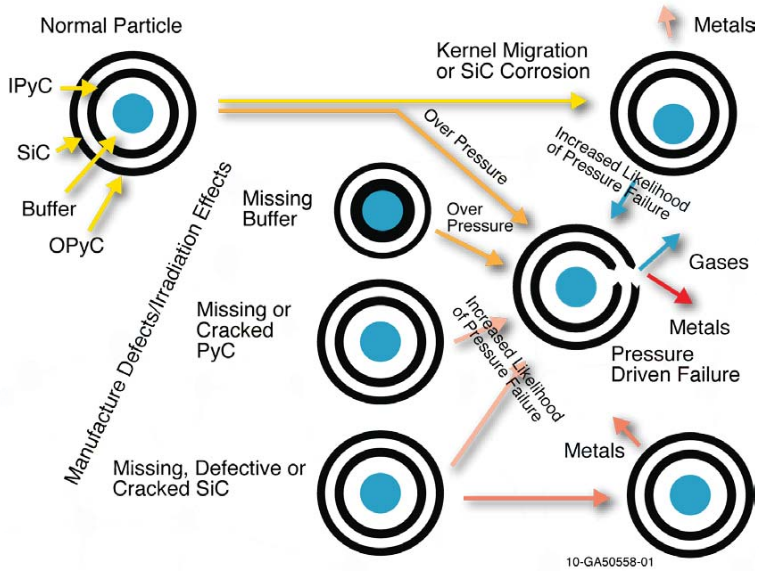






- Many of these mechanisms are precluded by improved particle design, improved manufactured fuel quality, and by operation of the fuel within its intended performance envelope

Fuel Failure Mechanisms



- SiC corrosion by CO(g) (in UO_2 fuel) and fission products (in UO_2 and UCO fuel) is the primary cause of SiC layer failure observed in modern TRISO fuel
- High-quality fuel manufacture and limitations on irradiation conditions (performance envelope) reduce failure fractions to acceptable limits

Irradiation Testing

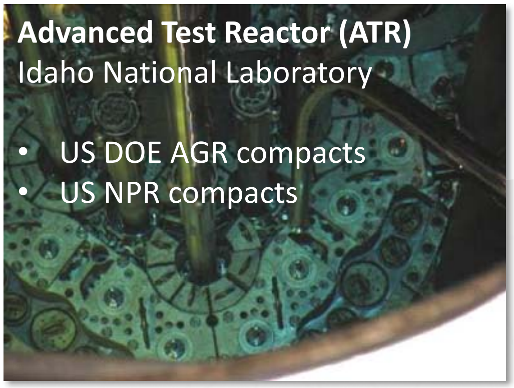
Prototype modular HTGRs

- Prototypical conditions (neutron spectrum and flux, burnup accumulation rate)
- Long duration
- Difficult online measurement of fuel performance
- Less certainty on fuel temperature

Materials Test Reactors (MTRs)

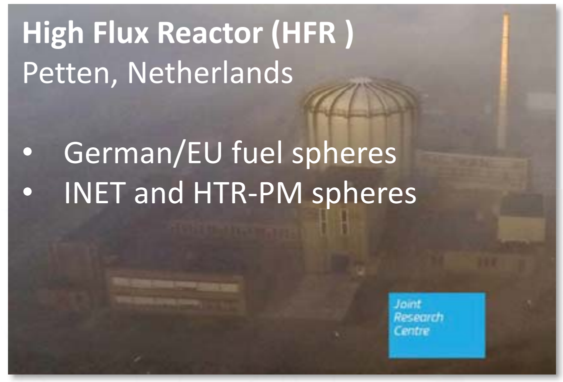
- Accelerated irradiation times
- Measurement and control of fuel temperature
- Real-time measurement of fission product release
- Conditions may differ somewhat from HTGRs (neutron spectrum and flux, burnup accumulation rate)

Irradiation Testing of TRISO Fuel in MTRs



Advanced Test Reactor (ATR)
Idaho National Laboratory

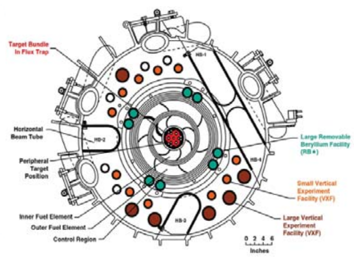
- US DOE AGR compacts
- US NPR compacts



High Flux Reactor (HFR)
Petten, Netherlands

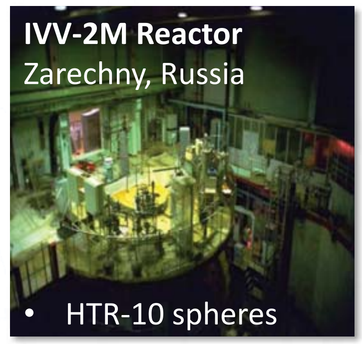
- German/EU fuel spheres
- INET and HTR-PM spheres

High Flux Isotope Reactor (HFIR)
Oak Ridge National Laboratory



- US DOE TRISO fuels

Many other MTRs have been used to test TRISO fuel



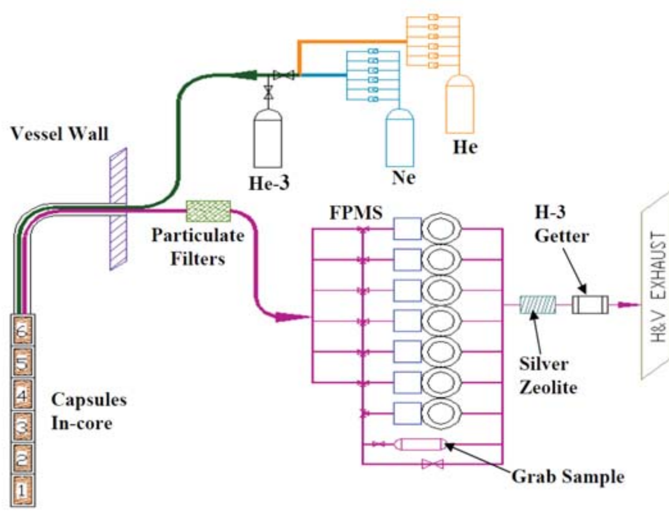
IVV-2M Reactor
Zarechny, Russia

- HTR-10 spheres

Irradiation Performance: R/B

- It is critical to have reliable measurement of fission gas release during irradiation (real-time or intermittent through gas capture and analysis)
- Fission gas release rate to birth rate ratio (***R/B***) is the main metric of fuel performance during irradiation
- Sweep gas (He + Ne) injected into the capsules controls capsule temperature and carries fission gas to the FGMS
- Gamma spectrometers quantify short-lived Kr and Xe isotopes

Kr-85m	Xe-131m
Kr-87	Xe-133
Kr-88	Xe-135
Kr-89	Xe-135m
Kr-90	Xe-137
	Xe-138
	Xe-139

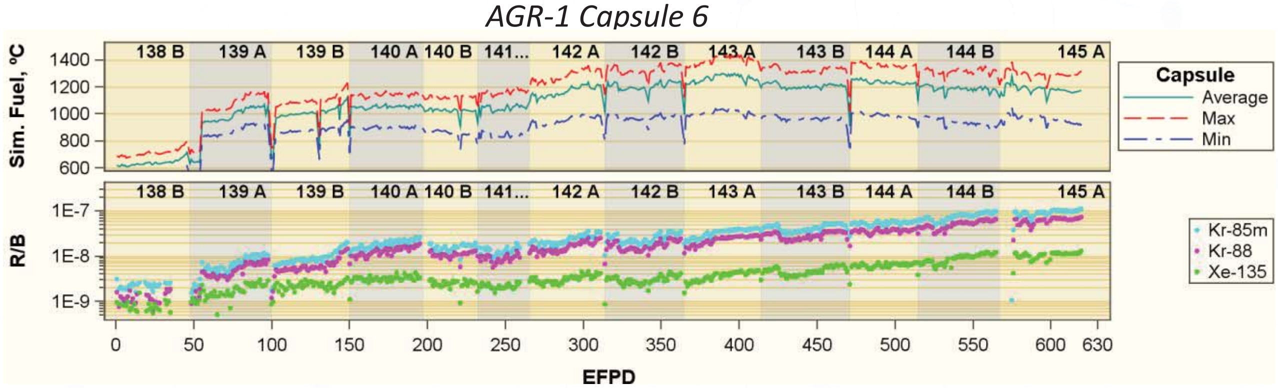


AGR-1 Fission Gas Monitoring System (FGMS)

Irradiation Performance: R/B (cont'd)

- Sources of fission gas release:
  - Uranium contamination outside of intact SiC layers
  - Exposed kernel defects (as-fabricated)
  - Exposed kernels from in-service coating layer failure
- R/B provides information on the extent of coating failures during irradiation
- Release rate is a function of temperature and half-life

AGR-1 Capsule 6



→ ***Data indicate zero as-fabricated exposed kernels or in-pile TRISO failures in this capsule***

Recent TRISO Fuel Irradiation Tests (2000 – Present)

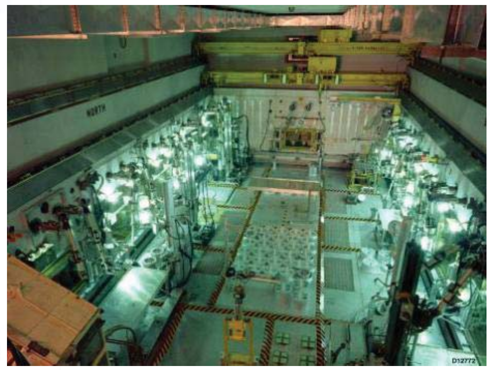
Irradiation test	Location	Fuel type	Spheres or compacts (particles)	Completed	Burnup (%FIMA)	Temperature (°C)[a]	EOL ^{85m}Kr R/B
US DOE/AGR (cylindrical compacts)							
AGR-1	ATR	UCO	72 (298,000)	Nov 2009	11.3 – 19.6	1069 – 1197	0.1 – 1×10^{-7}
AGR-2	ATR	UCO	36 (114,000)	Oct 2013	7.3 – 13.2	1080 – 1360	~10^{-6} [b]
		UO_2	12 (18,500)		9.0 – 10.7	1072 – 1105	10^{-7} [b]
AGR-5/6/7	ATR	UCO	194 (570,000)	In progress	7.4 – 18.6	~600 – 1500	*TBD*
Germany/EU (spheres)							
HFR-EU1	HFR	UO_2	3 (28,700)	Feb 2010	13.5 – 14.3	~950[c]	2.5×10^{-7}
HFR-EU1bis	HFR	UO_2	5 (47,800)	Oct 2005	~11	~1250	4×10^{-6}
China (spheres)							
HTR-10/ IVV-2M	IVV-2M	UO_2	4 (33,200)	Feb 2003	11.6 – 13.1	1000 ±50	0.1 – 8×10^{-5}
HFR-EU1	HFR	UO_2	2 (16,600)	Feb 2010	9.3, 11.6	900 – 940[c]	7×10^{-8}
HFR-PM	HFR	UO_2	5 (60,000)	Dec 2014	10.1 – 12.7	1050 ±50	~3×10^{-9}

[a] Time-average peak temperatures (except where noted)
[b] R/B values through the first three irradiation cycles
[c] Sphere surface temperatures

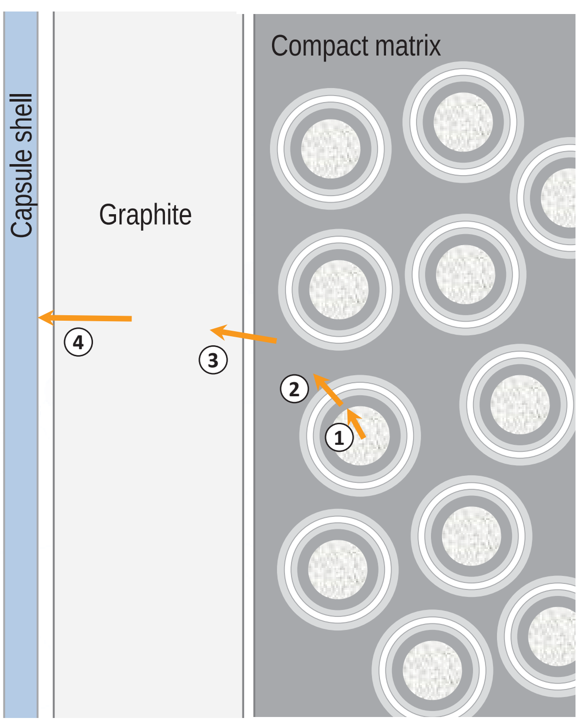
Excellent performance within intended fuel performance envelope

TRISO Fuel Post-Irradiation Examination and High-Temperature Accident Safety Testing

- Main objectives:
  - Measure fission product retention during irradiation
  - Measure fission product retention during high temperature post-irradiation heating
  - Examine kernel and coating microstructures to understand irradiation-induced changes and the impact on fuel performance
- Both conventional and specialized equipment used for TRISO fuel examinations



In-Pile Fission Product Release Evaluation



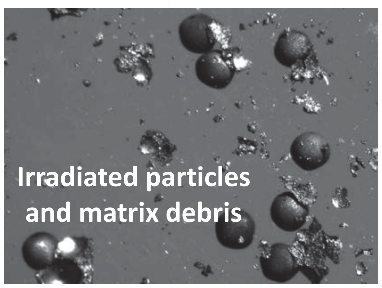
1. Release from kernel to coating layers
2. Release from coating layers to compact matrix
3. Release from compact matrix to structural graphite
4. Release from structural graphite to capsule shell (or reactor vessel)

Look for fission products:

- In fuel compacts
- On capsule components
- In compact matrix
- In individual particles

Compact Deconsolidation-Leach-Burn-Leach Analysis

Deconsolidation hardware



Disintegrate matrix and liberate loose particles — Electrolytic deconsolidation

Quantify isotope inventories — Nitric acid leach of particles and matrix debris (X2) → Analyze leachate for FPs and actinides

Oxidize carbon (matrix and OPyC layers) — Air oxidation ("burn") of particles and debris

Quantify isotope inventories — Nitric acid leach of remaining material (X2) → Analyze leachate for FPs and actinides

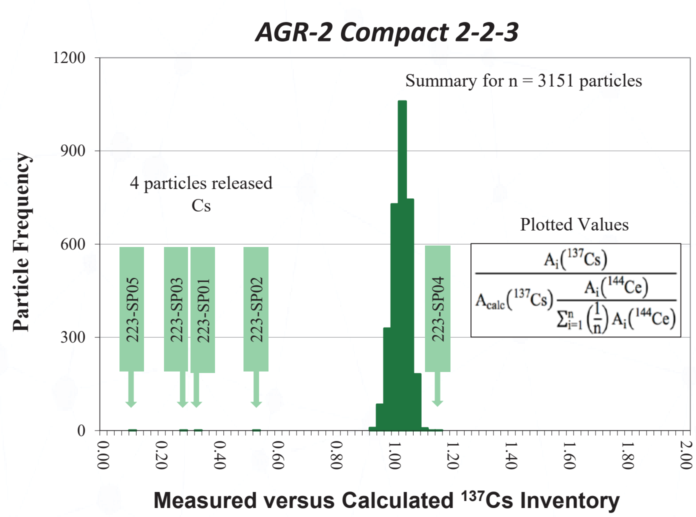
- **Process provides inventory of FPs and actinides in matrix outside of intact SiC**

Irradiated Particle Gamma Counting

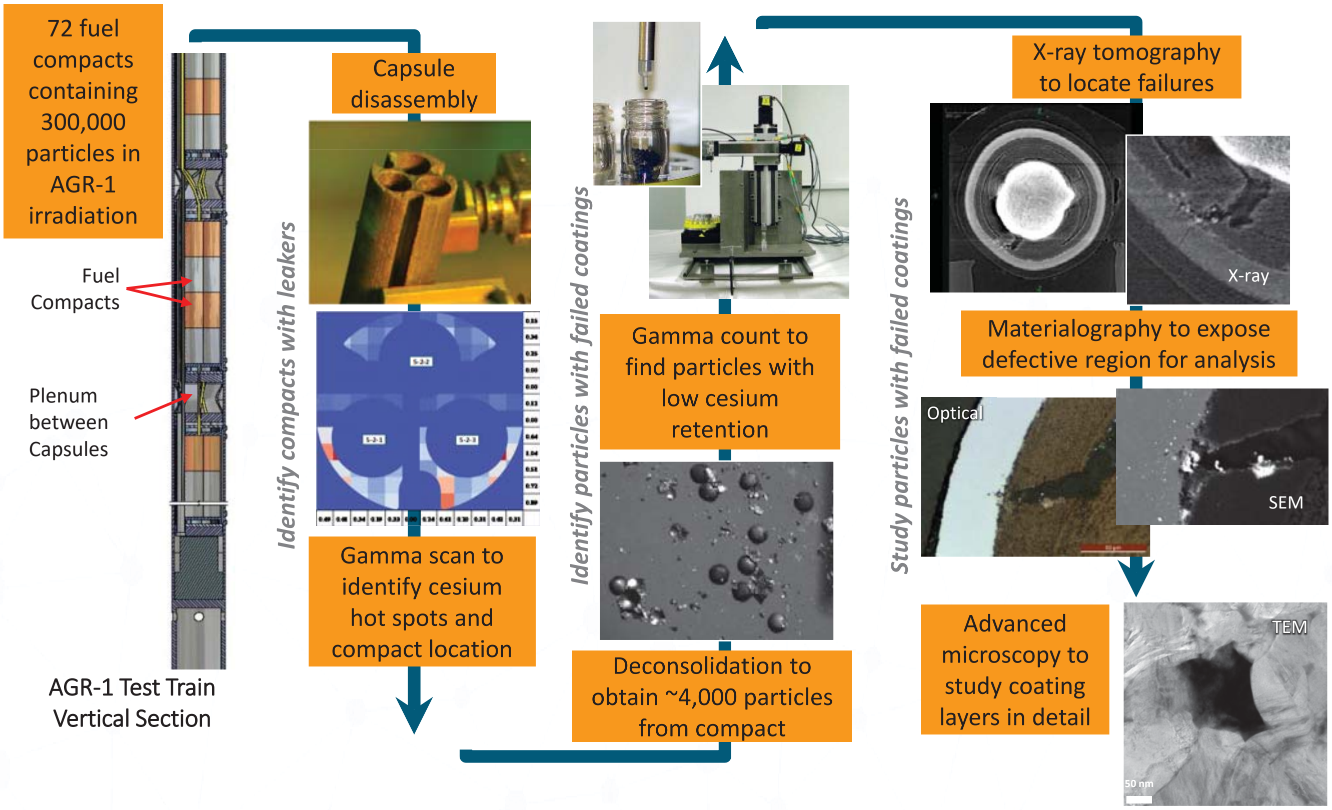
- Gamma count individual particles to quantify FP inventory (Ag-110m, Cs-134, Cs-137, Eu-154, Ce-144)
- Identify particles with abnormal inventory

ORNL Irradiated Microsphere Gamma Analyzer (IMGA)

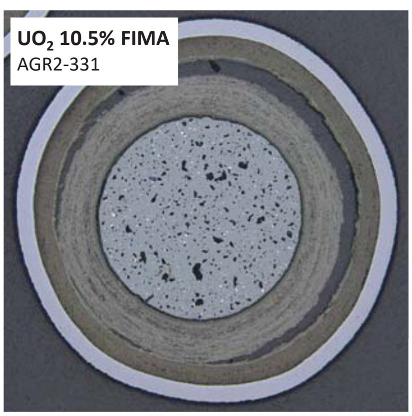
- Low Cs inventory indicates SiC failure and Cs release



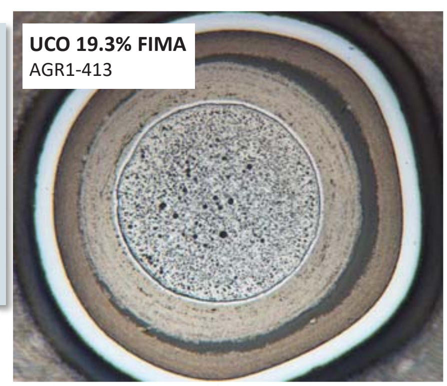
Studying failed particles greatly improves understanding of fuel performance

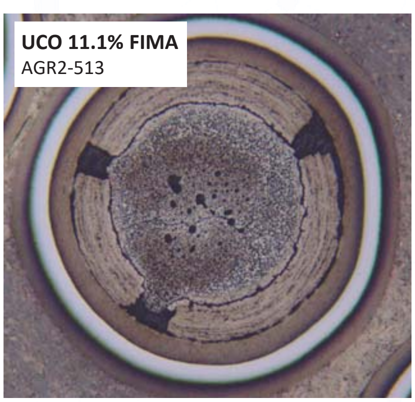


Kernel and Coating Behavior During Irradiation: AGR Particles



- Kernel swelling and pore formation
- Buffer densification and volume reduction
- Separation of buffer and IPyC layers



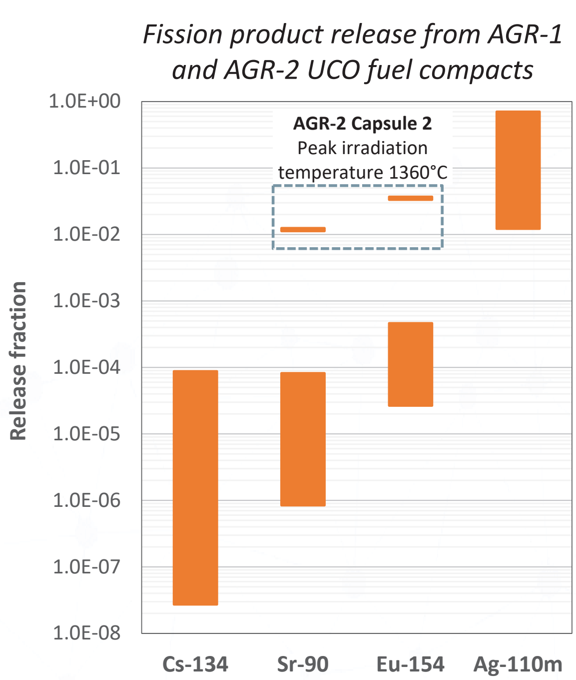


- Buffer fracture relatively common in UCO fuel particles
- Kernel can swell into gap
- Dependent on irradiation temperature and fast neutron fluence
- When buffer separates from IPyC, buffer fracture appears to have no detrimental effect on dense coating layers

Fission Product Behavior

Element	Behavior in TRISO Fuel
Kr, Xe, I	• Retained by intact PyC or SiC layers • Release is from uranium contamination and exposed kernels • Kr and Xe are key indicator of failed TRISO layers
Cs	• Retained by SiC but released through intact PyC • Key indicator of failed SiC
Sr	• Moderate retention in the fuel kernel • Modest release through intact coatings (T > 1100°C); significantly higher release for very high irradiation temperatures • Some retention in the compact matrix
Eu	• Similar to Sr, although evidence indicates slightly higher releases
Ag	• Significant release through intact SiC (T > 1100°C) • Relatively low retention in compact matrix

Fission Product Release from Fuel Compacts: AGR-1 and AGR-2 Examples



- Cs release is very low with intact SiC; higher releases are associated with a limited number of particles with failed SiC
- Sr and Eu can exhibit modest release; release is much higher with high in-pile temperatures (AGR-2 Capsule 2 time-average peak temperatures 1360°C)
- High Ag release

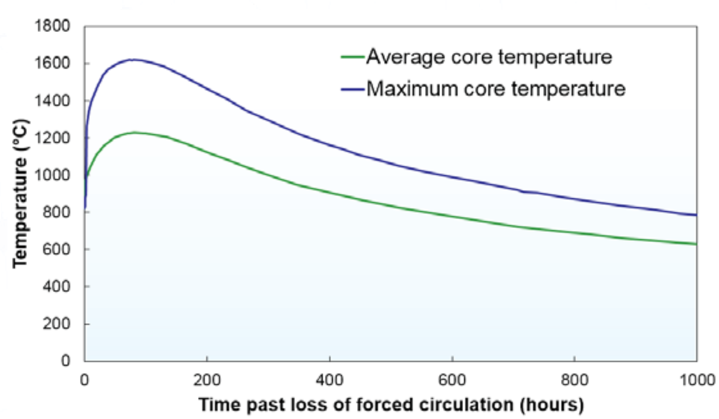
Outline

- TRISO fuel background and history
- Fuel fabrication and quality control
- Fuel irradiation performance
- Fuel accident performance
- Fuel performance and fission product transport modeling

HTGR Accident Safety Testing of TRISO Fuel

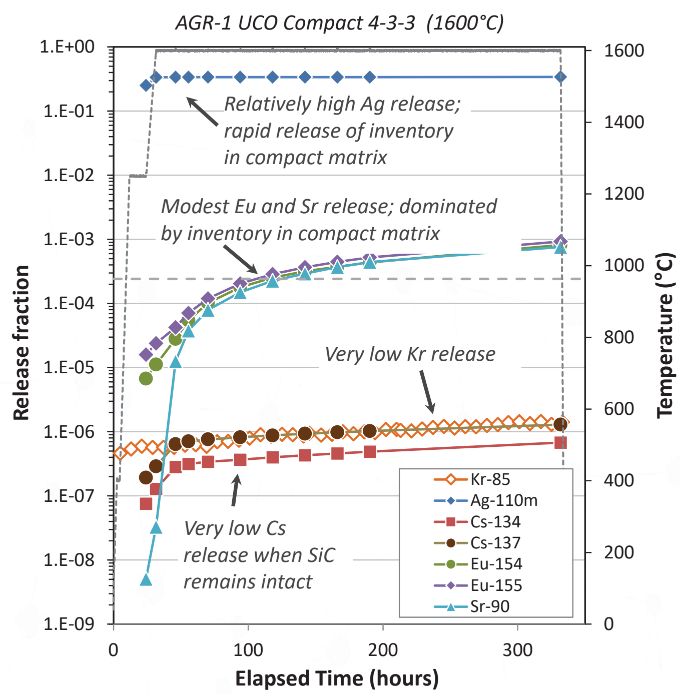
- Temperature transients are relatively slow (days)
- Peak fuel temperatures are limited to ~1600°C in modular HTGR designs
- Fuel particles are designed to withstand accident conditions while still retaining key safety-significant fission products
- Total duration at peak temperatures is tens of hours, and only a small fraction of the fuel in the core experiences temperatures near the peak.

- Assess fuel performance by post-irradiation heating tests while measuring fission product release at 1600—1800°C

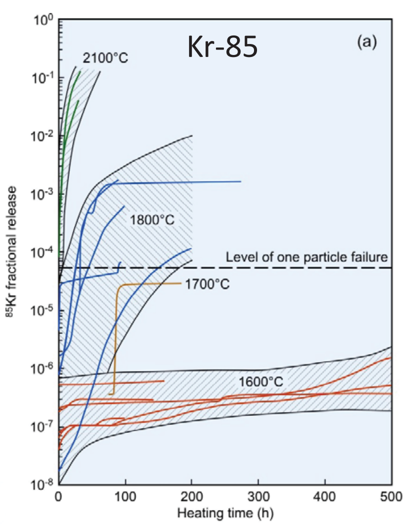


AGR-1 and AGR-2 Safety Test Performance

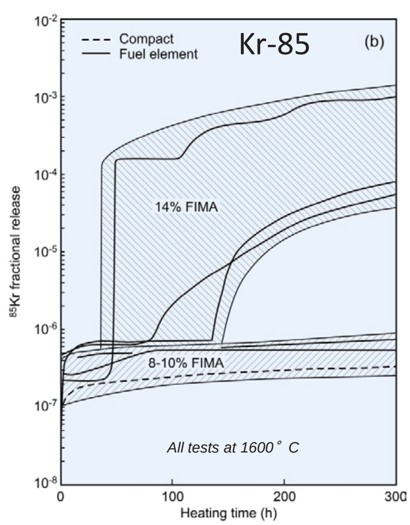
- **Low Cs release** (dependent on intact SiC)
- **Low Kr release**
- **Modest Sr and Eu release** (influenced by irradiation temperature)
- **High Ag release** (dominated by in-pile release from particles)
- **Excellent UCO performance up to 1800°C**
- **Low coating failure fractions** (UCO)
- **UO_2 demonstrates much higher incidence of SiC failure due to CO attack**



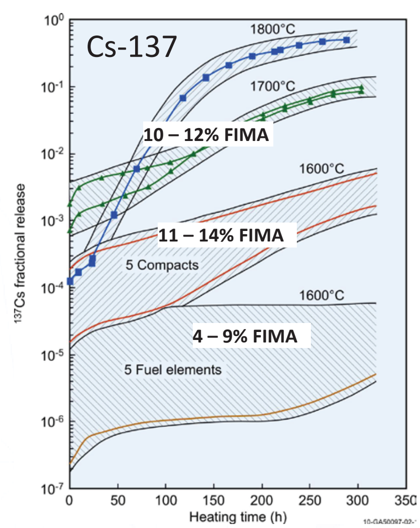
Safety Test Data: German UO_2 Results



- No TRISO failures at 1600°C
- TRISO failures occur after short periods at 1800°C



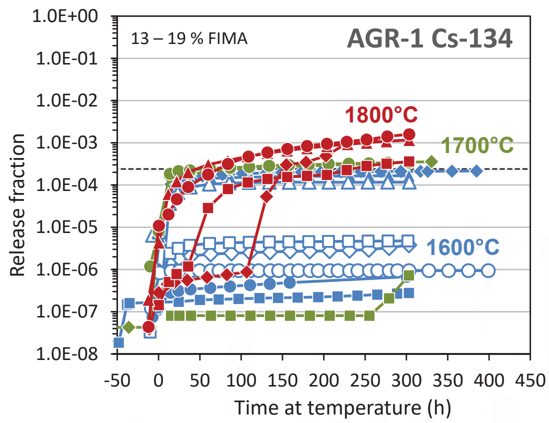
- No TRISO failures at 1600°C with burnup ≤10%
- TRISO failures occur at 1600°C with burnups ~14%

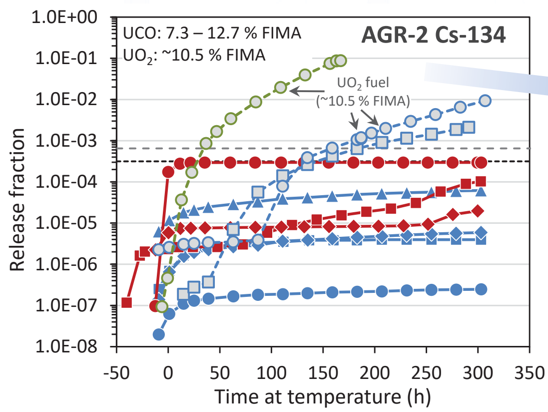


- At 1600°C and burnup <10% FIMA, Cs release remains relatively low
- Increasing burnup and temperature increases SiC layer degradation and Cs release

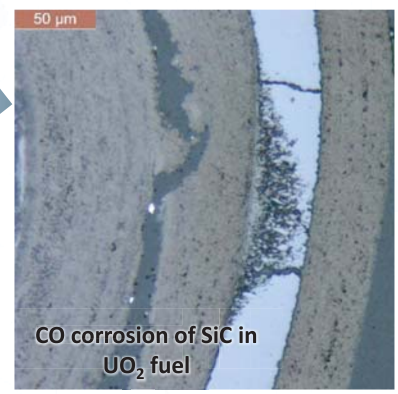
D.A. Petti et al., TRISO-Coated Particle Fuel Performance, in Konings R.J.M.,(ed.) Comprehensive Nuclear Materials (2012), vol. 3, pp. 151-213 Amsterdam: Elsevier.

Cesium Release Results: AGR Program Safety Testing



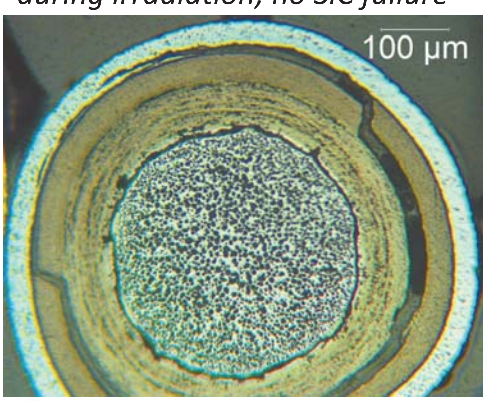


- **UCO fuel**: relatively low Cs release; release >10^{-4} results from discrete SiC layer failure in 1 or more particles
- **UO_2 fuel**: higher Cs release compared to UCO; driven by CO attack on the SiC layer causing more widespread SiC failure

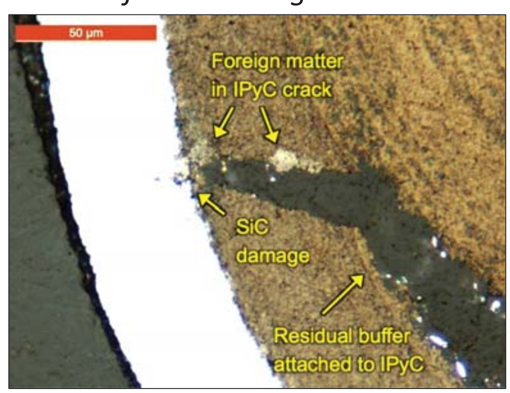


AGR UCO Particle SiC Failure

IPyC cracking and SiC separation during irradiation; no SiC failure

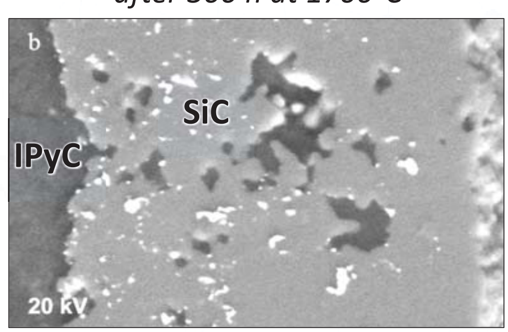


SiC failure during irradiation



- Buffer densification in conjunction with strong buffer-IPyC bonding can lead to IPyC cracking and separation from SiC layer
- Allows localized attack of SiC layer by fission products (especially Pd)
- Pd attack can eventually result in loss of FP retention by SiC layer
- Degradation is worse at higher safety test temperatures

SiC degradation and failure after 300 h at 1700°C



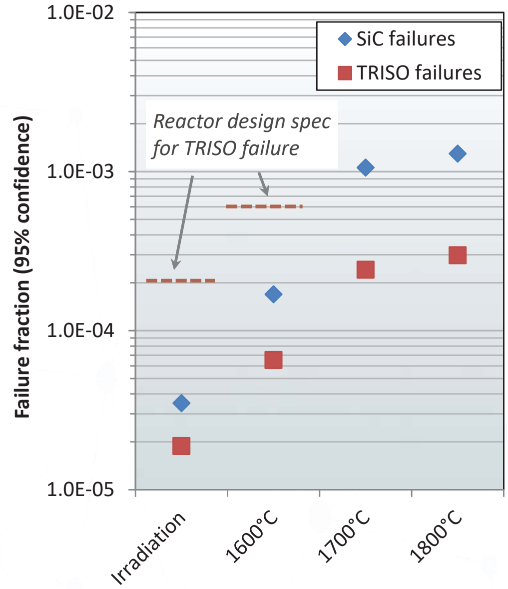
Fuel Design Safety Approach

Specifications for particle defects and failure fractions

Parameter	NGNP – 750°C Core Outlet Temperature	
	"Maximum Expected"	"Design"
As-Manufactured Fuel Quality		
HM contamination	≤ 1.0×10^{-5}	≤ 2.0×10^{-5}
Defective SiC	≤ 5.0×10^{-5}	≤ 1.0×10^{-4}
In-Service TRISO Failure		
Normal operation	≤5.0×10^{-5}	≤2.0×10^{-4}
Accidents	≤1.5×10^{-4}	≤6.0×10^{-4}

- Establish specifications for as-manufactured contamination levels and particle defects that can lead to fission product release
- Verify fuel quality with QC measurements
- Demonstrate failure fraction specifications are met during fuel qualification irradiation and safety testing

Experimental coating failure fractions for AGR-1 + AGR-2 (upper limit at 95% confidence)



AGR-1 and -2 TRISO failure fractions meet historic design specifications with ~10X margin

Core Oxidation

- Accident scenarios in gas-cooled reactors can include air or steam ingress into the core
- Specific conditions should be defined to the extent possible through models (temperatures, durations, oxidant partial pressure)
- Core behavior under these conditions should be evaluated
  - Graphite and matrix oxidation
  - Fission product volatilization from matrix/graphite and exposed kernels
  - Coated particle integrity
- Graphite oxidation data is available in literature
- Limited data on matrix oxidation is available from previous tests
- US AGR program is performing dedicated testing to obtain necessary data:
  - Matrix oxidation tests
  - Irradiated fuel heating tests in air and steam environments (starting ~2020)

Fuel Performance Summary

- There is an extensive database of TRISO irradiation testing in MTRs
  - Historic testing in the US, German program testing, and others
  - Recent demonstrations include EU tests (archived German fuel), HTR-PM fuel, and US AGR program
- Modern TRISO fuel exhibits very low R/B values during irradiation (low coating failures)
- TRISO fuel FP release behavior is well-characterized
- Extensive accident testing database
  - Fuel withstands 300 h at temperatures of 1600°C and above with low failure rates
- Observed failure fractions are well below historic reactor design specs

Outline

- TRISO fuel background and history
- Fuel fabrication and quality control
- Fuel irradiation performance
- Fuel accident performance
- Fuel performance and fission product transport modeling

Fuel Performance and Fission Product Transport Modeling

- Predict coating behavior as a function of particle properties and irradiation conditions → *Predict coating failure fractions*
- Predict fission product release
- Optimize particle design
- Help establish fuel product specifications
- Numerous codes developed in various countries dating to the 1960s

PARticle FUel ModEl
PARFUME
AGR program fuel performance modeling and analysis code

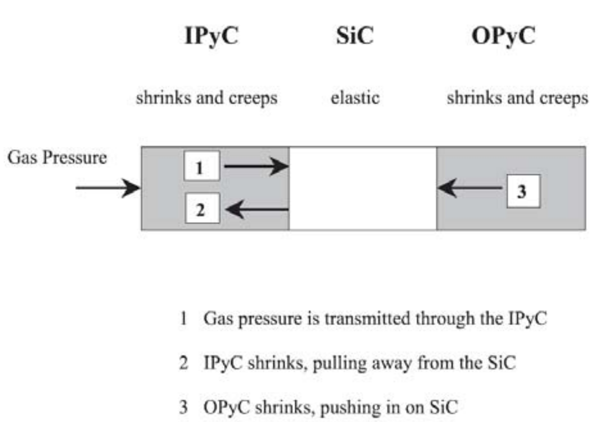
Mechanistic code
Thermal, mechanical, physico-chemical behavior of TRISO fuel particles

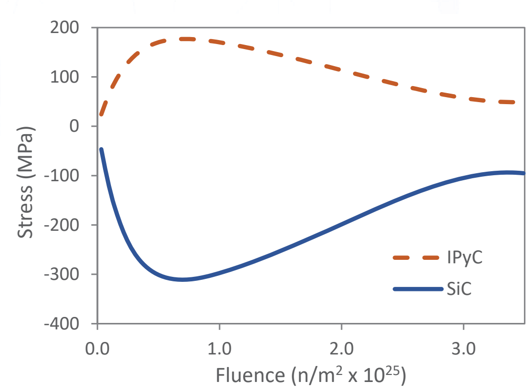
Probability of particle failure
Fission product fractional release

Coating Stress Calculations and Particle Failure Analysis

- Key inputs:
  - Fuel temperature, burnup, fast neutron fluence
  - PyC irradiation-induced creep and strain
  - SiC tensile strength and Weibull modulus
  - (Sensitivity studies indicate that many properties have little effect on particle failure)
- Particle failure probability based on Weibull statistics

$$P_f = 1 - e^{-\int_V \left(\frac{\sigma}{\sigma_o}\right)^m dV}$$

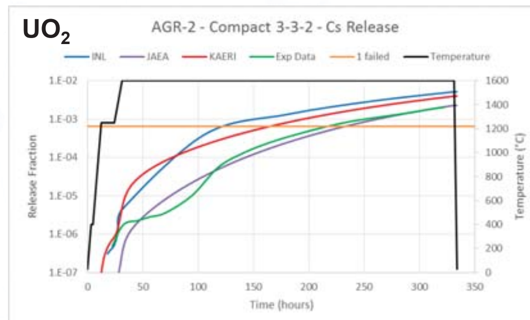




Stress histories at inner radii of the IPyC and SiC layers for an uncracked particle.

Fission Product Transport Modeling

- Fission product transport includes:
  - Release from failed particles
  - Release from uranium contamination in the compact
  - Diffusive release through intact coatings
- Requires FP diffusivities in:
  - Kernel
  - PyC
  - SiC
- Historic diffusivities come from UO_2 fuel fission product release observations
- Current models tend to overpredict fission product release by a significant margin



Results of computational modeling code benchmark of fission product release during high-temperature accident tests

(B. Collin et al., *Generation IV Benchmarking of TRISO Fuel Performance Models under Accident Conditions: Final Report, DRAFT*)

Summary

- TRISO fuel has a history spanning over 50 years
- High quality fuel can be fabricated to meet product specifications
- TRISO fuel has excellent performance during normal operation and accidents
- Fuel performance models predict behavior and tend to be conservative with respect to FP release

Suggested Reading

General TRISO Fuel

- 2010 HTGR Technology Course for the Nuclear Regulatory Commission
- P.A. Demkowicz et al., Coated particle fuel: Historical perspectives and current progress, J. Nucl. Mater. 515 (2019) 434-450
- M.J. Kania, H. Nabielek, H. Nickel, Coated Particle Fuels for High-Temperature Reactors, in Materials Science and Technology, Wiley 2015.
- D.A. Petti et al., TRISO-Coated Particle Fuel Performance, in Konings R.J.M.,(ed.) Comprehensive Nuclear Materials (2012), vol. 3, pp. 151-213 Amsterdam: Elsevier.
- High Temperature Gas Cooled Reactor Fuels and Materials, IAEA, TECDOC-1645 (2010).
- K. Verfondern, H. Nabielek, J.M. Kendall, Coated particle fuel for high temperature gas cooled reactors, Nucl. Eng. Tech. 39 (2007) 603-616.
- D.A. Petti et al., Key differences in the fabrication, irradiation and high temperature accident testing of US and German TRISO-coated particle fuel, and their implications on fuel performance, Nucl. Eng. Des. 222 (2003) 281-297.
- Fuel performance and fission product behavior in gas cooled reactors, IAEA, TECDOC-978 (1997).

AGR Program Results

- P.A. Demkowicz et al., "Key results from irradiation and post-irradiation examination of AGR-1 UCO TRISO fuel," Nucl. Eng. and Des. 329 (2018) 102–109.
- P.A. Demkowicz et al., AGR-1 Post Irradiation Examination Final Report, INL/EXT-15-36407, Idaho National Laboratory, 2015.
- J.D. Hunn et al., "Post-Irradiation Examination and Safety Testing of US AGR-2 Irradiation Test Compacts," Paper 10 in Proceedings of the 9th International Topical Meeting on High Temperature Reactor Technology (HTR-2018), Warsaw, Poland, October 8–10, 2018. Available at https://www.osti.gov/biblio/1489588
- J.D. Hunn et al., "Initial Examination of Fuel Compacts and TRISO Particles from the US AGR-2 Irradiation Test," Nucl. Eng. and Des., 329 (2018) 89–101.

Suggested Reading (cont.)

HTR-PM Fuel

- C. Tang et al., Comparison of two irradiation testing results of HTR-10 fuel spheres, Nucl. Eng. Des. 251 (2012) 453-458.
- S. Knol et al., HTR-PM fuel pebble irradiation qualification in the high flux reactor in Petten, Nucl. Eng. Des. 329 (2018) 82-88.
- D. Freis et al., Burn-up Determination and Accident Testing of HTR-PM Fuel Elements Irradiated in the HFR Petten, Proceedings of the 9th International Topical Meeting on High Temperature Reactor Technology (HTR-2018), 8-10 Oct. 2018, Warsaw, Poland

Fuel Performance and Fission Product Transport Modeling

- J.J. Powers, B.D. Wirth, A review of TRISO fuel performance models, J. Nucl. Mater. 405 (2010) 74-82
- G.K. Miller et al., PARFUME Theory and Model Basis Report, INL/EXT-08-14497, September 2018
- W. F. Skerjanc, B. P. Collin, Assessment of Material Properties for TRISO Fuel Particles used in PARFUME, INL/EXT-18-44631, August 2018

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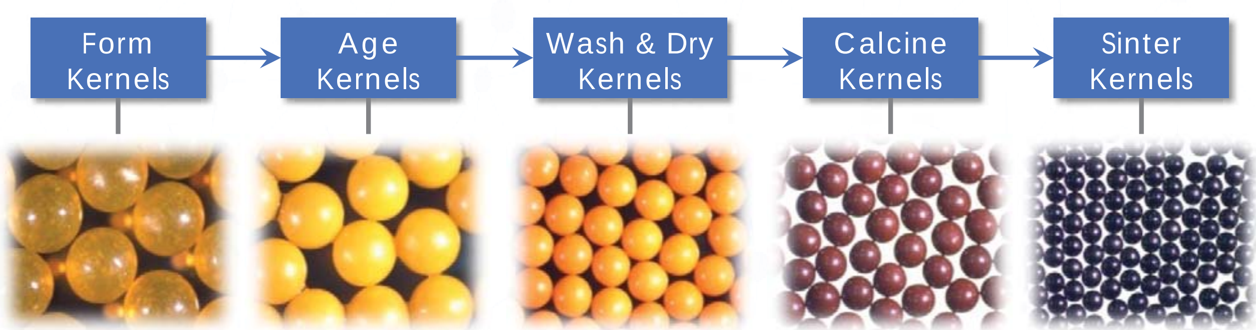
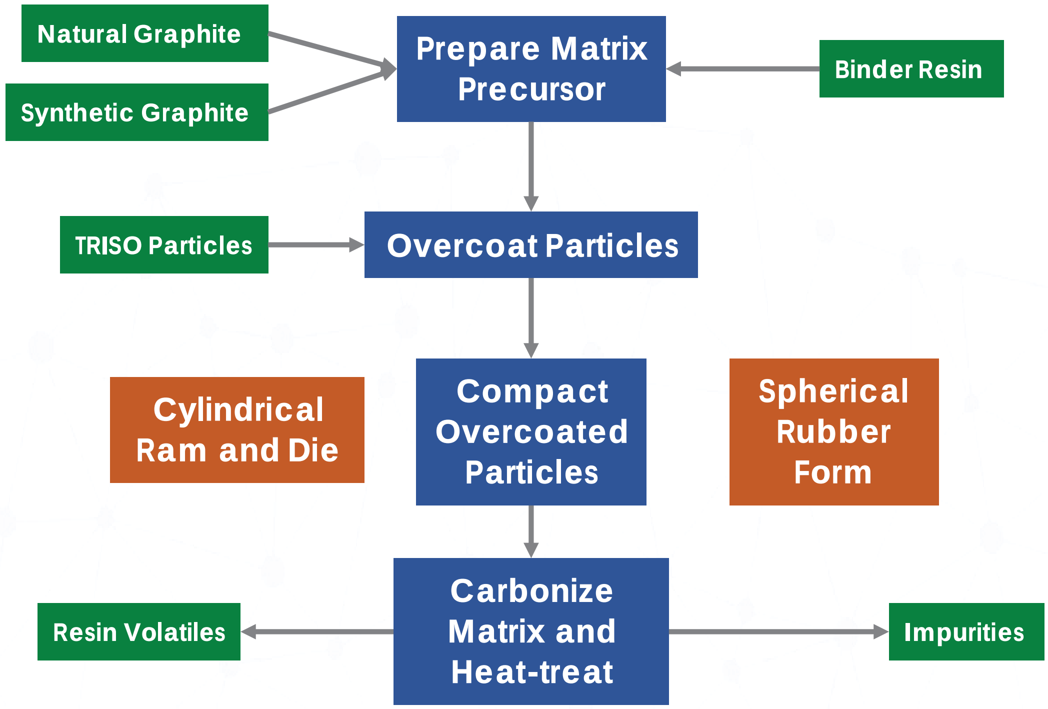
Kernel Fabrication

- Kernels are fabricated using a sol-gel process to form a spherical bead
- Dried spherical beads are heat treated to form the desired metal oxide and/or carbide phases and sinter the kernel



Fuel Compact/Sphere Fabrication



Modular High Temperature Gas-cooled Reactor: Safety Design Approach

Jim Kinsey

Modular High Temperature Gas-cooled Reactor: Safety Design Approach

Advanced Reactor Technologies

Idaho National Laboratory

Jim Kinsey

INL Regulatory Affairs

NRC HTGR Training July 16-17, 2019

Idaho National Laboratory

ART ADVANCED REACTOR TECHNOLOGIES

Modular HTGR Safety Design Objectives and Requirements

Deployment Objectives

- Flexibly co-locate with new industry users of nuclear energy
- Steam and electric cogeneration applications
- Direct process heat with temperature ranges from 700°C to 950°C

Enabling Requirements

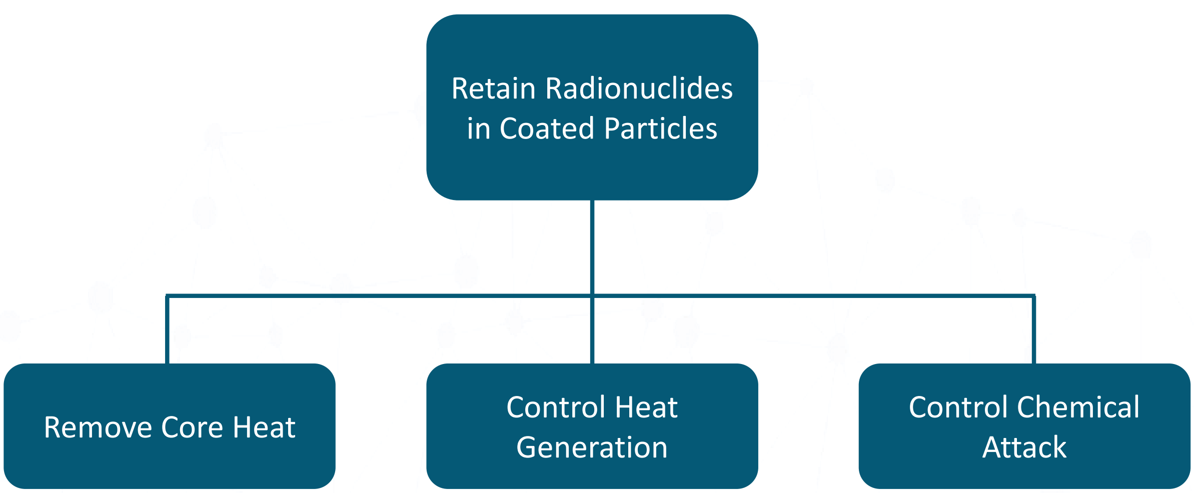
- Meet regulatory dose limits at the Exclusion Area Boundary (EAB)
  - 25 rem Total Effective Dose Equivalent (TEDE) for duration of the release from 10 CFR 50.34 (10 CFR 52.79) at Exclusion Area Boundary (EAB) for design basis accidents
  - EAB is estimated approximately 400 meters from the modular HTGR plant (to support co-location with industrial facilities)
- Meet safety goals for cumulative individual risk for normal and off-normal operation
- Design goal: meet EPA Protective Action Guides (PAGs) at EAB
  - 1 rem TEDE for sheltering
  - Design basis and beyond design basis events are considered
  - Realistically evaluated at the EAB
  - Emergency planning and protection

Modular HTGR Safety Design Approach

- Utilize inherent material properties as basis for safety
  - Helium coolant – neutronically transparent, chemically inert, low heat capacity, single phase
  - Ceramic coated (TRISO) particle fuel – high temperature capability, high radionuclide retention
  - Graphite moderator – high temperature stability, large heat capacity, long thermal response times
- Simple reactor design with inherent and passive safety features
  - Retain most radionuclides at source (i.e., within fuel)
  - Shape and size reactor to allow passive heat removal from reactor core using uninsulated reactor vessel
    - Heat is still removed if system is depressurized due to breach in reactor helium pressure boundary (HPB)
    - Heat is radiated from reactor vessel to RCCS panels
  - Large negative temperature coefficient supports intrinsic reactor shutdown
  - No reliance on AC-power to perform required safety functions
  - No reliance on operator intervention; insensitive to incorrect operator actions or inactions

Modular HTGR Safety Basis and Approach paper submitted Sept 2011 to NRC for information (INL/EXT-11-22708)

Radionuclide Retention within Modular HTGR Fuel Depends on Three Functions



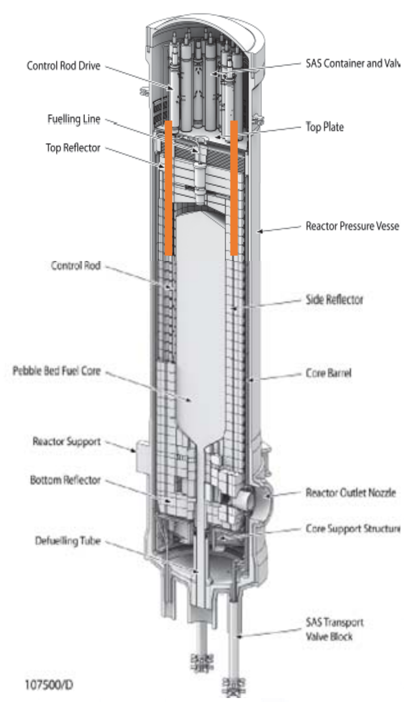
Control Heat Generation

Accomplished by Intrinsic Shutdown and Reliable Control Material Insertion

- Large negative temperature coefficient intrinsically shuts reactor down
- Two independent and diverse systems of reactivity control for reactor shutdown; drop by gravity on loss of power
  - Control rods
  - Reserve shutdown system
- Each system capable of maintaining subcriticality
- One system capable of maintaining cold shutdown during refueling
- Neutron control system measurement and alarms

Typical Reactivity Control

- Two independent, rod banks
- Articulated rods suspended from drives by chains to be lowered into the radial reflector
- Bypass flow cools the rods
- Rods may be partially inserted during power operation to provide Xe restart/load follow capability
- Prismatic – Shutdown rods can inserted into fuel blocks
- PBR – Small absorber spheres have been used in past designs (not in X-energy XE-100)
- Stronger negative fuel temperature feedback
  - HTGR: -7 pcm/K
  - PWR: -1 to -4 pcm/K



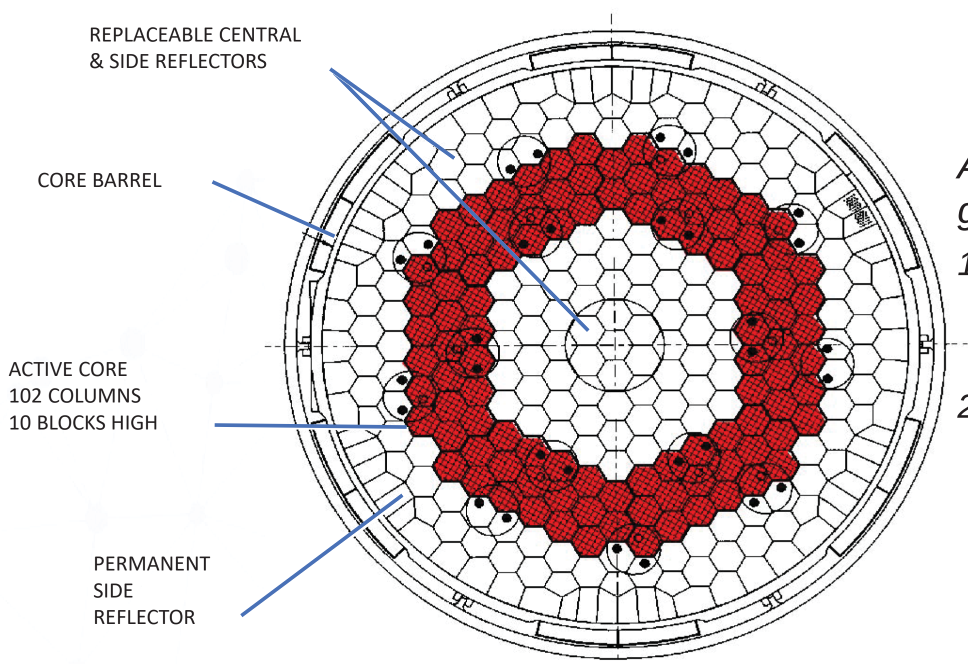
Both AVR and HTR-10 can be shut down without rods – circulators are stopped to affect a core heatup and Doppler shutdown.

Remove Residual Core Heat

Accomplished by Passive Design Safety Features

- Small thermal rating/low core power density
- Core geometry
  - Long, slender or annular cylindrical geometry
  - Heat removal by passive conduction and radiation
  - High heat capacity graphite
  - Slow heat up of massive graphite core
- Uninsulated reactor vessel
- Reactor Cavity Cooling System (RCCS)
  - Separate and distinct from reactor vessel system
  - Natural convective circulation of air or water during accident conditions
- Atmosphere is ultimate heat sink

Annular Core Optimizes Passive Heat Removal

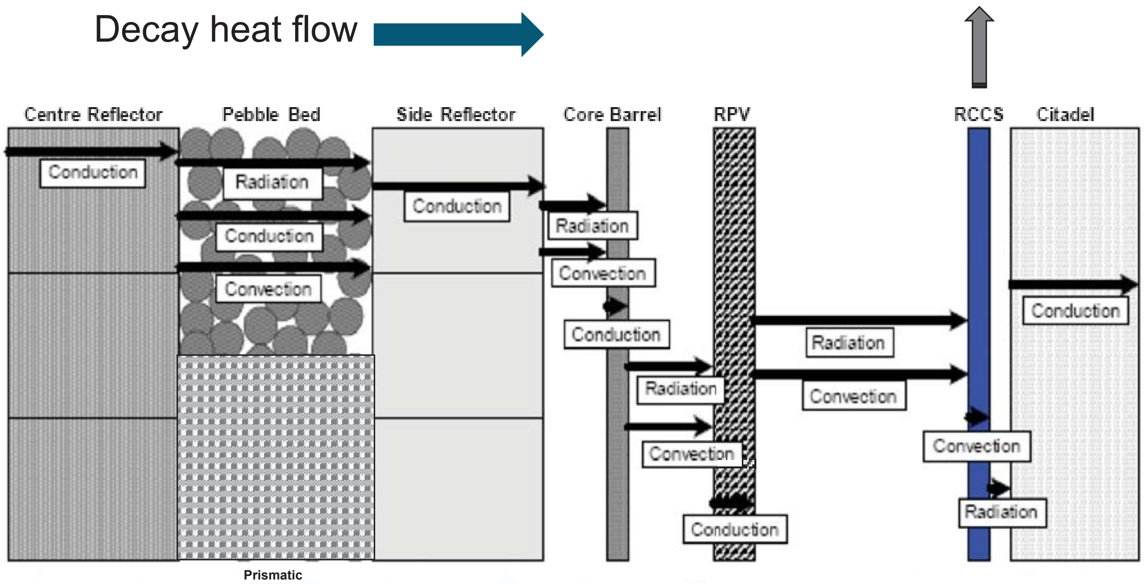


Annular core geometry:

1) Shortens heat conduction pathway

2) Enhances surface to volume ratio

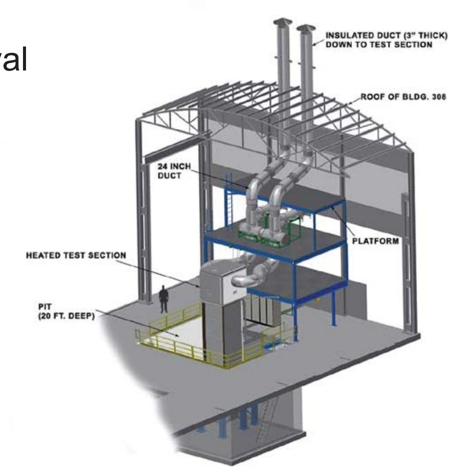
Passive Heat Transfer Path



Example: Annular Core Pebble Bed

Reactor Cavity Cooling System (RCCS)

- Typically safety-related in modular HTGR applications
- Consists of cooling panel structures that surround the reactor vessel
- Removes heat transmitted from vessel via radiation and convection
- Always operates to remove heat during both normal and off-normal operations
- All RCCS designs passively remove heat during all off-normal events via natural convection air or natural circulation water flow
- A simple and reliable means of residual heat removal
- Meets all requirements with ample margin and redundancy
- Natural convection Shutdown heat removal Test Facility (NSTF) at Argonne National Laboratory



Key RCCS Design Considerations

- RCCS maintains concrete cavity wall and reactor vessel temperatures
  - Concrete cavity temperatures are strongly related to RCCS performance
- RCCS operation is not typically required to protect fuel
- Heat removal rates are similar during normal operations and accident conditions
- RCCS is a simple system that functions passively when required during off-normal conditions
- Various air- or water-cooled RCCS configurations are possible
- Normal plant operation provides ongoing confirmation of RCCS system status

Control of Chemical Attack – Air

Assured by Passive Design Features and Inherent Characteristics

- Inert coolant (helium)
- High integrity nuclear grade pressure vessels make large breaks exceedingly unlikely
- Air ingress limited by core flow area and friction losses
- Reactor embedment and building vents close after venting, thereby limiting potential air in-leakage
- Graphite fuel form, fuel compact matrix, and ceramic coatings protect fuel particles
- Graphite exhibits slow oxidation rate (high purity nuclear grade graphite will not “burn”)

Control of Chemical Attack – Moisture

Assured by Passive Design Features and Inherent Characteristics

- Non-reacting coolant (helium)
- Limited sources of water in steam cycle plants
  - Moisture monitors
  - Steam generator isolation (does not require AC power)
  - Steam generator dump system
- Water-graphite reaction:
  - Endothermic
  - Requires temperatures > normal operation
  - Slow reaction rate
- Graphite fuel form, fuel compact matrix, and ceramic coatings protect fuel particles

Functional Radionuclide Containment

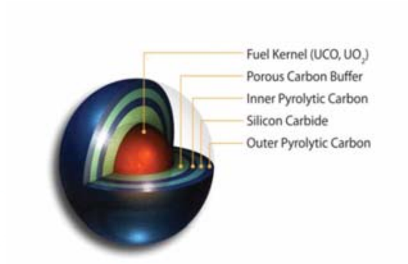
- Modular HTGRs employ “functional containment” for radionuclide control
- Eliminates need for “traditional” pressure retaining containment structure
- Functional containment is a collection of design choices that, when operated together, ensure that:
  - Radionuclides are retained within an independent multi-barrier system
  - Emphasis is on radionuclide retention at source (i.e., in the fuel)
  - NRC regulatory requirements (10 CFR 50.34/10 CFR 52.79) and plant design goals (PAGs) for release of radionuclides are met at the EAB
- See SECY-18-0096 and RG 1.232 for further information on functional containment performance criteria for non-LWRs

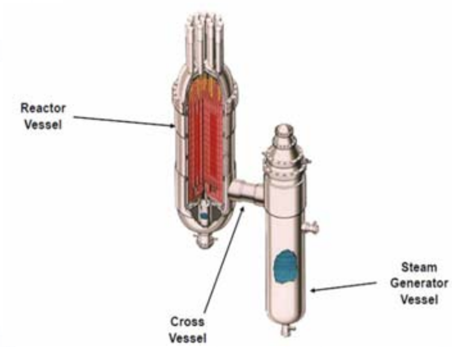
Modular HTGR Functional Containment

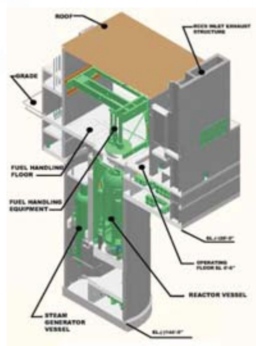
5 Radiological Release Barriers

- Fuel Kernel
- Fuel Particle Coatings
- Matrix/Graphite
- Helium Pressure Boundary
- Reactor Building

$\left.\begin{array}{l}\text{Fuel Kernel}\\ \text{Fuel Particle Coatings}\\ \text{Matrix/Graphite}\end{array}\right\}$ Fuel Element





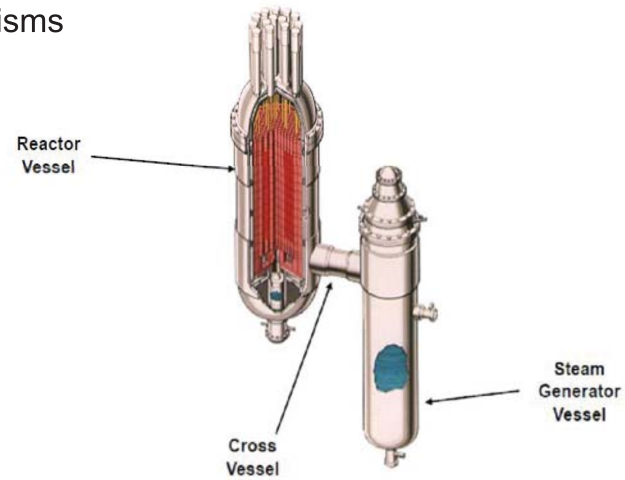


Fuel Particles Retain Radionuclides Well Above Normal Operation Temperatures

- Normal operating peak fuel temperature is <1250°C. Testing shows RN retention for hundreds of hours at >1600°C without fuel particle failure
- Large temperature margins enable:
  - Passive heat removal independent of coolant pressurization
  - Greater use of negative temperature coefficient for intrinsic reactor shutdown
- Most radionuclides reach steady state concentration/distribution in primary circuit
  - Exceptions are long lived isotopes (i.e., Cs-137 and Sr-90) where plateout inventory builds over time
- Concentration and distribution are affected by:
  - Radionuclide half-life
  - Initial fuel quality
  - Incremental fuel failures 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 (i.e., plateout)
  - Helium purification system performance

Helium Pressure Boundary (HPB) Releases

- Potential radionuclide release mechanisms
  - Primary coolant leaks
  - Liftoff (mechanical reentrainment)
  - Steam-Induced vaporization
  - Washoff (removal by liquid H_2O)
  - Primary coolant pressure relief
- Controlling parameters
  - Size/location of coolant leaks/breaks
  - Temperatures
  - Particulate matter
  - Steam/liquid H_2O ingress and egress



- Barrier performance
  - Condensable radionuclides (RNs) plate out during normal operation
  - Circulating Kr and Xe limited by Helium Purification System (HPS)
  - Plateout retained during leaks and largely retained during rapid depressurizations
  - RN holdup after core heatup due to thermal contraction of gas

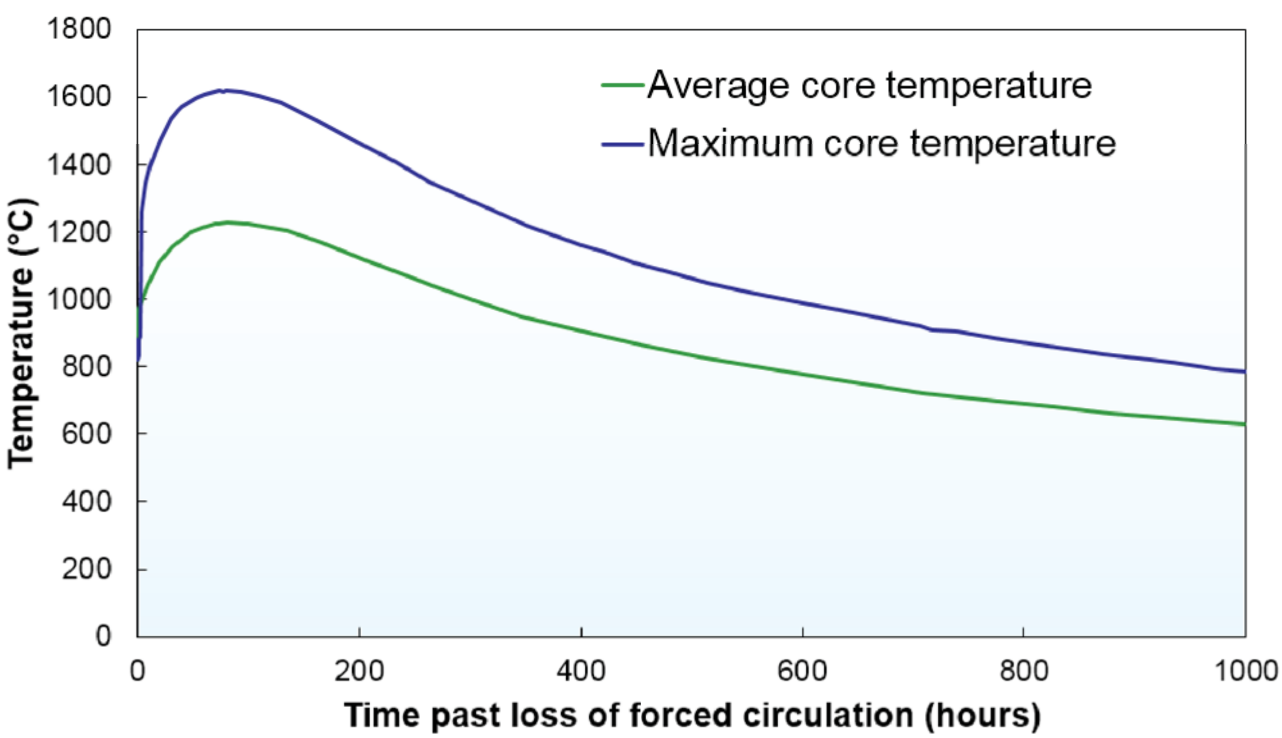
Initial RN Release Mechanisms for HPB Sources

- Circulating activity
  - Released from HPB with helium in minutes to days as a result of HPB leak/break
  - Amount of release depends on location of leak/break and any operator actions to isolate and/or intentionally depressurize
- Liftoff of plateout and resuspension of dust
  - For large breaks, fractional radionuclide amounts released from HPB with helium relatively quickly (minutes)
  - Amount of release depends on HPB break size and location
  - Surface shear forces must exceed those for normal operation to obtain liftoff or resuspension

Delayed RN Release Mechanisms From Core

- Delayed releases occur only for accidents involving a core heatup
- Partial release from contamination, initially failed/defective particles when temps exceed normal levels, and particle failures during event
- Timing of release is tens of hours to days
- Delayed inventory may be larger than circulating activity and liftoff mechanisms
- Releases from fuel depend on fraction of core above normal operation temperatures for a given time and on associated radionuclide volatility
  - Governed by amount of forced cooling
  - Dependent on size of leak or break
- Delayed releases from HPB depends on location/size of leak/break and timing relative to HPB gas expansion and contraction during core transient
  - Small leaks can potentially lead to a greater HPB RN release
  - Releases cease when internal HPB temps decrease due to core cooldown

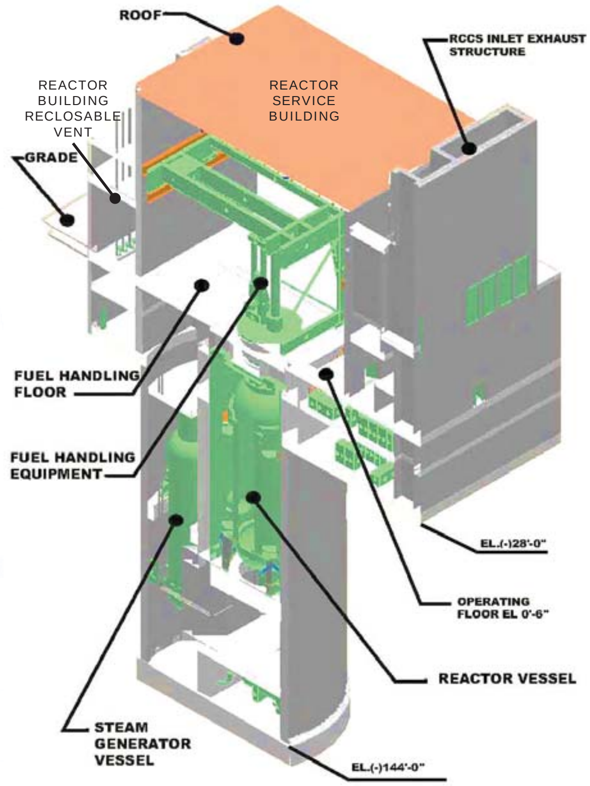
Typical Core Temperatures Following Depressurized Loss of Forced Cooling



350 MWt Prismatic Core Design

Role of Reactor Building in Safety Design

- Structurally protects pressure vessels and RCCS from internal and external hazards
- Limits air available for ingress after HPB depressurization
- Provides structural support for RCCS and helium depressurization pathway
- Provides additional radionuclide retention opportunity
- Is not relied upon for radionuclide retention to meet off-site dose regulatory requirements



Design Issues for Vented Reactor Building

- Matched to modular HTGR accident behavior
  - Reactor building is vented early in a helium pressure boundary break scenario (when the helium circulating activity is low)
  - The reactor building vent is closed later in the transient (when the particle fuel experiences maximum temperatures)
  - Prevents reactor building overpressure from release of non-condensing helium coolant
- Provides a more benign environment for the passive Reactor Cavity Cooling System (RCCS)
  - Heat
  - Pressure

The Modular HTGR Safety Approach

- Functional containment employs multiple independent and diverse barriers that work together to negate the need for a single-walled pressure-retaining structure
- Fuel has very large temperature margin in both normal and accident conditions
- TRISO fuel failure is function of time at temperature; no cliff-edge effects
- Fuel, helium, and graphite moderator are chemically compatible under all licensing basis conditions
- Safety is independent of primary circuit circulation or pressure; helium pressure loss does not transfer large energy load to reactor building
- Reactor response times are very long (i.e., days, not seconds or minutes)
- No inherent mechanism exists for runaway reactivity or power excursions

Key mHTGR Design Criteria

- MHTGR-DC 10
  - Specified acceptable fuel design limit (SAFDL) does not align with the mHTGR safety design approach
  - Replace with specified acceptable system radionuclide release design limits (SARRDL); to be defined by applicant to protect fuel during AOOs
- MHTGR-DC 16
  - Allows use of "functional containment" by multiple barriers
  - Eliminates need for pressure-retaining containment structure requirements
- MHTGR-DC 17
  - All SR power needs must be met for all applicable plant conditions
  - Battery power may be required for certain mHTGR event conditions
- MHTGR-DC 34
  - RCCS (passively) removes residual heat under off-normal conditions.
  - Provides for eliminating emergency core cooling system (ECCS)

Other mHTGR Design Criteria Considerations

- Reactor coolant makeup: helium pressure is not needed to remove heat from core (passive heat removal is used)
- Containment heat removal/atmospheric cleanup/cooling systems: mHTGRs do not employ LWR-style containment; heat removal is assured by other design criterion applicable to modular HTGRs
- Containment design/leak rate testing/containment isolation: functional containment design is addressed by the full range of mHTGR design criteria and includes new reactor building requirements
- New mHTGR reactor building design requirements
  - MHTGR-DC 70: Reactor vessel and reactor system structural design – maintain core integrity
  - MHTGR-DC 71: Reactor building design basis – protect and maintain passive cooling geometry and provide helium vent path
  - MHTGR-DC 72: Reactor building inspection – assure reactor building will perform required safety function

Major Take-Aways in Safety Design Approach

- Top-down mHTGR safety design emphasizes retention of radionuclides within very high quality TRISO fuel particles
- Independent barriers provide defense-in-depth that limit and attenuate radionuclide releases under all LBE conditions
- Residual core heat removal by passive means
- Large negative temperature coefficients
  - Shutdown without rod motion
- Overall plant design limits air/water ingress

Suggested Reading

- NGNP White Papers
  - NGNP Fuel Qualification, July 2010 (ML102040261)
  - Mechanistic Source Terms, July 2010 (ML102040260)
- **INL/EXT-11-22708, Modular HTGR Safety Basis and Approach, August 2011 (ML11251A169)**
- NGNP – Encl. 1, Summary Feedback on Four Key Licensing Issues, July 2014 (ML14174A774)
- INL/EXT-14-31179, Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors, Rev 1, December 2014 (ML14353A246, ML14353A248)
- RG-1.232, Guidance for Developing Principal Design Criteria for Non-Light Water Reactors, Appendix C – mHTGR-DC, April 2018 (ML17325A611)
- SECY-18-0096, Functional Containment Performance Criteria for Non-Light Water Reactors, w/ Encl. 1 and Encl. 2, September 28, 2018 (ML18115A157, ML18115A231, ML18115A367)
- ANL-SMR-8, Design Report for the 1/2 Scale Air-Cooled RCCS Tests in the NSTF, June 2014

INL

Idaho National Laboratory

Modular High Temperature Gas-cooled Reactor: Accident Analysis

Hans Gougar

High Temperature Gas-cooled Reactor: Accident Analysis

Advanced Reactor Technologies

Idaho National Laboratory

Hans Gougar, PhD
Nuclear Engineer

Gerhard Strydom
National Technical Director – DOE Advanced Reactor Technologies Gas-Cooled Reactor Campaign

NRC HTGR Training July 16-17, 2019

ADVANCED REACTOR TECHNOLOGIES

HTGR Accident Analysis – Overview

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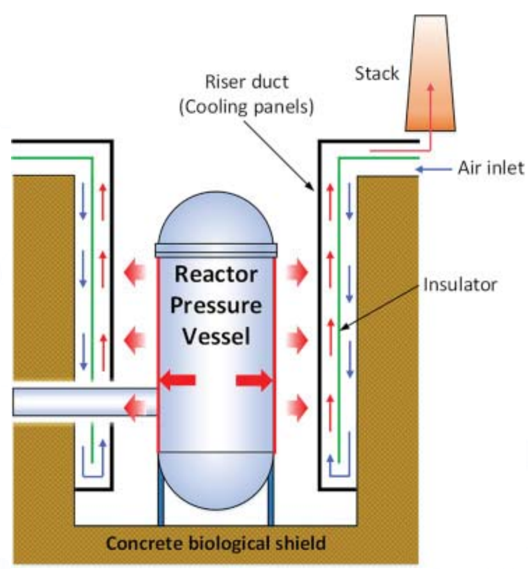
Reactor Cavity Cooling

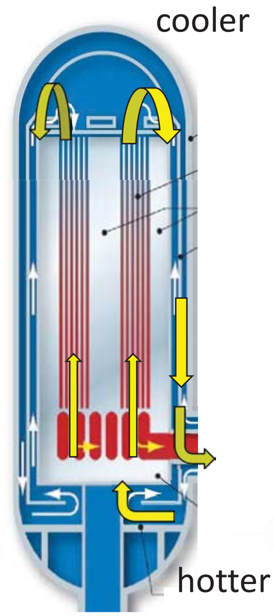


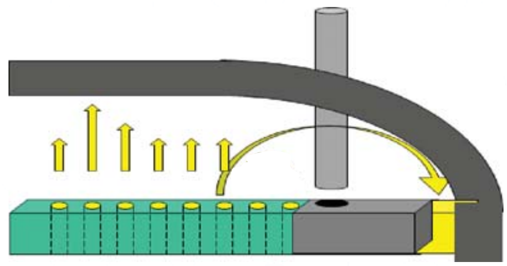
Dong-Ho Shin, Chan Soo Kim, Goon-Cherl Park, Hyoung Kyu Cho,Experimental analysis on mixed convection in reactor cavity cooling system of HTGR for hydrogen production, International Journal of Hydrogen Energy, Volume 42, Issue 34, 2017.

Pressurized Loss of Forced Cooling (PLOFC)

aka Pressurized Conduction Cooldown

- Blower trip leads to loss of forced flow through core. Doppler shuts down fission within first few seconds
- Forced downflow quickly yields to gravity-driven upflow through channels (or bed) - the transition flow is complex
- Core increases in temperature over many hours, then cools
- The hotter lower vessel structures drive 'plenum-to-plenum' currents and complex recirculation patterns
- RCCS pulls off heat from RPV
- If unmitigated (e.g., shutdown cooler), hot plumes impinging on upper plenum structures may damage CR guide tubes and the RPV head

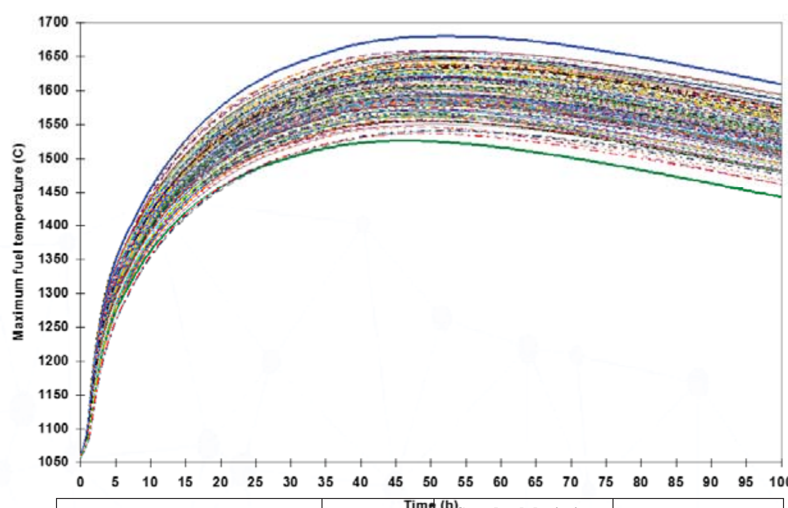




Valentin, F. I., N. Artoun, M. KawaJI and D. M. McEligot, 2018. Forced and mixed convection heat transfer at high pressure and high temperature in a graphite flow channel. J. Heat Transfer, 140, pp. 122502-1 to -10

DLOFC Uncertainties

- IAEA CRP-5 PEBBED model of the PBMR-400
- DLOFC transient sampled 200 times with SUSA uncertainty quantification code
- Input parameters sampled statistically.
- Obtains "band" of 200 peak fuel temperatures as function of time.
- 95th/95th tolerance limits of ~60°C observed (<4%).
- Only a small fraction of the fuel volume (<5%) reaches these temperatures for less than 150 hrs!



Parameter	Mean value	2 Standard deviations (2σ) value	PDF Type
Reactor power	400 MW	±8 MW (2%)	Normal & Uniform
Reactor inlet gas temperature (RIT)	500°C	±10°C (2%)	Normal & Uniform
Decay heat multiplication factor	1.0	±0.057 (5.7%)	Normal & Uniform
Fuel specific heat multiplication factor	1.0	±0.06 (6%)	Normal & Uniform
Reflector specific heat multiplication factor	1.0	±0.10 (10%)	Normal & Uniform
Fuel conductivity multiplication factor	1.0	±0.14 (14%)	Normal & Uniform
Pebble bed effective conductivity multiplication factor	1.0	±0.08 (8%)	Normal & Uniform
Reflector conductivity multiplication factor	1.0	±0.10 (10%)	Normal & Uniform

G. Strydom, 2010. PEBBED Uncertainty and Sensitivity Analysis of the CRP-5 PBMR DLOFC Transient Benchmark with the SUSA Code. INL/EXT-10-20531.

Air Ingress

- The amount of air that re-enters the primary system is a function of relative gas inventories and break location/orientation
- Oxidation of graphitic structures may ensue – mostly in the lower plenum; degrading structural integrity and perhaps causing further FP release if unmitigated. CO likely is generated
- Nuclear grade graphite does not burn (Windes, 2017) - but it does oxidize. Much of the oxygen is consumed by the lower graphite structures
- Moorman (2011) disagrees. Graphite oxidation remains misunderstood – much official OECD and IAEA documentation still erroneously refers to “graphite fires” at Windscale and Chernobyl accidents
- Graphite oxidation is temperature dependent.
  - Is it better to allow building circulation to cool the core structures or bottle it up to prevent O_2 exposure?

Oxidation/Degradation of graphite samples

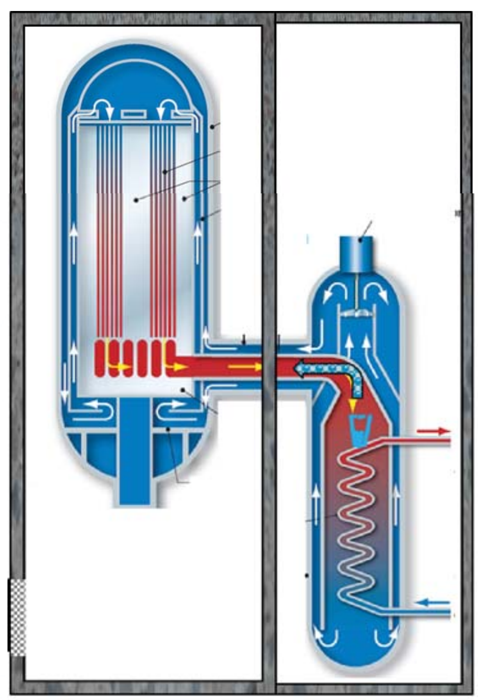
Type D Fire Extinguisher (graphite powder) used on electrical fires

Issue: How much oxygen can actually get back in? Sensitive to building air inventory and engineered vent pathways.

Windes, W. et al, “Discussion of Nuclear-Grade Graphite Oxidation in Modular High Temperature Gas-Cooled Reactors, 2017.
Moorman, R., “Phenomenology of Graphite Burning in Air Ingress Accidents of HTRs”, 2011.
Srinivasan, M., and Carlson, D. (US NRC), “Enhanced Graphite Oxidation Under Potential Accident Scenarios”, Proceedings of the 12th International Nuclear Graphite Specialists Meeting, Jeju, Korea, September 20-23, 2011.

Steam Generator (SG) Tube Rupture

- SG rupture sends water/steam into the RPV. Rupture may cause surrounding tubes to fail
- Reactivity insertion event (extra moderator)
- Moisture penetrates and oxidizes graphite surfaces. It picks up residual fission products normally trapped there. CO and volatile hydrocarbons formed
- Primary pressure relief valve opens, releasing circulating and leached FP into the building
- Relief valve closes but may reopen if more water enters and flashes. After 2-3 valve cycles, it is assumed to fail open
- Event is classified as a DLOFC with additional FP release



Issue: Amount (and phase) of water entering the core depends upon location of break. Fun multiphysics problem.

Rod Bank Withdrawal and Seismic Events

- Both are part of the reactivity insertion event class
- These events are challenging for modelers because the reactor may stay critical if not scrammed. Coupled neutronic/thermal-fluid simulations are computationally demanding for anything but simple point kinetics/homogenized core models
- Control rods in HTGRs are generally 'banked' (grouped). A spurious control signal may cause uncontrolled withdrawal, the rate of which determines rate of energy deposition and ultimate temperature increase (Rod 'ejection' is prevented by core design)
- If rapid, the heat surge will shut down the reactor (Doppler) before particle failure conditions are attained
- Explicit modeling of kernel energy deposition indicates that the lower-order (smeared) fuel models over-predict power and fuel temperature
- Likewise, seismically-induced pebble bed settling is computed to result in a positive reactivity insertion on the order of a rod withdrawal event.
- Earthquake effects on other plant structures would need to be evaluated

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Codes and Methods used for Past and Current HTGR Analysis – Prismatic

USED FOR LICENSING BENCHMARKED NEAMS

Purpose	Previously Used Codes	Codes for Today and tomorrow	Remark
Cross Section Generation	MICROX MICROR	SCALE/MCNP SERPENT	Slowing-down in graphite, heterogeneity, leakage, control rods
Criticality/Rod Worth Steady State Verification	DIF3D	Monte Carlo	
Steady State Design and Fuel Management	DIF3D/BURP	Monte Carlo with Burnup	
Time-dependent Reactor Dynamics	?	PARCS-AGREE, NEM-THERMIX, PHISICS-RELAP	Load-follow, steam ingress. PLOFC/DLFOC may work with point kinetics
Local Thermal-Fluidics	TAC-2D, TREVER, DEMISE	ANSYS, CFD	High fidelity conjugate heat transfer using finite element analysis
Core-wide Thermal Fluidics System Analysis	DETRAC,TAP, SINDA-FLUINT, RELAP5, GRSAC,	RELAP5-3D, AGREE, GASNET, RELAP7, SAM	1-D Channel Flow with input power trajectory. Flow mixing (network), Bypass flow
Thermomechanical Analysis	ANSYS	ANSYS, ABAQUS, COMSOL, GRIZZLY	2-D and 3-D solid mechanics with time history.
Seismic	ANSYS	ANSYS, MASTODON	
Fuel Performance Ex-Core FP transport	GA/KFA	PARFUME,COPA, TIMCOAT, BISON MELCOR, etc.	Fuel performance data and models may indicate that one need not take credit for retention in the building

Codes and Methods used for Past and Current HTGR Analysis – Pebble

USED FOR LICENSING BENCHMARKED NEAMS

Purpose	Previously Used Codes	Codes for Today and tomorrow	Remark
Cross Section Generation	GAM-ZUT-THERMOS	SCALE/MCNP SERPENT	Slowing-down in graphite, heterogeneity, leakage, control rods
Criticality/Rod Worth Steady State Verification	MCNP/ MonteBurns	Monte Carlo	
Steady State Design and Fuel Management	VSOP PEBBED	PARCS-AGREE MAMMOTH-PRONGHORN	Must account for flowing and mixing of fuel, including during the running –in period. Only VSOP does all of this currently
Time-dependent reactor dynamics	TINTE	PARCS-AGREE, NEM, RATTLESNAKE-PRONGHORN	Load-follow, steam ingress. PLOFC/DLFOC may work with point kinetics
Local Thermal-Fluidics	ANSYS	CFD (Fluent, Star-CCM, NEK5000)	High fidelity conjugate heat transfer using finite element analysis
Core-wide Thermal Fluidics System Analysis	THERMIX-KONVEK	RELAP5-3D, AGREE, GASNET, PRONGHORN, RELAP7,SAM, FLOWNEX, SURVEY	Porous medium conjugate heat transfer with subgrid pebble conduction for the core. Bypass flow in the reflector
Thermomechanical Analysis	ANSYS	ANSYS, ABAQUS, COMSOL, GRIZZLY	2-D and 3-D solid mechanics with time history.
Seismic	ANSYS	ANSYS, MASTODON	2-D and 3-D time-dependent structural mechanics with time history
Fuel Performance Ex-Core FP Transport	PANAMA, FIPREX-GETTER	PARFUME,COPA,TIMCOAT, STACY, BISON MELCOR, etc.	Semi-analytical models of FP transport in fuel.

Tolerances in General Atomic's Neutronic Codes (C-E)/E)

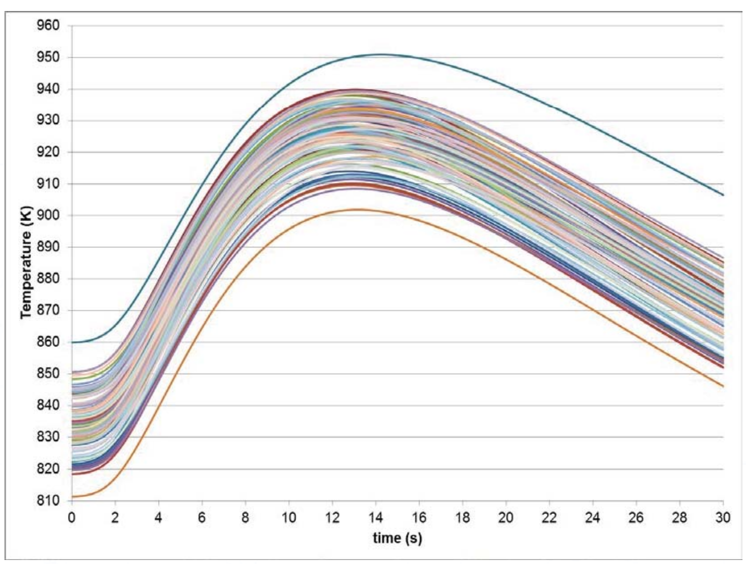
Facility	Temp. Defect	C. R. Worth	Power Distr.	K_{eff}	Water Ingress	Decay Heat
HEU-CORES						
Peach Bottom Critical	±14%	-11%	±10%	±0.7%	NA	-
Peach Bottom	-11% to +4%	-6% to +10%	±10%	±0.7%	-	NA
HTGR Critical	+6%	+4% to 13%	-	-0.1% to +1.0%	-	-
Fort St. Vrain	-9% to +12%	±10%	±15%	±0.5%	-	NA
HTLTR	±8%	-	-	-	-	-
KAHTER	-	NA	NA	-0.3% to +6%	±13%	-
DRAGON	NA	-11%	NA	-	-	NA
HEU/LEU CORES						
AVR	-25%	-5% to +15%	-	±11%	-	NA
LEU CORES						
HITREX-2	-	-	±10%	±0.5%	-	
HITREX-2	-	-	±10%	±0.5%	-	

Baxter, A.. (General Atomics) Module 5b - Prismatic Nuclear Design, HTGR Technology Course for the Nuclear Regulatory Commission, 2010.

Recent Uncertainty Assessment

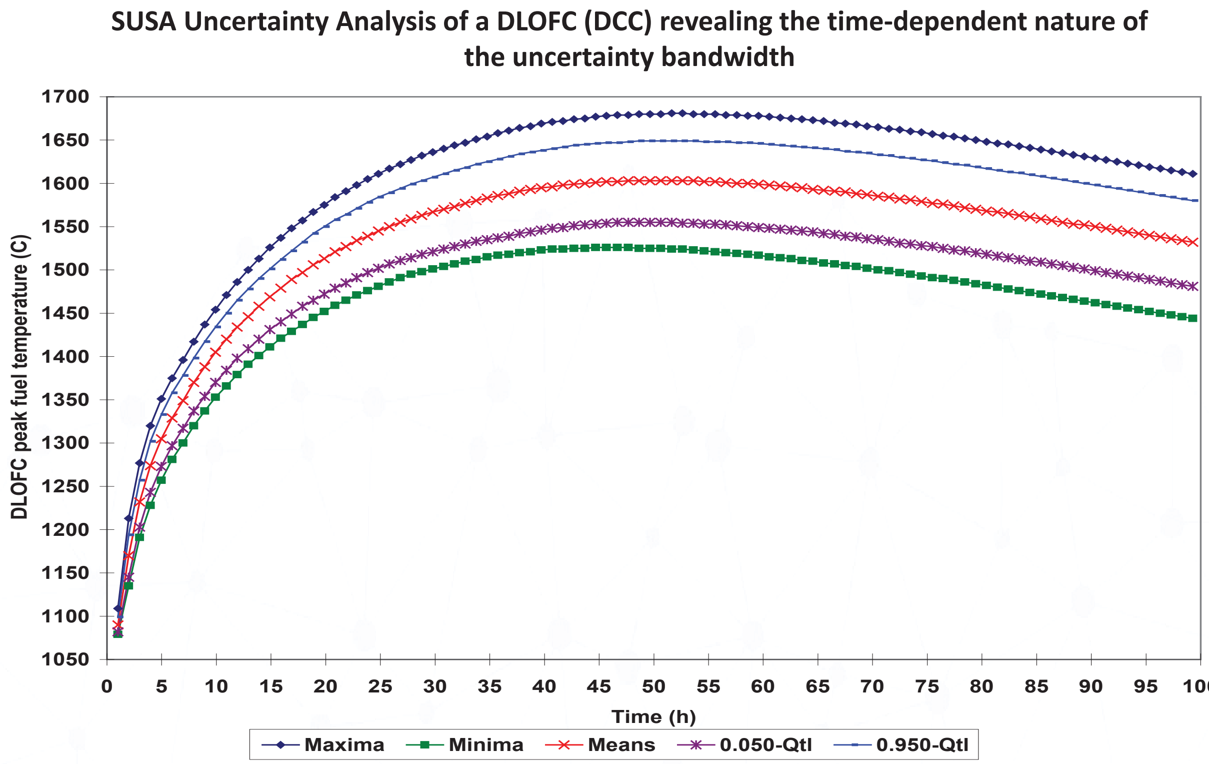
Input Parameter	2σ Uncertainty
Core Bypass Flow (or gap width)	± 5.9%
Reactor Inlet Temperature	± 2%
Helium Mass Flow	± 2%
Pebble Bed Thermal Conductivity	± 8%
Fuel Sphere Graphite Thermal Conductivity	± 14%
Fuel Sphere Graphite Specific Heat	± 6%
Reflector Thermal Conductivity	± 10%
Reflector Specific Heat	± 10%
Reflector Emissivity	± 7%
Core Barrel Emissivity	± 5%
Reactor Pressure Vessel (RPV) Emissivity	± 7%
Core Barrel Thermal Conductivity	± 5%
Helium Thermal Conductivity	± 5%
RPV Thermal Conductivity	± 5%
Core Barrel Specific Heat	± 5%
Helium Specific Heat	± 5%

Fuel temperature response for 1,000 perturbed CFX calculations (slow power ramp transient)



Strydom G., Bostelmann, F., and Yoon, S. J., 2015, Results for Phase 1 of the IAEA Coordinated Research Project on HTGR Uncertainties, INL/EXT-14-32944, Rev. 2.

Time-dependent Uncertainties



Strydom, G., Uncertainty and Sensitivity Analysis of a Pebble Bed HTGR Loss of Cooling Event, Science and Technology of Nuclear Installations, Volume 2013, Article ID 426356.

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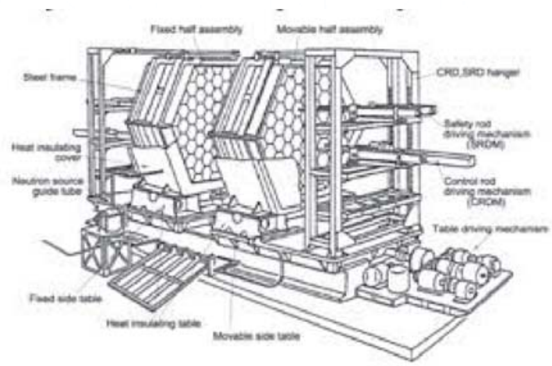
Critical Experiments for Neutronics

The HTR Proteus experiment from above.

PEBBLE BED
- HTR-Proteus critical experiments
  - 1980's, Paul Scherrer Institute, Switzerland
  - Bess 2014
- HTR-10 Initial Criticality
  - ~2000, INET, China
  - IAEA 2003, 2013
- ASTRA
  - Mid 1990's , Kurchatov Institute, Russia
  - IAEA 2013
- HTR-PM – scheduled to go critical within a year. INET has offered up physics test results to support a GIF benchmark

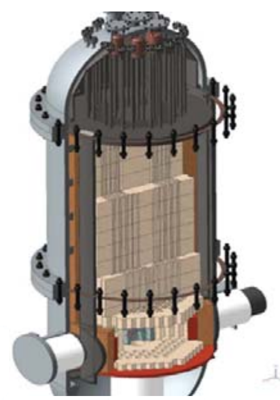
PRISMATIC
- VHTRC
  - Mid-1980s, Japan
  - Ref: Bostelmann 2016
- HTTR
  - Ref: IAEA 2003, 2013
- Fort St. Vrain
  - Ref: Martin, 2016

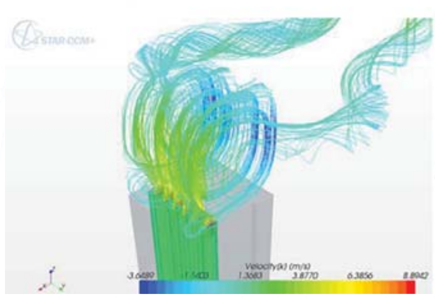


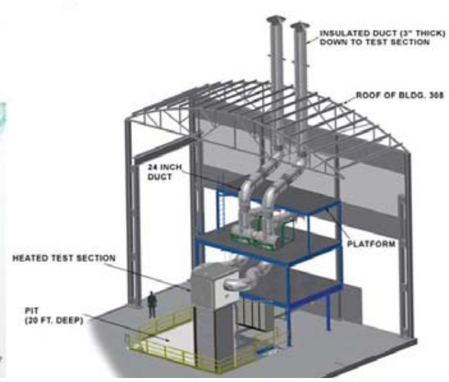
Sketch of the VHTRC Experiment

Thermal Fluid Integral Experiments Sponsored by DOE

- High Temperature Test Facility at Oregon State University
- Natural Circulation Shutdown Heat Removal Facility at Argonne National Lab
- Vendors participated in the design and test matrix planning for the HTTR and NSTF experiments.
- Framatome and X-Energy facilitated the conversion of NSTF to a water-cooled configuration.
- The NRC sponsored the design and construction of HTTF

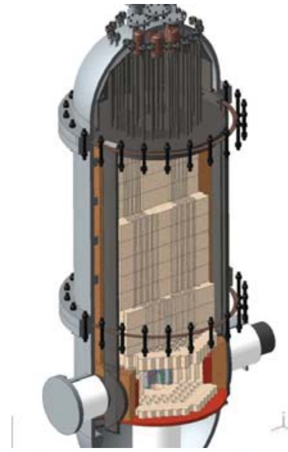


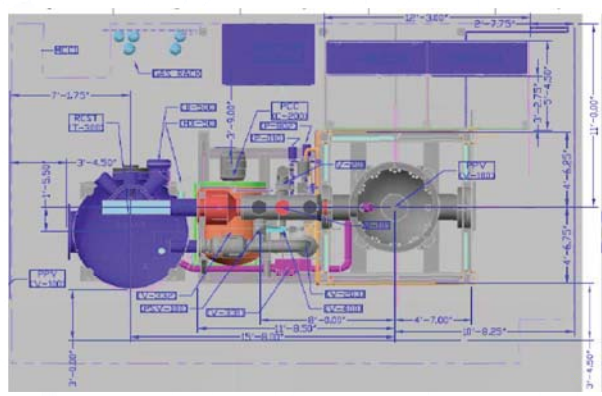




High Temperature Test Facility (HTTF) – Oregon State University

- Designed to simulate core behavior during a depressurized loss of forced cooling accident
- ¼-scale MHTGR
- Design allows different pipe break configurations to characterize the exchange of helium and air between the primary loop and building
- 428 experiment measurements (362 thermocouples, 48 gas sensors, 18 others) and 31 process instruments
- Primary focus is on depressurized conduction cooldown transient, but other experiments are planned as well.
- Matrix testing resumed in April 2019





High Temperature Test Facility – Oregon State University (cont.)

- HTTR encountered local over-heating during initial testing. The heaters and instrumentation have been re-designed and rebuilt. Four ceramic blocks were replaced

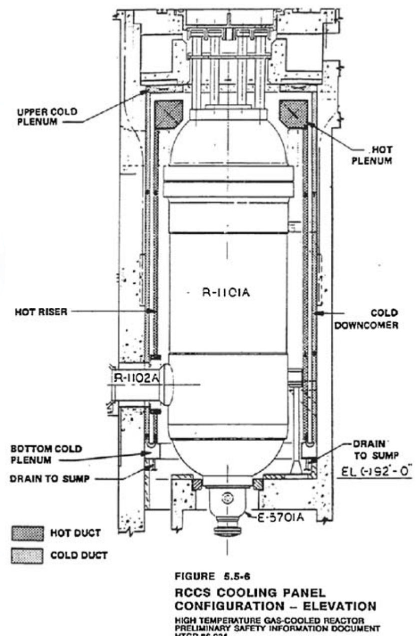
New ‘dogbone’ heater rod

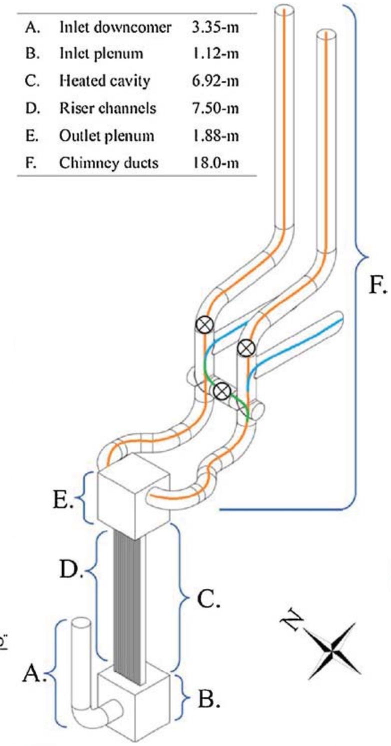
Current heater rod stack

Damaged ceramic core block resulting from block shifting that degraded heater rod electrical continuity causing localized hot spots - 4 need to be replaced

Reactor (Vessel) Cavity Cooling System

- Active or passive heat removal via absorption of thermal radiation (shine) emitted from a hot uninsulated reactor pressure vessel
- Ultimately rejects heat to the atmosphere
- Air-cooled, water-cooled, or hybrid configurations

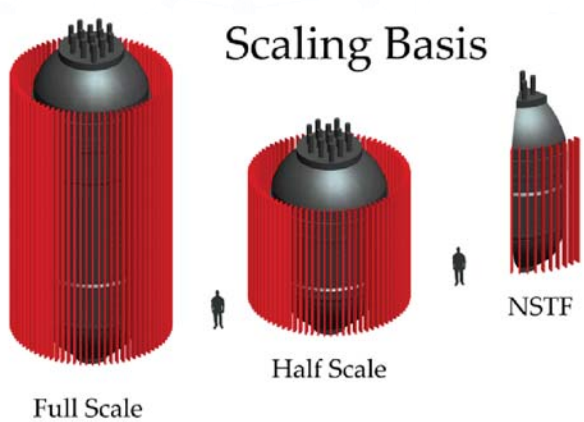




Lisowski, D.D. et al, Experimental Observations of Natural Circulation Flow in the NSTF, Nuclear Engineering and Design 306, (2016) 124-132.

Reactor Vessel (Cavity) Cooling System Experiments at ANL's Natural Circulation Shutdown Heat Removal Facility (NSTF)

- Originally constructed to support General Electric PRISM (Power Reactor Inherently Safe Module) development refurbished to half-MHTGR scale under the Next Generation Nuclear Plant project
- Air-cooled experiments completed in 2016
- Conversion to water-cooled configuration (Framatome SC-HTGR). Experiments have commenced.



Finned water-cooled riser channels in the NSTF

Numerous NEUP-funded Experiments

- Separate and Mixed Effects studies in:
  - Bypass Flow
  - Core Heat Transfer
  - Air Ingress
  - Plenum-to-Plenum Heat Transfer
  - Lower plenum flow
  - Building Response to depressurization

High Pressure, High Temperature Facility for Natural Circulation Experiments, City College of New York , NEUP Project 11-3218,Kawaji)

Phenomena Characterized in:

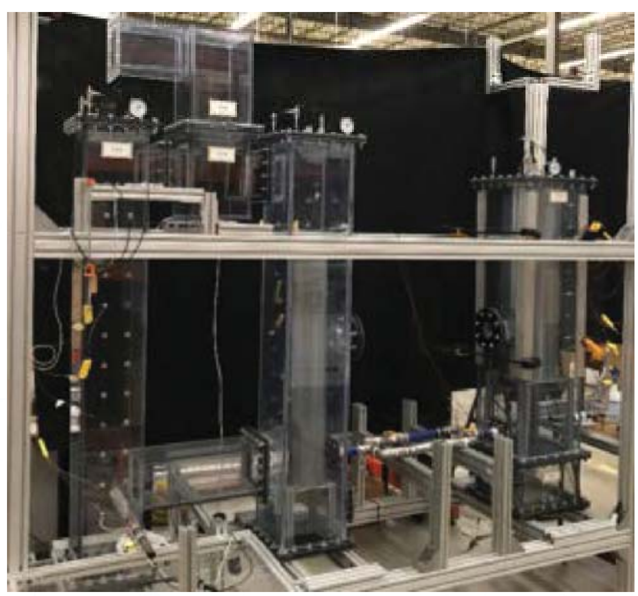
Schultz, R.R., Gougar, H., Lommers, L., Identification and Characterization of Thermal Fluid Phenomena Associated with Selected Operating/Accident Scenarios in Modular High Temperature Gas-Cooled Reactors, Paper 2018-0177, Proceedings of HTR 2018, Warsaw, Poland, October 8-10, 2018.

Final Reports downloadable from https://neup.inl.gov

Building Response

- Advanced Reactor Concepts grant to Texas A&M with cost-share with AREVA
- Designed to look at flow in the reactor building subsequent to pipebreak and depressurization
- Initial tests were completed. Further experiments solicited in the 2019 NEUP call

1/32-scale Building Response Experiment at Texas A&M



Se Ro Yang, Ethan Kappes, Thien Nguyen, Rodolfo Vaghetto, Yassin Hassan, Experimental study on 1/28 scaled NGNP HTGR reactor building test facility response to depressurization event, Annals of Nuclear Energy, Volume 114, 2018.

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- Types of Potential Accidents and Reactor Response
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Safety Analysis Approach

Each scenario must be evaluated in the context of:

- Phenomenology and sequence timing (what happens and when)
- Break size, break location, orientation
- Graphite structural material (nuclear or non-nuclear)
- Building response

Design-dependent

Design implications

- Mitigation systems?
- Accident management procedures?

Design-dependent: redundancies, diversities, etc.

Credible break size:

- Design basis?
- Beyond design basis?
- Best Estimate or conservative approach (Code of Federal Regulations [CFR])
- Acceptance criteria?

"Cliff-edges" have been largely eliminated but 'knife-edge' transitions can effect source terms, structural integrity

Summary

- **Safety margins are huge** in terms of radiological release. Accident scenarios develop slowly and have few consequences for the fuel. Other structures may be vulnerable
- **Graphite can oxidize** and air ingress can degrade structural integrity if not designed away or mitigated. Graphite does not burn
- **Moisture ingress** (steam generator tube rupture) may be the limiting case with respect to fission product release
- Codes system designed for HTGRs exist and have improved since the first HTGRs were licensed (but they were adequate for the purpose). Computational power is driving **more extensive use of high fidelity tools**. Margins, however, still allow **approximate methods** to be used effectively
- Uncertainties can be **large, time-dependent** and are mostly attributable to uncertainties in material properties and tolerances, not so much to neutronic uncertainties
- Critical experiment data are limited but probably adequate. Integral experiments are underway at ANL and Oregon State University to **confirm gross thermal-fluid behavior**. Numerous SET and MET experiments have been conducted. (International integral tests and engineering reactors were not discussed but may be useful.)
- Safety Analysis must factor individual design features but the general approach applies to all modular HTGRs. “Cliff-edges” really do not appear in existing design concepts but **“knife-edge” phenomena should be identified and understood to characterize margins to FP release**

Suggested Reading List

- John D. Bess, Leland M. Montierth, Oliver Köberl & Luka Snoj (2014) Benchmark Evaluation of HTR-PROTEUS Pebble Bed Experimental Program, Nuclear Science and Engineering, 178:3, 387-400, DOI: 10.13182/NSE14-13
- Bostelmann,F., Hammer, H. Ortensi, J. Strydom, G. Velkov, K., Zwermann, W., Criticality calculations of the Very High Temperature Reactor Critical Assembly benchmark with Serpent and SCALE/KENO-VI, Annals of Nuclear Energy, Volume 90, 2016,
- Dong-Ho Shin, Chan Soo Kim, Goon-Cherl Park, Hyoung Kyu Cho, Experimental analysis on mixed convection in reactor cavity cooling system of HTGR for hydrogen production, International Journal of Hydrogen Energy, Volume 42, Issue 34, 2017.
- HTGR Technology Course for the Nuclear Regulatory Commission, 2010.
- INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of High Temperature Gas Cooled Reactor Performance: Benchmark Analysis Related to Initial Testing of the HTTR and HTR-10, IAEA-TECDOC-1382, IAEA, Vienna (2003).
- INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of High Temperature Gas Cooled Reactor Performance: Benchmark Analysis Related to the PBMR-400, PBMM, GT-MHR, HTR-10 and the ASTRA Critical Facility, IAEA-TECDOC-1694, IAEA, Vienna (2013).
- Lisowski, D.D. et al, Experimental Observations of Natural Circulation Flow in the NSTF, Nuclear Engineering and Design 306, (2016) 124-132.
- Martin, W, Creation of a Full-Core HTR Benchmark with the Fort St. Vrain Initial Core and Assessment of Uncertainties in the FSV Fuel Composition and Geometry, Battelle Memorial Institute United States. Department of Energy. Office of Scientific and Technical Information.
- Moorman, R., “Phenomenology of Graphite Burning in Air Ingress Accidents of HTRs”, 2011.
- Schultz, R.R., Gougar, H., Lommers, L., Identification and Characterization of Thermal Fluid Phenomena Associated with Selected Operating/Accident Scenarios in Modular High Temperature Gas-Cooled Reactors, Paper 2018-0177, Proceedings of HTR 2018, Warsaw, Poland, October 8-10, 2018.
- Strydom G., Bostelmann, F., and Yoon, S. J., 2015, Results for Phase 1 of the IAEA Coordinated Research Project on HTGR Uncertainties, INL/EXT-14-32944, Rev. 2.
- Strydom, G., Uncertainty and Sensitivity Analysis of a Pebble Bed HTGR Loss of Cooling Event, Science and Technology of Nuclear Installations, Volume 2013, Article ID 426356
- Se Ro Yang, Ethan Kappes, Thien Nguyen, Rodolfo Vaghetto, Yassin Hassan, Experimental study on 1/28 scaled NGNP HTGR reactor building test facility response to depressurization event, Annals of Nuclear Energy, Volume 114, 2018.
- Valentin, F. I., N.Artoun, M. KawaJI and D. M. McEligot, 2018. Forced and mixed convection heat transfer at high pressure and high temperature in a graphite flow channel. J. Heat Transfer, 140, pp. 122502-1 to -10
- Windes, W. et al, “Discussion of Nuclear-Grade Graphite Oxidation in Modular High Temperature Gas-Cooled Reactors, 2017.

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Modular High Temperature Gas-cooled Reactor: Accident Analysis Continued

Jim Kinsey

Modular High Temperature Gas-cooled Reactor: Accident Analysis

Advanced Reactor Technologies

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NRC HTGR Training July 16-17, 2019

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ART ADVANCED REACTOR TECHNOLOGIES

Outline

- Licensing Modernization Project (LMP)
  - Risk Informed Approach
  - Selection of Licensing Basis Events (LBEs)
  - Frequency-Consequence Target
  - LBE Cumulative Risk Targets
  - Structures, Systems, and Components (SSC) Safety Categories Classification
  - Evaluation of Defense in Depth (DID)
- Use of Probabilistic Risk Assessment (PRA) in LMP Process
  - PRA Policy Statement
  - American Nuclear Society (ANS) Non-Light-Water Reactor (non-LWR) PRA Standard

Risk-Informed Approach

- NRC PRA Policy Statement motivates risk-informed, performance-based (RIPB) approach to modular High Temperature Gas-cooled Reactor (HTGR) licensing
- Complements traditional deterministic design approach to increase use of risk insights in design and licensing decisions
- Risk-informed approach:
  - Explicit consideration to a broader set of challenges
  - Logical prioritization of challenges
  - Consideration of broader set of resources to defend against challenges
  - Explicitly identifying and quantifying sources of uncertainty
  - Better decision making by testing for sensitivity to key assumptions
- Performance-Based Approach:
  - Measurable (or calculable) parameters for monitoring
  - Objective criteria to assess performance

PRA Development

- Early introduction of PRA into design process facilitates risk-informing design decisions
- Scope and level of detail consistent with scope and level of detail of design and site information and fit for purpose in RIPB decisions
- PRA event-sequences include those involving single and multiple reactor modules and risk significant non-reactor sources
- Non-LWR PRA standard specifically designed to support LMP PRA applications
- Limitations and uncertainties associated with PRA addressed in the evaluation of defense-in-depth adequacy

Objectives of the Licensing Modernization Project (LMP)

- From draft LMP Guideline Document (NEI 18-04):
  - The scope of this document is focused on establishing guidance for advanced (i.e., non-LWR) designs so license applicants can develop inputs that can be used to comply with applicable regulatory requirements, …
  - Technology inclusive
- Based on 10 CFR 50.34 and other regulatory requirements, an applicant must answer the following questions:
  - What are the plant initiating events, event sequences, and accidents that are associated with the design?
  - How does the proposed design and its SSCs respond to initiating events and event sequences?
  - What are the margins provided by the facility's response, as it relates to prevention and mitigation of radiological releases within prescribed limits for the protection of public health and safety?
  - Is the philosophy of DID adequately reflected in the design and operation of the facility?

LMP – Licensing Basis Events

- LBEs are defined broadly to include all the events used to support the safety aspects of the design and to meet licensing requirements
- They cover a comprehensive spectrum of events from normal operation to rare, off-normal events
- Categories defined as Normal Operations, including Anticipated Operational Occurrences (AOO), Design Basis Events (DBE), Beyond Design Basis Events (BDBE) and Design Basis Accidents (DBA)
- LBE definitions generally consistent with Next Generation Nuclear Plant (NGNP) white papers
- Draft LMP guidance document (NEI 18-04) includes glossary to clarify differences in terminology with regulatory terms

LBE Categories

Anticipated Operational Occurrences (AOOs). Anticipated event sequences expected to occur one or more times during the life of a nuclear power plant, which may include one or more reactor modules. Event sequences with mean frequencies of 1×10^{-2}/plant-year and greater are classified as AOOs. AOOs take into account the expected response of all SSCs within the plant, regardless of safety classification.

Design Basis Events (DBEs). Infrequent event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than an AOO. Event sequences with mean frequencies of 1×10^{-4}/plant-year to 1×10^{-2}/plant-year are classified as DBEs. DBEs take into account the expected response of all SSCs within the plant regardless of safety classification. The objective and scope of DBEs to form the design basis of the plant is the same as in the NRC definition.

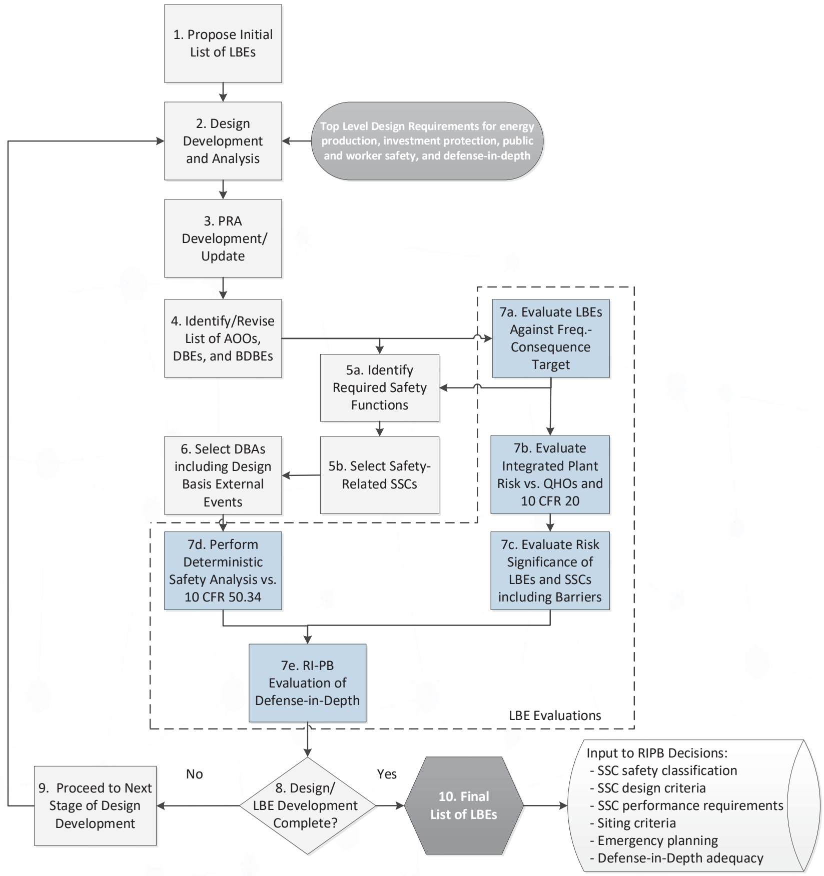
Beyond Design Basis Events (BDBEs). Rare event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than a DBE. Event sequences with mean frequencies of 5×10^{-7}/plant-year to 1×10^{-4}/plant-year are classified as BDBEs. BDBEs take into account the expected response of all SSCs within the plant regardless of safety classification.

Design Basis Accidents (DBAs). Postulated accidents that are used to set design criteria and performance objectives for the design and sizing of SSCs that are classified as safety-related. DBAs are derived from DBEs based on the capabilities and reliabilities of safety-related SSCs needed to mitigate and prevent accidents, respectively. DBAs are derived from the DBEs by prescriptively assuming that only SSCs classified as safety-related are available to mitigate postulated accident consequences to within the 10 CFR 50.34 dose limits.

LMP – Selection and Evaluation of LBEs

- AOOs, DBEs, and BDBEs are defined in terms of event sequence families from a design-specific PRA
- AOOs, DBEs, and BDBEs are evaluated:
  - Individually for risk significance using a Frequency-Consequence (F-C) chart
  - Collectively by comparing the total integrated risk against a set of cumulative risk targets
- DBEs and high consequence BDBEs are evaluated to define Required Safety Functions necessary to meet F-C Target
- Designer selects Safety Related SSCs to perform required safety functions among those available on all DBEs
- DBAs are derived from DBEs by assuming failure of all non-safety related SSCs and evaluating the consequences conservatively vs. 10 CFR 50.34

LBE Selection and Evaluation



LMP – Frequency-Consequence Target

- Purpose is to evaluate risk significance of individual LBEs and to help define the Required Safety Functions
- Derived from the NGNP F-C Target and frequency bins for AOOs, DBEs, and BDBEs
- Addressed the “staircase” issue with previous F-C targets (NGNP and NUREG-1860)
- F-C Target anchor points based on:
  - 10 CFR 20 annual dose limits and ISO-risk concept
  - Avoidance of offsite protective actions for lower frequency AOOs
  - 10 CFR 50.34 dose limits for lower frequency DBEs
  - Consequences based on 30 day TEDE dose at EAB
  - Doses at EAB are used to assure meeting QHO for prompt fatality individual risk

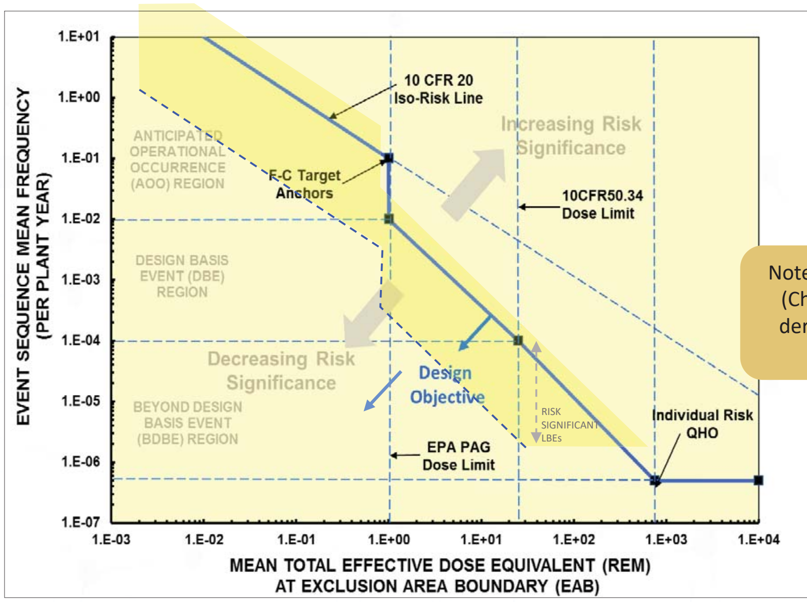
LMP – F-C Target

The F-C Target values shown in the figure should not be considered as a demarcation of acceptable and unacceptable results. The F-C Target provides a general reference to assess events, SSCs, and programmatic controls in terms of sensitivities and available margins.

AOOs

DBEs

BDBEs



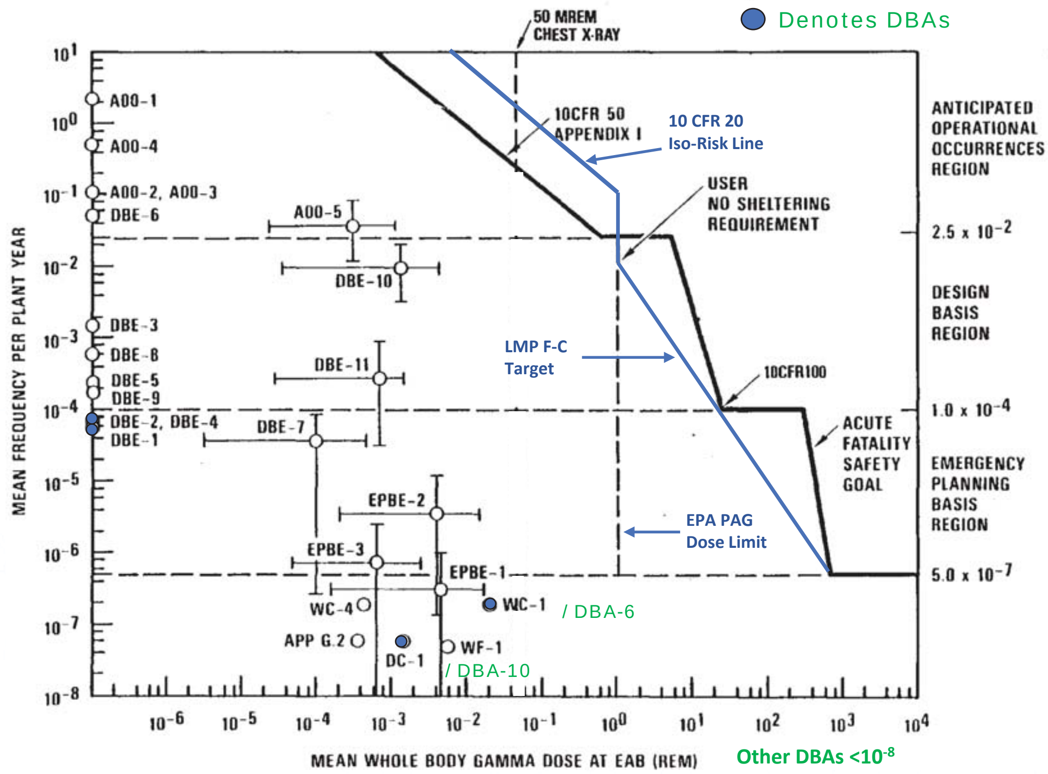
Note that DBAs (Chapter 15) derived from DBEs

* F-C Target considered along with cumulative risk metrics, safety classification, and assessment of defense in depth

LMP – LBE Cumulative Risk Targets

- The total frequency of exceeding an offsite boundary dose of 100 mrem shall not exceed 1/plant-year to ensure that the annual exposure limits in 10 CFR 20 are not exceeded
- The average individual risk of early fatality within the area 1 mile of the Exclusion Area Boundary (EAB) shall not exceed 5×10^{-7}/plant-year to ensure that the NRC Safety Goal Qualitative Health Objective (QHO) for early fatality risk is met
- The average individual risk of latent cancer fatalities within the area 10 miles of the EAB shall not exceed 2×10^{-6}/plant-year to ensure that the NRC safety goal QHO for latent cancer fatality risk is met

Historical Example: MHTGR DBEs, DBAs, and BDBEs on F-C Plot (circa 1987)



LMP – SSC Approach Highlights

- Adopts three SSC safety classification categories in NGNP SSC white paper
- Proposes criteria for SSC risk significance based on absolute risk metrics
- Incorporates concepts from 10 CFR 50.69 and NEI-00-04
- Includes SSC requirements to address single and multi-module risks
- Expands on guidance for deriving performance requirements beyond those in NGNP SSC white paper

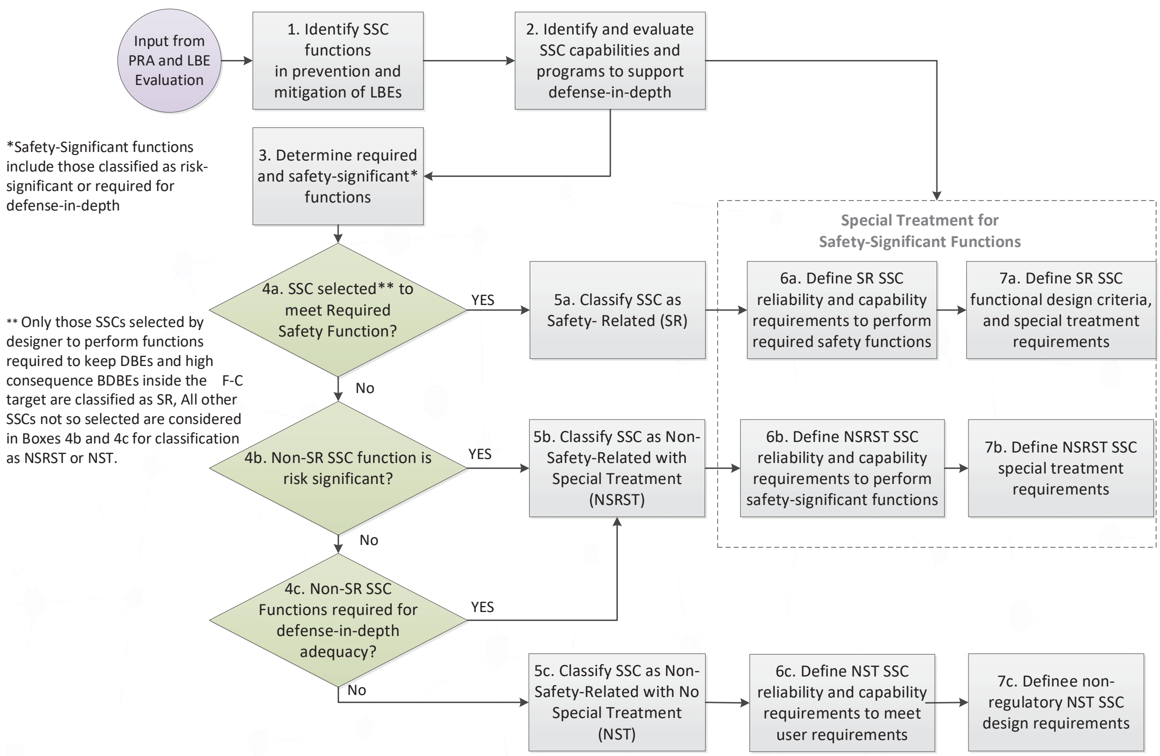
LMP – Proposed SSC Safety Categories

- Safety-Related (SR):
  - SSCs selected by the designer to perform required safety functions to mitigate the consequences of DBEs to within the F-C target, and to mitigate DBAs to meet the dose limits of 10 CFR 50.34 using conservative assumptions.
  - SSCs selected by the designer to perform required safety functions to prevent the frequency of BDBEs with consequences greater than 10 CFR 50.34 dose limits from increasing into the DBE region and beyond the F-C target.
- Non-Safety-Related with Special Treatment (NSRST):
  - Non-safety related SSCs relied on to perform risk significant functions. Risk significant SSCs are those that perform functions that keep LBEs from exceeding the F-C target, or make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs.
  - Non-safety related SSCs relied on to perform functions requiring special treatment for DID adequacy.
- Non-Safety-Related with No Special Treatment (NST):
  - All other SSCs.

LMP – SSC Risk Significance

- A prevention or mitigation function of the SSC is necessary to meet the design objective of keeping all LBEs within the F-C target
  - The LBE is considered within the F-C target when a point defined by the upper 95%-tile uncertainty of the LBE frequency and dose estimates are within the F-C target
- The SSC makes a significant contribution to one of the cumulative risk metrics used for evaluating the risk significance of LBEs
  - A significant contribution to each cumulative risk metric limit is satisfied when total frequency of all LBEs with failure of the SSC exceeds 1% of the cumulative risk metric limit. The cumulative risk metrics and limits include:
    - The total frequency of exceeding of a site boundary dose of 100 mrem < 1/plant-year (10 CFR 20)
    - The average individual risk of early fatality within 1 mile of the EAB < 5×10^{-1}/plant-year (QHO)
    - The average individual risk of latent cancer fatalities within 10 miles of the EAB shall not exceed 2×10^{-6}/plant-year (QHO)

LMP – SSC Safety Classification Approach



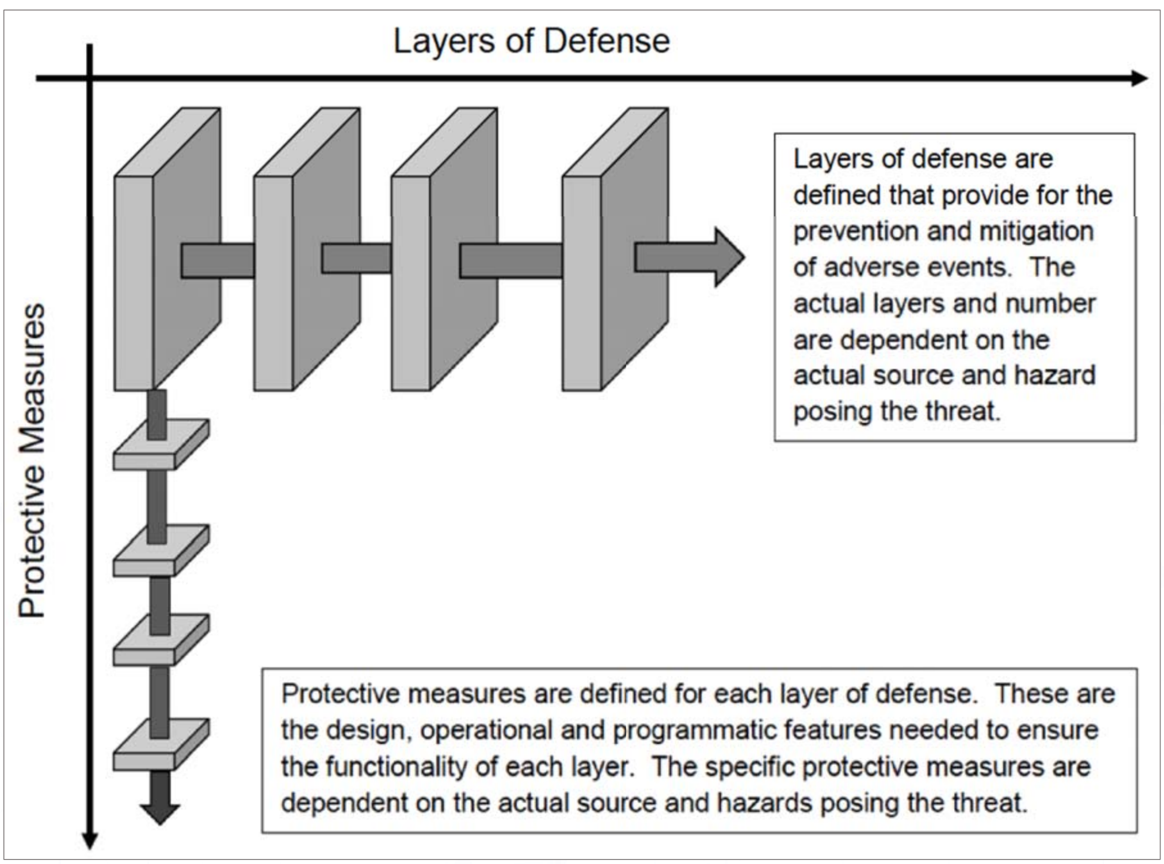
LMP – Derivation of Special Treatment Requirements

- SR SSCs:
  - Functional Design Criteria derived from required safety functions
  - Lower level design criteria derived from SRDC
- SR and NSRST SSCs:
  - SSC reliability and capability performance targets
  - Focus on prevention and mitigation functions from LBEs
  - Integrated decision making process to derive specific special treatment requirements
  - Reflects concepts from 10 CFR 50.69 and NEI-00-04
  - Reflects Commission’s expectations for risk-informed and performance based regulation from SRM to SECY 98-0144

LMP – SSC Classification Summary

- LMP retains the NGNP SSC safety categories of SR, NSRST, and NST
- All safety significant SSCs classified as SR or NSRST
- Absolute risk metrics proposed for SSC and LBE risk significance
- All SR SSCs are classified as risk significant
- NSRST SSCs include other risk significant SSCs and SSCs requiring some special treatment for Defense In Depth (DID) adequacy
- Specific special treatment for capabilities and reliabilities in the prevention and mitigation of accidents
- Special treatment defined via integrated decision panel

DID Concept from NUREG/KM-0009



LMP – DID Adequacy Approach

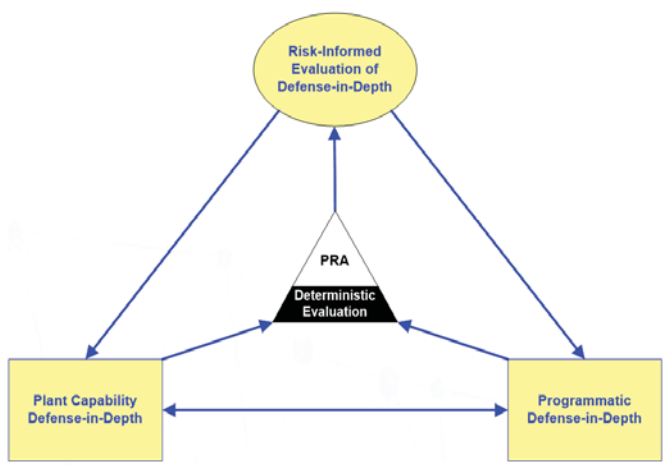
- Builds on NGNP DID approach also reflected in ANS-53.1
- Evaluation of DID adequacy is both risk-informed and performance-based
- The “layers of defense” and attributes of the NRC and IAEA DID frameworks are more visibly represented
- DID attributes for plant capability and programmatic DID have been enhanced for consistency with the measures defined in the LMP Guidance Document
- This process is used to evaluate each LBE and to identify the DID attributes that have been incorporated into the design to prevent and mitigate accident sequences and to ensure that they reflect adequate SSC reliability and capability
- Those LBEs with the highest levels of risk significance are given greater attention in the evaluation process
- The practicality of compensatory actions for DID purposes are considered in the context of the individual LBE risk significance and in a cumulative manner across all LBEs

LMP – Defense In Depth Adequacy Basic Structure

Plant Capability DID

Plant Functional Capability DID - This capability is introduced through systems and features designed to prevent occurrence of undesired LBEs or mitigate the consequences of such events.

Plant Physical Capability DID - This capability is introduced through SSC robustness and physical barriers to limit the consequences of a hazard.



Programmatic DID

Programmatic DID addresses uncertainties when evaluating plant capability DID. It incorporates special treatment during design, manufacturing, constructing, operating, maintaining, testing, and inspecting of the plant and the associated processes to ensure there is reasonable assurance that the predicted performance can be achieved and verified throughout the lifetime of the plant.

Risk-Informed Evaluation of DID

This element provides a systematic, holistic, integrated, and transparent process for examining the DID adequacy achieved by the combination of plant capability and programmatic elements. This evaluation is performed by a risk-informed integrated decision-making (RIDM) process to assess and establish whether DID is sufficient to enable consideration of different alternatives for achieving commensurate safety levels at reduced burdens.

DID Adequacy Evaluation Process

- DID Baseline Evaluation documented by Integrated Decision Panel (IPD) and updated at each design/licensing stage
- Defense-in-depth is deemed by IDP as adequate when:
  - Plant capability DID is deemed to be adequate
  - Plant capability DID guidelines in Table 5-2 (NEI 18-02) are satisfied
  - Review of LBEs is completed with satisfactory results
  - Programmatic DID is deemed to be adequate
  - Performance targets for SSC reliability and capability are established
  - Sources of uncertainty in selection and evaluation of LBE risks are identified
  - Special treatment for all SR and NSRST SSCs is sufficient

Commission Policy: Use of PRA Methods in Nuclear Regulatory Activities

"The Commission believes that the use of PRA in regulatory activities should be increased to the extent supported by the state-of-the-art PRA methods and data in a manner that complements the NRC deterministic approach."

"[T]he expanded use of PRA technology will continue to support the NRC's defense-in-depth philosophy by allowing quantification of the levels of protection and by helping to identify and address weaknesses or overly conservative regulatory requirements applicable to the nuclear industry. Defense-in-depth is a philosophy used by NRC to provide redundancy for facilities with 'active' safety systems, e.g., a commercial nuclear power (sic), as well as the philosophy of a multiple-barrier approach against fission product releases."

(FR, Vol. 60, No. 158, pg. 42622-42629, August 16, 1995)

The Commission's 1995 PRA Policy Statement

- "A probabilistic approach to regulation enhances and extends this traditional, deterministic approach, by:
  1) Allowing consideration of a broader set of potential challenges to safety,
  2) Providing a logical means for prioritizing these challenges based on risk significance, and
  3) Allowing consideration of a broader set of resources to defend against these challenges."
- The LMP approach is consistent with this policy

Non-LWR PRA Standard

- ASME/ANS started the development of a non-LWR PRA standard in 2006 and produced a trial use standard ASME/ANS-Ra-S-1.4-2013
- Approximately 80% of the technical requirements are common to the LWR PRA standards; remaining 20% address:
  - Risk metrics appropriate for all advanced non-LWRs
  - PRAs on multi-module plants
  - PRAs that support event sequence frequencies and consequences
  - PRAs that are performed at early stages in design
- Trial use standard is currently being revised towards a ballot for an ASME/ANS standard in 2019

LMP – Treatment of Passive Systems

- The PRA standard requires a quantitative uncertainty analysis of phenomena to quantify the failure probability of passive systems
- NEI 18-04 (draft) does not require assuming complete failure of passive SSCs or inherent features. However, NEI 18-04 does require SSC failure mode determinations by the developer as part of safety case development, and also requires the definition of Required Safety Functions and plant features responsible for fulfilling them.
- This topic is covered in other available references, including:
  - ASME/ANS-RA-S-1.4 Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants
  - NUREG-0800 Standard Review Plan Chapter 19, and
  - Other regulatory guidance such as NUREG-1855

Suggested Reading

- NEI 18-04, Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development, Draft Report Revision N
- Draft Regulatory Guide 1353, Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Approach to Inform the Content of Applications for Licenses, Certification, and Approvals for Non-Light-Water Reactors (ML18271A164)
- Draft SECY, Technology-Inclusive, Risk-Informed, and Performance-Based Approach to Inform the Content of Applications for Licenses, Certification, and Approvals for Non-Light-Water Reactors (ML18270A334)
- NGNP Licensing Strategy Report to Congress describes a risk-informed approach for NGNP Licensing, 2008.
- ASME/ANS RA-S-1.4-2013, Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants, 2013.
- ANSI/ANS-53.1-2011 – Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants.

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TRISO Fuel: Mechanistic Source Term

Paul Demkowicz

TRISO Fuel: Mechanistic Source Term

Advanced Reactor Technologies

Idaho National Laboratory

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NRC HTGR Training July 16-17, 2019

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ADVANCED REACTOR TECHNOLOGIES

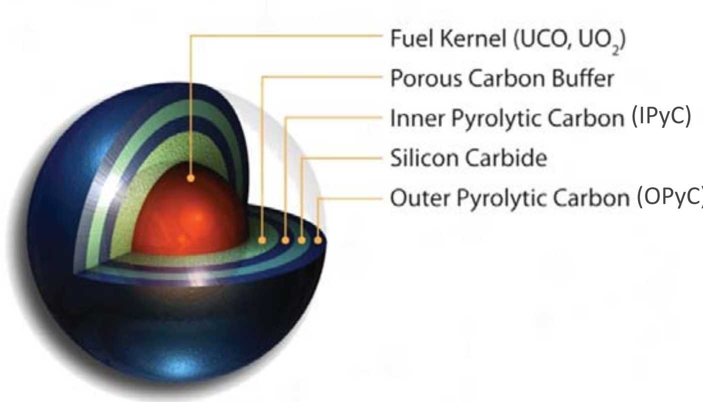
Outline

- Barriers to radionuclide (RN) release in high-temperature gas-cooled reactors (HTGRs)
- Radionuclide Design criteria
- Computational tools to predict radionuclide release
- Simple model to estimate source term from HTGRs

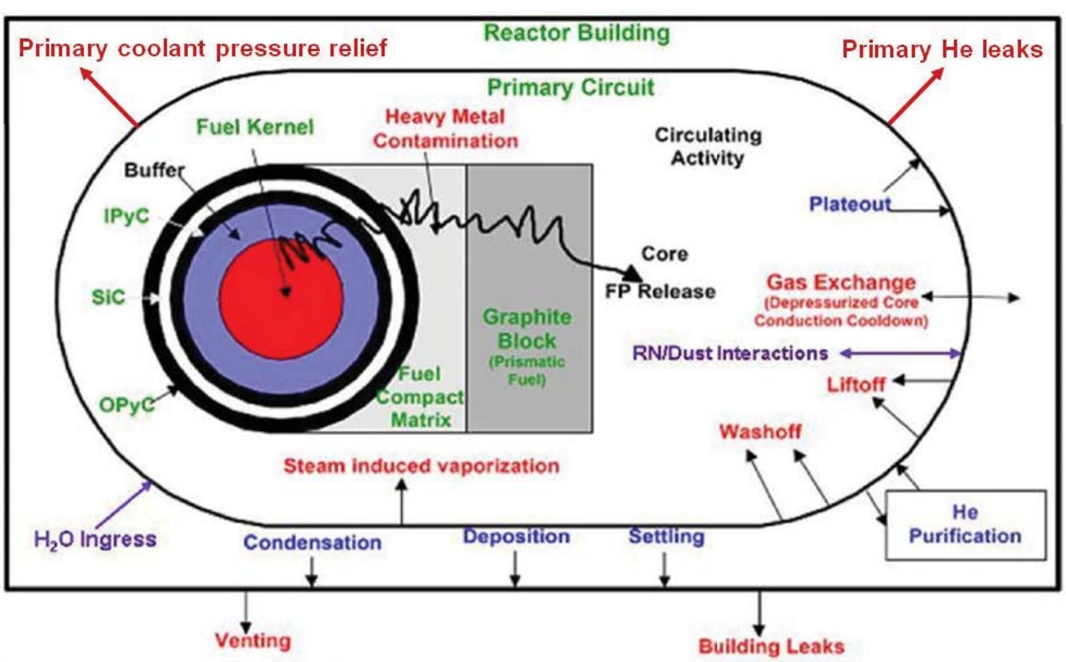
The Training Course delivered to the NRC in 2010 included a module discussing radionuclide behavior in HTGRs (Module 13). You are encouraged to review that course material for additional details.

Radionuclide Barriers

- HTGR designs employ multiple radionuclide release barriers
  - Fuel kernels
  - Particle coatings (most important barrier)
  - Fuel-element matrix and fuel-element graphite (prismatic reactor)
  - Primary coolant pressure boundary
  - Reactor building (RB)
- These multiple radionuclide barriers provide defense in depth

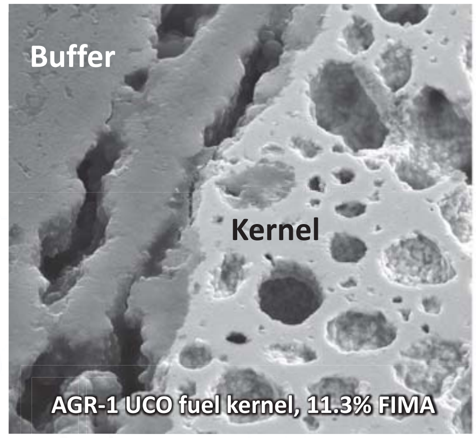


HTGR Radionuclide Sources and Pathways



Radionuclide Release Barrier: Fuel Kernel

- Potential release mechanisms
  - Fission recoil
  - Diffusion
  - Hydrolysis (reaction with H_2O)
- Controlling parameters
  - Fuel temperatures
  - Time
  - H_2O concentration
  - Burnup
- Barrier performance
  - Fractional gas release function of time/temperature history
  - Increased gas release in case of hydrolysis
  - Partial diffusive release of volatile fission metals (Ag, Cs > Eu, Sr)
  - Other radionuclides, including actinides, very well retained
  - UO_2 expected to stabilize certain elements to a greater extent than UCO (Sr, Eu)



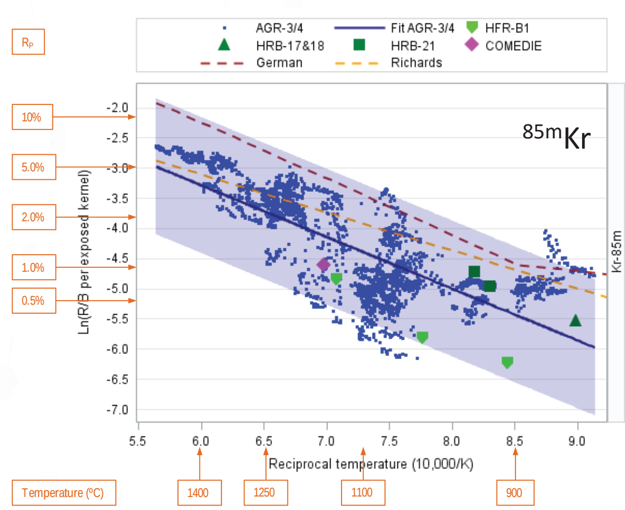
Fission Gas Release

- Short-lived fission gas release in-pile evaluated using the “release-rate-to-birth-rate” (R/B) ratio
- Release is a function of element (Kr, Xe), isotope half life and fuel temperature
- Gas release calculated using a Booth model

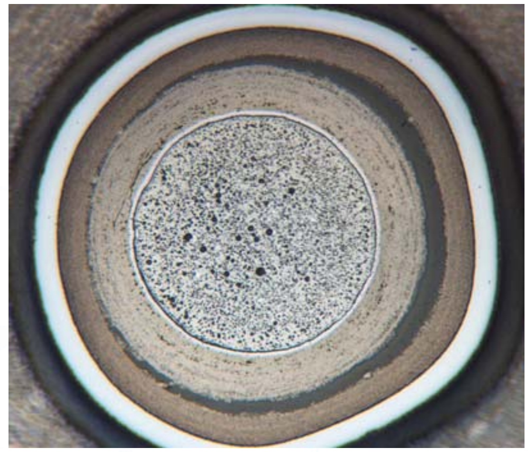
$$R_p \approx \frac{3}{x} = 3\left[\frac{D_0 e^{-\frac{E_a}{RT}}}{\lambda a^2}\right]^n$$

- Various experiments have been performed to determine the R/B ratio for exposed kernels (including recent AGR-3/4 irradiation)
- Isotopes commonly measured:

Kr-85m	Xe-131m
Kr-87	Xe-133
Kr-88	Xe-135
Kr-89	Xe-135m
Kr-90	Xe-137
	Xe-138
	Xe-139



Radionuclide Release Barriers: Particle Coatings



- Potential release mechanisms
  - Diffusion through intact coatings
  - As-fabricated coating defects
  - In-service coating failure
- Controlling parameters
  - Fuel temperatures
  - Time
  - Fast neutron fluence
- Barrier performance
  - Transport through intact coatings:
    - Ag significantly released
    - Other fission products (e.g., Sr, Eu) exhibited more modest release
  - Gases retained by OPyC with defective/failed SiC
  - Metals released when SiC fails
  - TRISO failure rates are very low in modern TRISO fuel

Radionuclide Release Barriers: Matrix/graphite

- Potential release mechanisms
  - Diffusion/vaporization
  - Matrix/graphite oxidation
- Controlling parameters
  - Temperature
  - Time
  - Fast neutron fluence
  - Oxidant concentration
- Barrier performance
  - Sr and Eu exhibit fairly high retention
  - Cs and Ag exhibit lower levels of retention
  - Kr, Xe, and I not retained
  - Sorbed metals assumed to be released by oxidation

Radionuclide Release Barriers: Primary Coolant Circuit

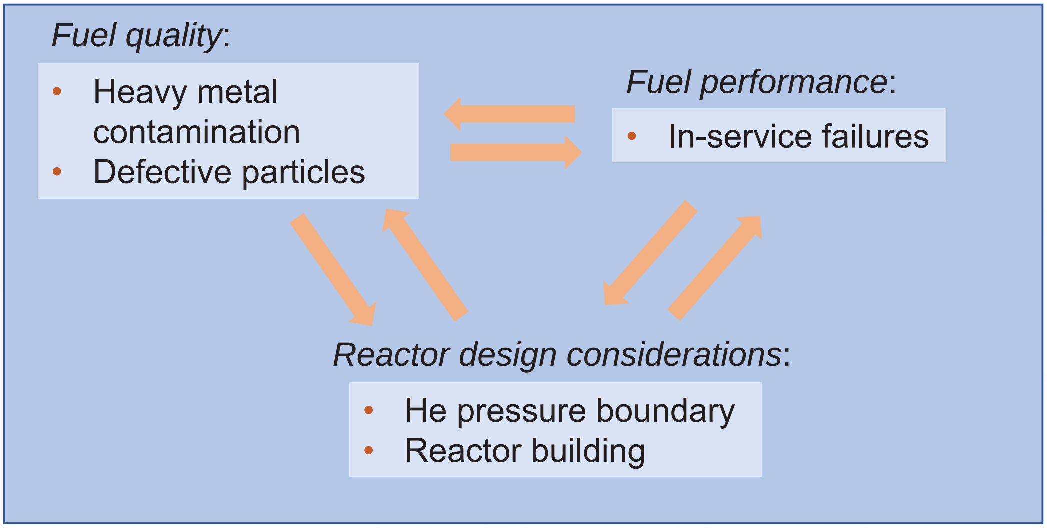
- Potential release mechanisms
  - Primary coolant leaks
  - Liftoff (mechanical re-entrainment)
  - Primary coolant pressure relief
  - Steam-induced vaporization
  - Washoff (removal by liquid H_2O)
- Controlling parameters
  - Temperatures in primary circuit
  - Size/location of coolant leaks
  - Particulate matter in primary circuit
  - Steam/liquid H_2O ingress and egress
- Barrier performance
  - Condensable radionuclides plate out during normal operation
  - Circulating Kr, Xe and H-3 limited by He purification system
  - Plateout largely retained during rapid blowdowns

Radionuclide Release Barriers: Reactor Building

- Potential release mechanisms
  - Venting through louvers
  - Building leakage
- Controlling parameters
  - Leak path(s) and rates
  - Contaminated steam/liquid H_2O
  - Contaminated particulate matter
  - Temperatures along leak path(s)
- Barrier performance
  - Noble gases decay during holdup
  - Condensable fission products, including iodine, deposit
  - Contaminated steam condenses
  - Contaminated dust settles out and deposits

Radionuclide Release Tradeoffs

- Tradeoffs exist between the relative allocations of the performance of some of the barriers



- Example: allowable fuel contamination levels may be higher if in-service failures are much lower than expected

Particulate Matter ("Dust") in Primary Circuit May Alter Fission Product Transport Behavior

- Potential sources of dust in HTGRs
  - Foreign material from initial construction or refueling
  - Abrasion/attrition of spherical fuel elements (pebble bed)
  - Erosion or corrosion of fuel or reflector blocks (prismatic)
  - Foreign material from interfacing systems (e.g., HPS)
  - Spallation of friable metallic surface films
  - Carbon deposition from CO decomposition
- Potential impact on fission product (FP) transport
  - Altered FP plateout distributions in primary circuit
  - Enhanced FP release from primary circuit into reactor building
  - Altered FP transport behavior in reactor building
- Experience:
  - Prismatic: very little dust formation; minor impact (PB1, FSV, HTTR)
  - Pebble bed: Measured and characterized in AVR; had impact on plant D & D

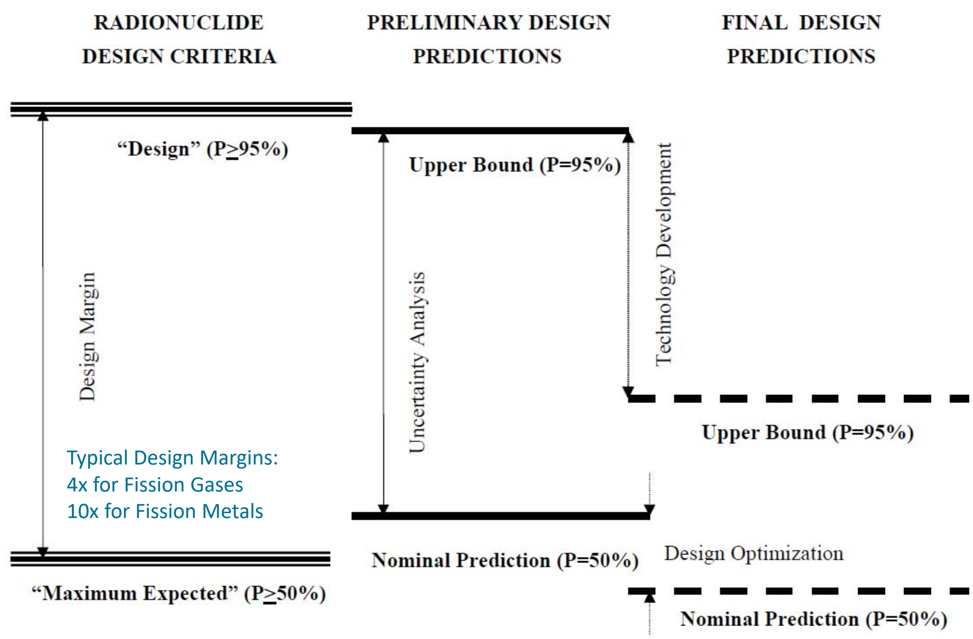
Tritium Release in HTGRs

- Tritium (H-3) will be produced by nuclear reactions
  - Ternary Fission (Yield = ~1 x 10^{-4})
  - Neutron activation of impurities (He-3 in coolant; Li in graphite)
  - Neutron capture in boron control materials
- Some H-3 will accumulate in primary helium
  - Controlled by He Purification System
  - Significant sorption on core graphite
- Fraction of circulating H-3 in He will permeate through intermediate heat exchanger (IHX) and SG with potential to contaminate process gases and steam
  - Generally not a concern with 750°C outlet temperatures; becomes more important at outlet of 900°C
- H-3 will contribute to public and occupational exposures
  - Environmental releases from plant (liquid discharge)
  - Contaminated products (e.g., hydrogen, bitumen)
- Data on H-3 transport in reactors and in relevant materials have been obtained with dedicated experiments and through reactor operating experience
- In operating reactors, offsite H-3 release has been below regulatory limits

Radionuclide Design Criteria

- "Top down" approach used to determine allowable radionuclide releases within the functional containment system
- Start with imposed requirements (e.g., site-boundary dose limits)
- Allowable radionuclide inventories in primary circuit derived from radionuclide control requirements
  - Two-tier set of "Radionuclide design criteria" defined to explicitly include safety factors in plant design

Design Margins (Safety Factors) Are Explicitly Included in Radionuclide Design Criteria (Prismatic Example)



A Review of Radionuclide Release from HTGR Cores During Normal Operation, EPRI report 1009382 (2003)

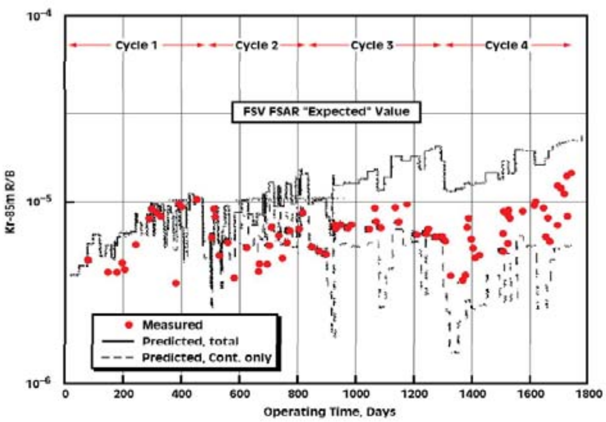
Computational Tools to Predict Radionuclide Behavior in HTGRs

- Design methods for predicting FP transport in HTGRs derived from experimental data
  - Typically, design codes model multiple radionuclide release barriers
  - Core analysis codes typically model fuel performance as well
  - Core codes are typically design specific (i.e., prismatic or pebble)
  - Phenomenological component models derived from data
  - Material property data (e.g., diffusivities, etc.) required as input
- Many comparisons of code predictions with experimental data
  - Reactor surveillance, in-pile tests, etc.
  - Agreement between predictions and measurements has been reasonably good, with predictions somewhat conservative relative to measurements
  - Codes not completely verified and validated
  - Comparisons are often with data from integral release tests; validation of contributions from specific mechanisms is difficult.

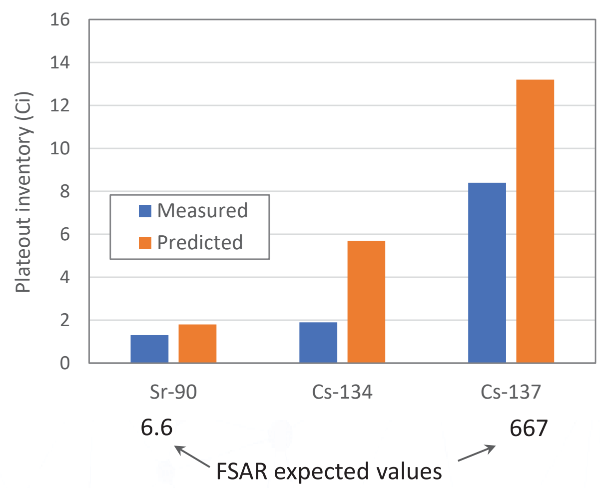
Comparison of Code Predictions with Data

- Past radionuclide release predictions have been reasonably accurate and conservative

Comparison of FSV predicted and measured Kr-85m release



Comparison of FSV predicted and measured fission metal release

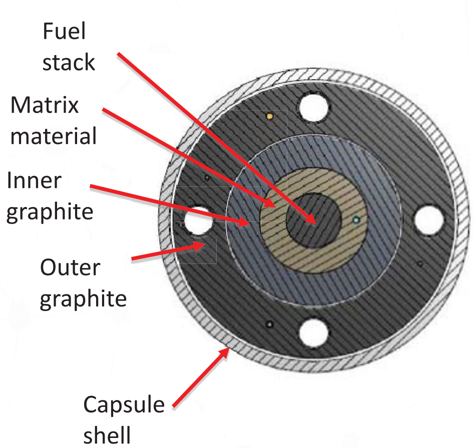


Many additional examples presented in TECDOC-978:

Fuel performance and fission product behavior in gas cooled reactors, IAEA, TECDOC-978 (1997)

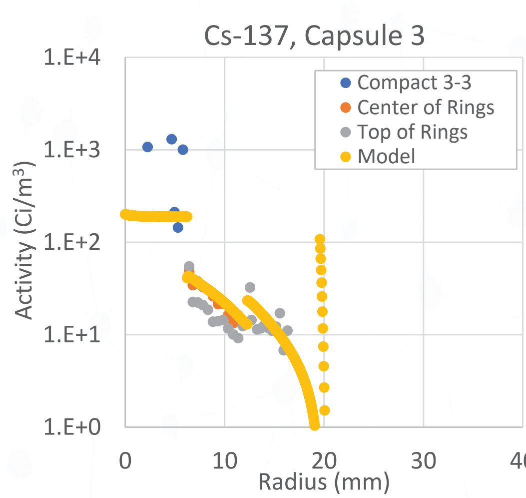
Dedicated Experiments to Refine Source Term Input Parameters

- US Advanced Gas Reactor (AGR) Program AGR-3/4 Irradiation Experiment
- Measure fission product transport in matrix and graphite materials



AGR-3/4 Capsule Cross Section

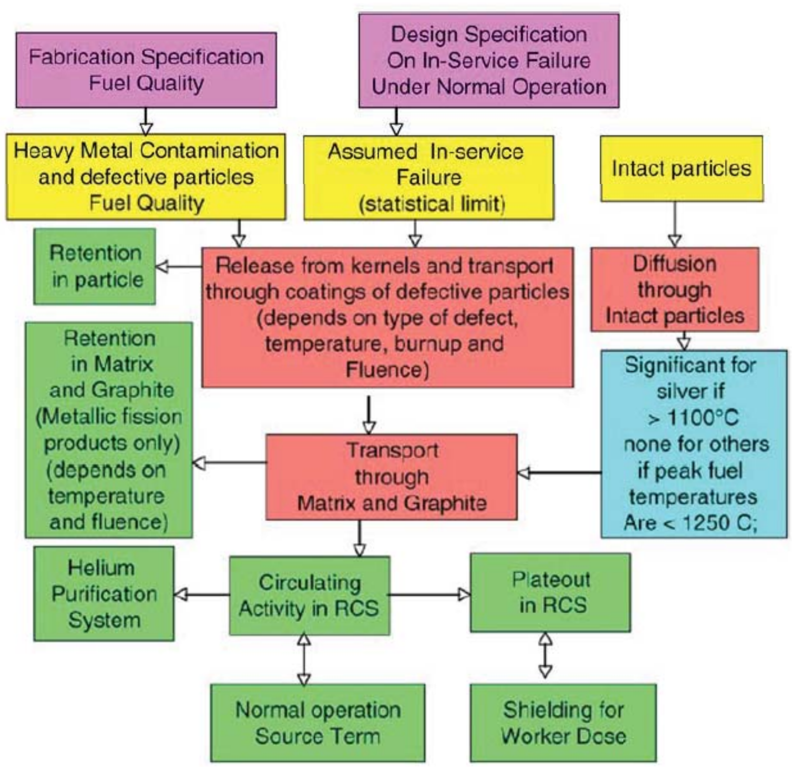
Comparison of measured and predicted Cs profile in AGR-3/4 graphite rings



Simple Model to Estimate Source Terms and Demonstrate the Impact of Barrier Performance

- D.A. Petti et al., Representative source terms and the influence of reactor attributes on functional containment in modular high-temperature gas-cooled reactors, Nucl. Tech. 184 (2013) 181-197
- Radionuclide release from core during normal operation from 4 sources:
  - Release from heavy metal contamination
  - Release from TRISO fuel with SiC defects
  - Release from in-service particle failures
  - Diffusive release through fuel particle coatings
- Total radionuclide inventory in the fuel calculated (e.g., ORIGEN)
- "Attenuation factors" (AFs) are applied at various levels to account for retention of radionuclides (calculated at 50% and 95% confidence)
  - Kernels
  - Coatings
  - Graphite
  - He pressure boundary (liftoff)
  - Reactor building
- AFs determined based on expert opinion (informed by experimental data, calculations)
- Accommodate uncertainties in defect/failure fractions and AFs using Monte Carlo approach

mHTGR Source Term During Normal Operation



D.A. Petti et al., *Representative source terms and the influence of reactor attributes on functional containment in modular high-temperature gas-cooled reactors*, Nucl. Tech. 184 (2013) 181-197

Calculating Inventory Released from Separate Sources

Release from heavy metal contamination (includes contamination outside SiC layers and exposed kernels)

$$R^i_{HMC} = \frac{Inv^i * HMC}{AF^i_{HMC} * AF^i_G}$$

Release from particles with defective SiC and in-service failure

$$R^i_{DSiC+ISF} = \frac{Inv^i * (DSiC + ISF)}{AF^i_K * AF^i_G}$$

HMC	= level of heavy metal contamination
Inv^i	= inventory of fission product *i*
R^i_{HMC}	= release of fission product *i* from HMC (Ci)
$R^i_{DSiC+ISF}$	= release of fission product *i* from SiC defects and in-service failures (Ci)
AF^i_{HMC}	= attenuation factor of fission product *i* for HMC
AF^i_G	= attenuation factor of fission product *i* in graphite
AF^i_K	= attenuation factor of fission product *i* in kernel
$DSiC$	= level of SiC defects
ISF	= level of in-service failures

- Similar equations for release through intact particles, inventory retained in the graphite, and plate-out inventory

Radionuclide Release Attenuation Factors

Prismatic reactor barrier core-average attenuation factors during normal operations (700°C Reactor Outlet Temperature)

Fission Product Class	Heavy Metal Contamination		Fuel Particle Kernel		Diffusive Release through Fuel Particle Coatings		Graphite (Compact Matrix and Fuel Element)		Helium Pressure Boundary	
Confidence Limit	AF_{HMC} 50%	AF_{HMC} 95%	AF_K 50%	AF_K 95%	AF_{Diff} 50%	AF_{Diff} 95%	AF_G 50%	AF_G 95%	AF_{HPB} 50%	AF_{HPB} 95%
Noble Gases	10	3	50	17	10^8	10^7	1	1	1	1
I, Br, Se, Te	10	3	50	17	10^8	10^7	1	1	10^6	10^5
Cs, Rb	1	1	3	1	10^8	10^6	5	2	10^6	10^5
Sr, Ba, Eu	1	1	50	20	10^3	200	10^3	300	10^6	10^5
Ag, Pd	1	1	2	1	500[a]	100[a]	2	1	10^6	10^5
Sb	1	1	2	1	10^8	10^6	20	2	10^6	10^5
Mo, Ru, Rh, Tc	1	1	500	30	10^8	10^7	10^3	300	10^6	10^5
La, Ce	1	1	500	30	10^8	10^7	10^3	300	10^6	10^5
Pu, actinides	1	1	10^3	100	10^8	10^7	10^4	10^3	10^6	10^5

[a] Values presented here for Ag-110m. For Ag-111, the values for the diffusive release through the coating are increased by a factor of 5 to account for the effect of the half-life on the release.

D.A. Petti et al., *Representative source terms and the influence of reactor attributes on functional containment in modular high-temperature gas-cooled reactors*, Nucl. Tech. 184 (2013) 181-197

Radionuclide Inventories During Normal Operation: Representative Examples for Prismatic Reactor Design

Mean values for I-131, Cs-137, and Sr-90 inventories (curies) released to the helium pressure boundary and retained in the fuel matrix and graphite

Reactor Design Configuration	I-131		Cs-137		Sr-90	
	In Fuel Matrix and Graphite	In He Pressure Boundary	In Fuel Matrix and Graphite	In He Pressure Boundary	In Fuel Matrix and Graphite	In He Pressure Boundary
600 MW(t) Prismatic 700°C ROT	nil	30	24	5	2750	0.1
600 MW(t) Prismatic 900°C ROT	nil	74	226	254	5680	31

ROT: Reactor outlet temperature

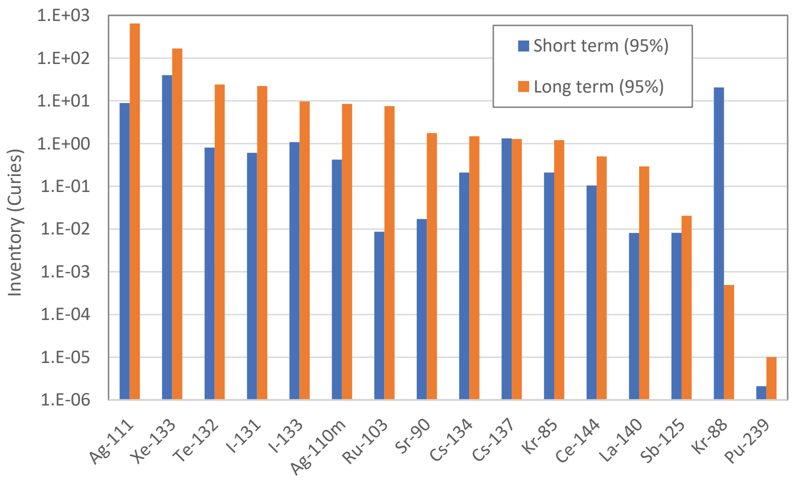
D.A. Petti et al., *Representative source terms and the influence of reactor attributes on functional containment in modular high-temperature gas-cooled reactors*, Nucl. Tech. 184 (2013) 181-197

Evaluating Radionuclide Release During Licensing Basis Events

- Similar approach taken for accident scenarios to determine total source term released from reactor building
- Radionuclide sources in accidents
  - Release from heavy metal contamination
  - Release from TRISO fuel with SiC defects
  - Release from in-service particle failures
  - Diffusive release through fuel particle coatings
  - Inventory in the graphite/matrix from normal operation
  - Lift-off of inventory plated out on the coolant boundary during normal operation
- Different accident scenarios have specific attenuation factors depending on accident conditions (temperature, dry/wet conditions, etc.)
  - e.g., moisture can increase release from exposed kernels and lift-off from the pressure boundary
- Separate calculations for short- and long-term release for accidents (driven by differing half-lives of radioisotopes)

Radionuclide Source Terms During LBEs: Representative Example for Prismatic Reactor Design

Source terms for a break in He pressure boundary, upper 95% confidence levels
600 MW(t) Prismatic 700°C reactor outlet temperature (Table X)



D.A. Petti et al., *Representative source terms and the influence of reactor attributes on functional containment in modular high-temperature gas-cooled reactors*, Nucl. Tech. 184 (2013) 181-197

Summary

- HTGR designs employ multiple radionuclide release barriers to meet radionuclide control requirements
- Radionuclide transport in HTGRs has been extensively investigated
- Design methods available to predict performance of the radionuclide release barriers during normal operation and accidents
  - Codes have been used extensively for reactor design and analysis, including operating HTGRs
- Many comparisons of code predictions with data
  - Reactor surveillance, in-pile tests, etc.
  - Codes not completely verified and validated
  - Comparisons are often with data from integral release tests; validation of contributions from specific mechanisms is difficult.
- Additional data from ongoing programs (e.g., US DOE AGR program) will help refine transport parameters and reduce uncertainties
- Contemporary analyses indicate that radionuclide releases during accidents are within acceptable regulatory limits

Suggested Reading

- 2010 HTGR Technology Course for the Nuclear Regulatory Commission
- A Review of Radionuclide Release from HTGR Cores During Normal Operation, EPRI report 1009382 (2003)
- D.A. Petti et al., Representative source terms and the influence of reactor attributes on function containment in modular high-temperature gas-cooled reactors, Nucl. Tech. 184 (2013) 181-197
- Fuel performance and fission product behavior in gas cooled reactors, IAEA, TECDOC-978 (1997)
- High Temperature Gas Cooled Reactor Fuels and Materials, IAEA, TECDOC-1645 (2010)
- Advances in High Temperature Gas Cooled Reactor Fuel Technology, IAEA, TECDOC-1674 (2012)

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Modular High Temperature Gas-cooled Reactor: Licensing Experience

Jim Kinsey

Modular High Temperature Gas-cooled Reactor: Licensing Experience

Advanced Reactor Technologies

Idaho National Laboratory

Jim Kinsey

INL Regulatory Affairs

NRC HTGR Training July 16-17, 2019

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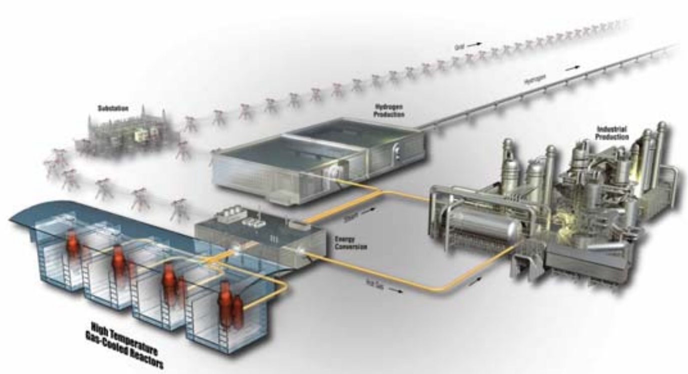
Past U.S. HTGRs and Licensing Approaches



US HTGR Licensing History

US Program	Licensing Period	Organization	Stage
Peach Bottom-1	1958 – 1966	PECO	OL Issued Decommissioned
Ft. St. Vrain (Prismatic)	1966 – 1972	PS Colo.	OL Issued Decommissioned
Summit (Prismatic)	1972 – 1975	GA	CP-LWA Submitted
MHTGR (Prismatic)	1986 – 1995	DOE/GA	Pre-App Review
Exelon DC (Pebble)	2001 – 2002	Exelon	Pre-App Review
PBMR DC (Pebble)	2006 – current	PBMR (Pty.) Ltd	Pre-App Review
NGNP (Prismatic/Pebble)	2009 – current	DOE	Pre-App Review

Peach Bottom Experience (1966-1974)

- Peach Bottom 1 – very successful - 40 MW(e)
  - Demonstrated variety of nuclear industry performance records
  - Average gross efficiency - 37.2%
  - Availability - 85%
  - No steam generator tube failures
  - Operator doses less than 10 man-rem/year
  - Load following demonstrated
  - Post examination of materials performed
- Lessons learned
  - Fuel element and coated particle design improvements

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Fort St. Vrain Experience (1976-1989)

- Demonstrated excellent fuel performance, low operator doses, and core physics
- Demonstrated fuel handling / refueling approach
- Lessons learned
  - Helium circulator and seals leaked bearing water
  - Water cooling pump cavitation
  - Reserve shutdown malfunction
  - Hot helium bypass on control rod drives
  - Core thermal fluctuations
  - Core support floor - liner cooling system

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General Atomics (GA) Modular HTGR: Pre-Application

- After Peach Bottom-1 and Ft. St. Vrain, the next major HTGR licensing effort was associated with the General Atomics Modular HTGR
- The GA design and licensing effort was based on a functional performance approach and included a number of key concepts that are similar to the current designs that are being refined and would likely be coming to NRC for review, such as:
  - Utilize inherent material properties
    - Helium coolant – neutronically transparent, chemically inert, low heat capacity, single phase
    - Ceramic coated fuel – high temp capability, high radionuclide retention
    - Graphite moderator – high temp stability, large heat capacity, long response times
  - Develop simple modular reactor design with passive safety
    - Retain radionuclides at their source within the fuel
    - Configure and size reactor for passive core heat removal from reactor vessel with or without forced or natural circulation of pressurized or depressurized helium primary coolant
    - Large negative temperature coefficient for intrinsic reactor shutdown
    - No reliance on AC-power
    - No reliance on operator action and insensitive to incorrect operator actions

GA MHTGR Policy and Licensability Issues

- Key Policy and Licensability Issues are summarized in NRC's Pre-Application Safety Evaluation Report for the MHTGR (NUREG 1338) and include:
  - Fuel Performance
  - Fission Product Transport
  - Source Term
  - "Unconventional" Containment
  - Accident Selection and Evaluation
  - Safety Classification of Structures, Systems, and Components
  - Emergency Planning

Identification of Key Policy Issues

- Key issues for modular HTGRs have been consistently confirmed:
  - MHTGR (NRC Draft SER NUREG-1338, 1989 and 1995)
  - Exelon PBMR licensing activities (2001)
  - NRC SECY documents (various, incl. 2002)
  - PBMR US design certification program (2005)
  - Jointly developed DOE-NRC licensing strategy for NGNP (2008)
  - NRC SECY 10-0034 (2010); "Policy and Technical Issues for SMRs"

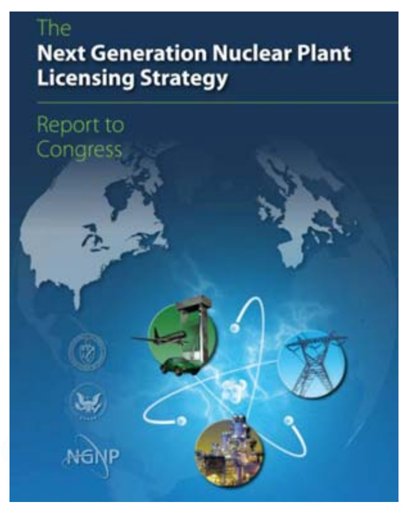
Summary of
Next Generation Nuclear Plant Experience

NGNP Next Generation Nuclear Plant

DOE-NRC Report to Congress (August 2008)

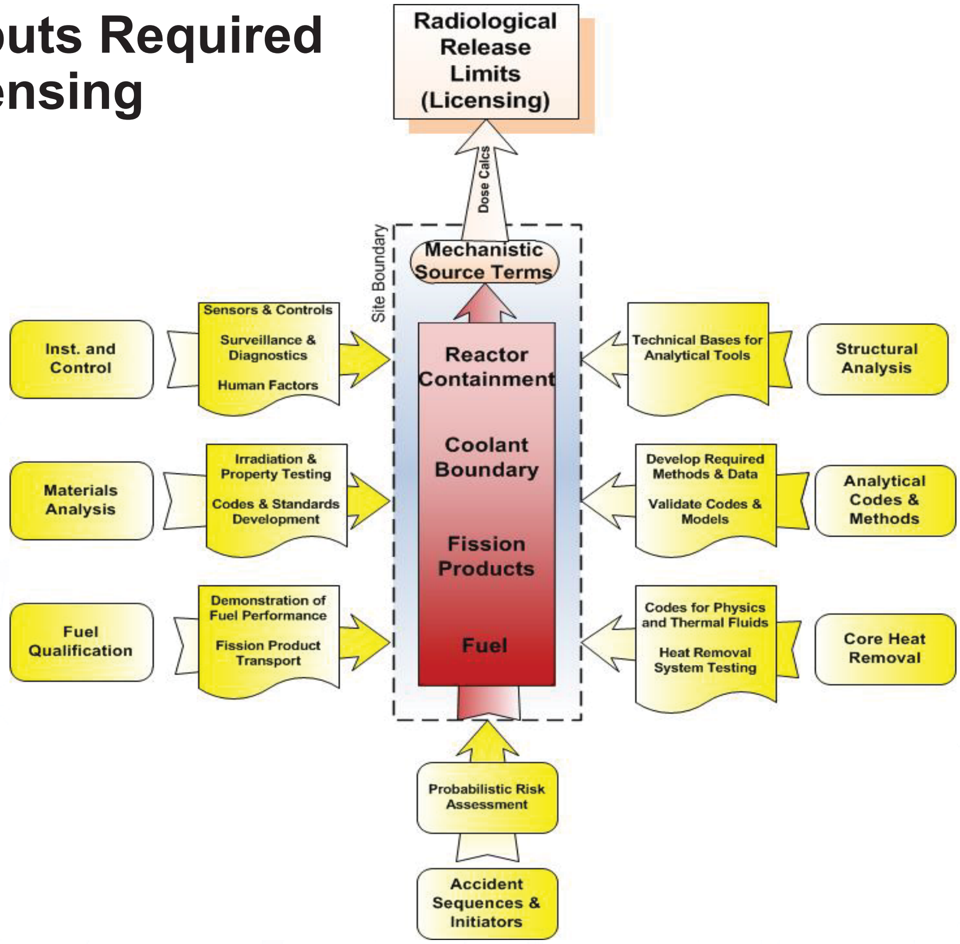
"It will be necessary to resolve the following NRC licensing technical, policy, and programmatic issues and obtain Commission decisions on these matters":

- Acceptable basis for event-specific mechanistic source term calculation, including the siting source term;
- Approach for using frequency and consequence to select licensing-basis events;
- Allowable dose consequences for the licensing-basis event categories;
- Requirements and criteria for functional performance of the NGNP containment as a radiological barrier



The best approach to establish the licensing and safety basis for the NGNP will be to develop a risk-informed and performance-based technical approach that adapts existing NRC LWR technical licensing requirements in establishing NGNP design-specific technical licensing requirements.

Key Inputs Required for Licensing



NGNP Licensing Working Group

- NGNP implemented a Licensing Working Group Concept
  - Provided a design-neutral licensing path that can be implemented by any modular HTGR design selected for deployment
  - Promoted a "single path" HTGR issue resolution efficiency for NRC
- Members included:
  - Three reactor vendors (AREVA, GA, Westinghouse/PBMR)
  - Representative owner-operator organization (Entergy)
  - INL - NGNP Research and Development
  - INL - NGNP Engineering
  - INL - NGNP Regulatory Affairs
- All NRC white paper submittals and follow-on interactions went through this process and represented the collaborative position of the domestic HTGR "fleet"

NGNP Licensing Framework Status – early 2012

- NRC issued two assessment reports providing the results of its working group review in the following areas:
  - Risk Informed Performance Based Approach to
    - Licensing Basis Event Selection
    - Classification of Structures, Systems, and Components
    - Defense in Depth
  - Fuel Qualification and Mechanistic Source Terms
- This NRC working group concluded: “No obvious fundamental issues that would prevent development of related licensing submittals that meet regulatory requirements…”
- NRC management clarified that the assessment reports reflected working group assessments that may not be consistent with broader NRC staff outputs

NGNP Letter – Requested NRC Staff Positions

- To achieve broader NRC staff feedback, NGNP submitted a request to NRC on July 6, 2012, to provide a description of the specific licensing framework topics where NRC staff positions are requested. Priority remains on the four key NGNP policy and technical focus areas:
  - Containment functional performance
  - Licensing basis event selection
  - Source terms
  - Emergency planning
- Work on TRISO particle fuel qualification topics also continued due to its safety case importance and close connection to the source term and functional containment topics

Fuel Qualification and Source Term White Papers

- Fuel Qualification White Paper – Purpose
  - Identify existing regulations, regulatory guidance, and licensing precedents relative to the qualification of fuel for NGNP
  - Review reactor and fuel designs and resulting fuel service conditions and performance requirements
  - Describe planned fuel fabrication, irradiation, testing activities
  - Obtain feedback from the NRC staff on the proposed approach to qualify the fuel

- Mechanistic Source Terms White Paper – Purpose
  - NGNP definition of event-specific mechanistic source terms for the HTGR is acceptable
  - Approach to calculating event-specific mechanistic source terms for HTGR technology is acceptable (subject to validation of the design methods and supporting data that form the bases of the calculations)
  - That the approach of planned fission product transport tests under the NGNP/AGR Fuel Development and Qualification Program, as supplemented by the existing irradiation and post-irradiation heating databases to validate these fission product transport analytical tools, is acceptable.

Fuel Qualification and Source Term Outcomes

White Paper	Submittal Date	NRC Public Meeting(s)
NGNP Fuel Qualification White Paper INL/EXT-10-18610	July 21, 2010	September 2, 2010 October 19, 2011 April 17, 2012
HTGR Mechanistic Source Terms White Paper INL/EXT-10-17997	July 21, 2010	July 24, 2012 September 20, 2012 November 14, 2012

- NGNP team responded to approx. 140 NRC RAIs
- Fuel Qualification
  - NRC's NGNP assessment concluded that the fuel qualification approach was generally reasonable, with certain caveats and open issues to be addressed
  - Advanced Gas Reactor (AGR) Program still ongoing
  - EPRI topical report planned for 2019 submittal requesting formal NRC review
- Source Terms
  - NRC's NGNP assessment determined that the proposed event-specific mechanistic approach is reasonable, but remains subject to resolution of several follow up items. Expected to be resolved as the AGR Program and HTGR design efforts proceed to completion
  - NRC has more recently issued SECY-16-0012, expanding and clarifying the use of the mechanistic source term approach to various advanced non-LWR designs

NRC Approval of the NGNP QAPD

- NGNP's Quality Assurance Program Description (QAPD) was submitted to the NRC for review
  - Original submittal in August 2010
  - Updated submittal in May 2011
  - NGNP then engaged in a series of follow-on discussions and provided written responses to NRC questions during the review
- NRC provided its approval of the QAPD for use in NGNP technology development and high level design activities (September 2012)
  - Approval assures that data and insights gained from currently ongoing R&D activities (particularly the AGR Fuel Qualification Program) can later be used directly by designers and license applicants

Note: The NGNP program structure and submittal was the first in the nuclear industry to utilize the NRC-endorsed guidance of American Society of Mechanical Engineers Standard NQA-1-2008, with 2009 addenda

Risk-Informed and Performance Based (RIPB) Approach to Event Identification and Evaluation

- The RIPB approach was developed and proposed through a series of four NGNP white papers:
  - Licensing Basis Event (LBE) Selection
  - Classification of Structures, Systems, and Components
  - Defense-In-Depth
  - Probabilistic Risk Assessment
- The bulk of NRC interactions were associated with the LBE Selection paper, with a focus around establishing a well-defined process for selecting LBEs, including:
  - Establishing acceptable limits on event sequence consequences,
  - Establishing the kinds of events, failures, and natural phenomena to be evaluated during the analysis
  - Identification of the design basis accidents to be included in Chapter 15 of the safety analysis

RIPB Interactions with NRC

White Paper	Submittal Date	NRC Public Meeting(s)
NGNP Defense-in-Depth Approach INL/EXT-09-17139	December 9, 2009	March 8, 2010
NGNP Licensing Basis Event Selection White Paper INL/EXT-10-19521	September 16, 2010	November 2, 2010 April 16, 2012 May 16, 2012 July 10, 2012 August 22, 2012 September 19, 2012 November 14, 2012
NGNP Structures, Systems, and Components Safety Classification White Paper INL/EXT-10-19509	September 21, 2010	November 2, 2010 July 10, 2012 September 6, 2012
NGNP Probabilistic Risk Assessment White Paper INL/EXT-11-21270	September 20, 2011	April 12, 2012 September 19, 2012

Emergency Planning Interactions with NRC

White Paper	Submittal Date	NRC Public Meeting(s)
Determining the Appropriate EPZ Size and Emergency Planning Attributes for an HTGR INL/MIS-10-19799	October 28, 2010	January 26, 2011 November 14, 2012

- NGNP proposed a consequence-based approach to emergency planning
- NGNP's proposal was later followed by similar inputs from NEI on behalf of the broader advanced reactor community
- In response, NRC issued SECY-11-0152 outlining high level guidance for moving forward with the proposed approaches

Licensing Interactions – Other Topics

White Paper	Submittal Date	NRC Public Meeting(s)
High Temperature Materials White Paper INL/EXT-09-17187	June 25, 2010	September 1, 2010
Licensing Structure for Multi- Module Facilities INL/EXT-10-18178	August 10, 2010	None
NGNP Nuclear-Industrial Facility and Design Certification Boundaries INL/EXT-11-21605	July 22, 2011	None

NRC Feedback – Assessment Outputs – Other Topics

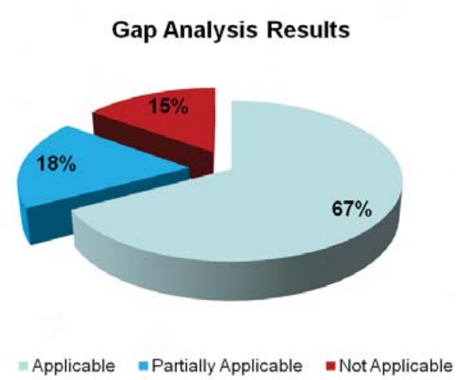
- "High Temperature Materials White Paper"
  - Principal materials proposed for NGNP primary systems were identified with approaches for regulatory compliance
  - NGNP responded to 108 NRC RAI's and NRC then issued assessment report (May, 2012)
  - NRC staff further stated an intention to not provide final conclusions regarding the design and qualification of any NGNP components, materials, or their use in the plant design, until such time as an NGNP COLA or DC application is submitted
  - White paper was updated to reflect results of NRC interactions and re-issued in August 2012
- "License Structure for Multi-Module Facilities White Paper"
  - Described the NGNP proposal regarding multi-module HTGR plant licensing with a single NRC review, hearing, and safety evaluation report
  - In response to the NGNP white paper and other related industry initiatives, the NRC issued SECY-11-0079, "License Structure for Multi-Module Facilities Related to Small Modular Nuclear Power Reactors"
- "NGNP Nuclear-Industrial Facility and Design Certification Boundaries White Paper"
  - Proposed to establish agreement regarding the boundary between a nuclear facility under NRC regulatory jurisdiction (i.e., within the scope of the DC and COLA) and the interface to energy end use facility(s) that fall outside the scope of nominal NRC authority (i.e., the industrial facility)
  - Not reviewed by NRC due to resource limitations (agreed by NGNP)

Licensing Framework Interactions with NRC (cont.)

White Paper	Submittal Date	NRC Public Meeting(s)
Modular HTGR Safety Basis and Approach INL/EXT-11-22708 (submitted for information only)	September 6, 2011	None

NGNP Regulatory Gap Analysis

- Evaluated ~2,600 individual regulatory requirements and regulatory guidance positions for applicability to modular HTGRs
- Identified 15 existing regulations that would need to be modified or otherwise addressed for HTGRs
  - 10 CFR 50 Appendix I which addresses ALARA limits for LWR effluents
  - Appendix J which describes how an LWR containment structure must be leak tested
- Confirmed overall approach of limiting rulemaking to extent possible
  - Adapt existing NRC LWR technical licensing requirements in establishing NGNP design-specific technical licensing requirements
  - NRC positions established through guides or SECY papers
- Gap analysis results summarized in INL/EXT-11-23216



NGNP Regulatory Gap Analysis (cont.)

In addition to the gap analyses results summarized in the “Applicable,” “Partially Applicable,” and “Not Applicable” categories, the analysis also identified unique modular HTGR topics that would require additional consideration, including:

- HTGR Fuel Design and Qualification
- High Temperature Ceramic Materials and Composites
- Functional Containment of Radionuclides
- Establishment of Risk Metrics (alternative to CDF and LERF)
- Passive Safety System Performance Requirements
- Helium Leak Detection
- Accident Analysis
- Classification of Structures, Systems, and Components

Summary of NRC Interactions

- DOE and NRC efforts on NGNP were aligned with the jointly developed NGNP Licensing Strategy (2008 Report to Congress)
- First phase of NRC interactions occurred late 2008 through late 2011
  - NRC working group assessment issued early 2012
- Energy Secretary suspended design/deployment efforts in October 2011, but directed that R&D and regulatory framework development should continue
- Second phase of NRC interaction focused on agreed upon priority Commission policy topics and TRISO particle fuel qualification, and resulted in August 2014 NRC Assessment Report
  - Also see draft report provided to ACRS in March 2013
- Administrative Information:
  - Assigned NRC Project (Docket) number for NGNP is: PROJ0748
  - NGNP submitted a total of 11 white papers, and responded to approximately 450 RAIs
  - There were approximately 30 public meetings associated with the NRC Staff’s review of NGNP proposals

Suggested Reading

"NRC Licensing Status Summary Report for NGNP," Rev. 1, INL/EXT-13-28205 (Nov. 2014) https://www.osti.gov/biblio/1236815-nrc-licensing-status-summary-report-ngnp


INL/EXT-13-28205
Revision 1

NRC Licensing Status Summary Report for NGNP

November 2014

The INL is a U.S. Department of Energy National Laboratory operated by Battelle Energy Alliance

Other Key Outcomes Since NGNP Affecting HTGRs

- ARDC Reg. Guide 1.232 for developing principal design criteria
  - Result of DOE-NRC joint initiative
- SECY 18-0096 and SRM on Functional Containment Performance Requirements
- Ongoing Emergency Planning Rulemaking
- NRC Guidance on Prototype Reactors

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High Temperature Gas-cooled Reactor: Materials

Richard Wright

Structural Alloys for HTGR and VHTR Systems

Advanced Reactor Technologies

Idaho National Laboratory

Richard Wright, Emeritus Laboratory Fellow

NRC HTGR Training July 16-17, 2019

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ADVANCED REACTOR TECHNOLOGIES

ASME Section III Division 5 Framework for Component Design (Part I)

- Section III Division 1 rules cover light water reactor systems
  - These rules do not allow time dependent deformation
  - Upper temperature limit for ferritic materials is 375°C and for austenitic materials is 425°C
- Section III Division 5 "Rules for Construction of Nuclear Facility Components – High Temperature Reactors" has replaced Section III Division 1 for construction of high temperature reactors
  - Section III Division 1 Subsection NH was first included in the 1995 with 1996 and 1997 Addenda version of the ASME BPVC Code.
  - Section III Division 5 was added in the 2010 with 2011 Addenda version of the Code and considered separate from Section III Division 1 Subsection NH
  - ASME BPVC 2017 is the first version of the code to come without Section III Division 1 Subsection NH
- These rules are applicable to high temperature reactor systems, including HTGR, LMR and MSR
  - ASME BPVC does not consider environment effects for metals
  - For example, Alloy 617 contains up to 15% Co and would not be appropriate in a neutron environment, but the Code would not specifically prohibit it. (Note Alloy 617 is being explored for use in the secondary heat exchanger. As such, it will not experience neutron radiation and the cobalt level is not a concern)

ASME Boiler and Pressure Vessel Code, 2017 Edition, Section III Division 1 and Section III Division 5

ASME Section III Division 5 Framework for Component Design (Part II)

- Only five alloys are allowed for nuclear components under these rules:
  - 2.25Cr-1Mo and V modified 9Cr-1Mo ferritic steels
  - Type 304 and Type 316H stainless steels and Alloy 800H
  - Sixth alloy, Inconel 617, is under review

Material	Fe	Ni	Cr	Co	Mo	Al	C	Mn	Si	S	Ti	Cu	B	P	V	N	Nb
304/304H	Bal	8.0-10.5	18.0-20.0		-	-	0.04-0.08/0.10	2.0 max	0.75 max	0.03 max	-	-	-	0.045 max	-	0.10 max	-
316/316H	Bal	10.0-14.0	16.0-18.0	-	2.0-3.0	-	0.04-0.08/0.10	2.0 max/ 0.04-0.10	0.75 max	0.03 max	-	-	-	0.045 max	-	0.10 max	-
800H	39.5 min	30.0-35.0	19.0-23.0	-	-	0.15-0.60	0.05-0.10	-	-	-	0.15-0.60	-	-	-	-	-	-
2.25Cr-1Mo	Bal	-	2.0-2.5	-	0.90-1.1	-	0.07-0.15	0.30-0.60	0.50 max	0.025 max	-	-	-	0.025 max	-	-	-
9Cr-1Mo-V	Bal	0.40 max	8.0-9.5	-	0.85-1.05	0.04 max	0.08-0.12	0.30-0.60	0.20-0.50	0.010 max	-	-	-	0.020 max	0.18-0.25	0.30-0.70	0.06-0.10
617	3.0 max	44.5 min	20.0-24.0	10.0-15.0	8.0-10.0	0.8-1.5	0.05-0.15	1.0 max	1.0 max	0.015 max	0.6 max	0.5 max	0.006 max	-	-	-	-

ASME Boiler and Pressure Vessel Code, 2017 Edition, Section III Division 1 and Section III Division 5

ASME Section III Division 5 Framework for Component Design (Part III)

- For each allowed material, limits are set for upper temperature and time, e.g., for Alloy 800H 750°C and 300,000 hours
- In addition to time dependent deformation, design rules accounting for creep-fatigue are incorporated
  - The creep-fatigue interaction model takes into account the deleterious effects of creep and fatigue together
  - If creep and fatigue were solely considered separately, design models would be non-conservative, as creep-fatigue interactions cause failure earlier in life than would be expected
- Note: All temperature in degrees Celsius are rounded off per ASME metric convention. Maximum use temperature are expressed in degrees Fahrenheit in Division 5

ASME Boiler and Pressure Vessel Code, 2017 Edition, Section III Division 1 and Section III Division 5

ASME Section III Division 5 Framework for Component Design (Part IV)

- Material classes allowed in Subsection HA, and max temperature allowed (T_{max})

Materials	T_{max}, °F (°C)
Carbon steel	700 (370)
Low alloy steel	700 (370)
Martensitic stainless steel	700 (370)
Austenitic stainless steel	800 (425)
Nickel-chromium-iron	800 (425)
Nickel-copper	800 (425)

- When safety-related components exceed the appropriate temperature limits from Subsection HA, then Subsection HB is used

Materials	Temp. not exceeding, °F (°C)
304 SS	1500 (816)
316 SS	1500 (816)
800H	1400 (760)
2.25Cr-1Mo	1100 (593)
9Cr-1Mo-V	1200 (650)

ASME Boiler and Pressure Vessel Code, 2017 Edition, Section III Division 1 and Section III Division 5

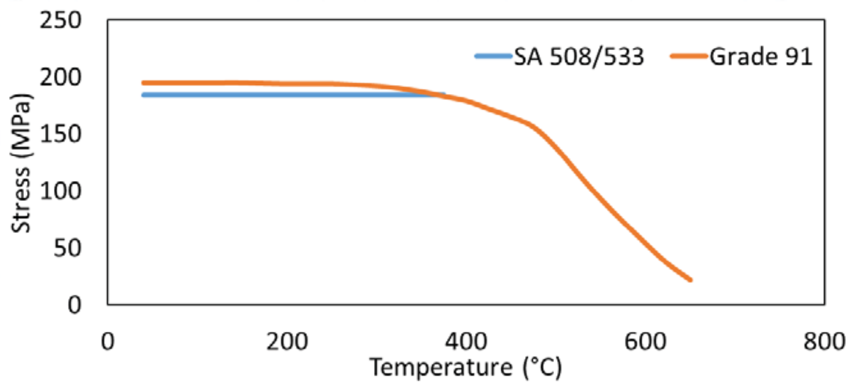
Overview of Pressure Vessel Steels (Part I)

- VHTR pressure vessels tend to be large – 600MW thermal design concept specified 8m diameter and 250 mm thickness
- VHTR goal outlet temperatures between 700-950 °C
- Conventional SA 508 (forging grade) and SA 533 (rolled product form) low alloy bainitic steel commonly used in light water reactors can be used if the vessel temperature is held to 370°C or less
  - Mandatory Appendix HBB-II (Of Section III Division 5) allows for use of these steels and their weldments for Class A nuclear components with metal temperatures above 370°C during operating conditions associated with Level B (upset), C (emergency) and D (faulted) service limits
  - Temperature shall not exceed 425°C for Level B and 540°C for Level C and D
  - Component design shall be based on a maximum cumulative time of 3,000 hr at metal temperatures above 370°C

ASME Boiler and Pressure Vessel Code, 2017 Edition, Section III Division 5
Next Generation Nuclear Plant Reactor Pressure Vessel Materials Research and Development Plan INL Document PLN-2803, Rev. 1, 2010
R. Wright, 2014, Creep of A508/A533 Pressure Vessel Steel, INL/EXT-14-32811

Overview of Pressure Vessel Steels (Part II)

- V modified 9Cr-1Mo (Grade 91) steel could be used at higher temperature and is allowed in Section III Division 5 for elevated temperature design
- Grade 91 steel has been considered for use in new French and Japanese fast reactor applications and widely used in tubing in fossil plants; there is currently no capacity to melt or forge sections sizes typical of VHTR vessels



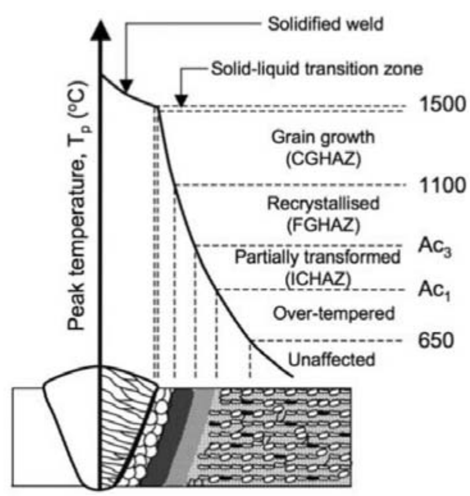
Allowable Stress Intensity Values (100,000 hr life for Grade 91 which affects values above 475°C)

ASME Boiler and Pressure Vessel Code, 2017 Edition, Section II Part D and Section III Division 5

Next Generation Nuclear Plant Reactor Pressure Vessel Materials Research and Development Plan INL Document PLN-2803, Rev. 1, 2010

Overview of Pressure Vessel Steels (Part III)

- Grade 91 steel is susceptible to Type IV cracking in the heat affected zone of the base metal, above a certain temperature where creep damage occurs
  - This is a form of creep cracking in fine grained recrystallized material in the base metal adjacent to welds (HAZ)
  - Cracks form from creep damage and can be rapid as the crack links voids from creep damage



Schematic of welded ferritic alloys like Grade 91. Type IV cracking occurs in the FGHAZ region.

Francis, et al, 2006, Mat. Sci. and Tech. Vol. 22 No. 12, pp. 1387-1395

Next Generation Nuclear Plant Reactor Pressure Vessel Materials Research and Development Plan INL Document PLN-2803, Rev. 1, 2010

Overview of Pressure Vessel Steels (Part IV)

- Type 4 cracking can only be avoided by a re-normalizing heat treatment after welding or by reducing the temperature of the vessel below the creep regime
- Properties of Grade 91 steel are very sensitive to austenitizing temperature and subsequent tempering treatment; there is currently no NDE method that can assure proper heat treatment was achieved through-thickness in heavy sections
- The US VHTR program made the determination that use of conventional steels (SA 508 and SA 533) was the only feasible near term option

Next Generation Nuclear Plant Reactor Pressure Vessel Materials Research and Development Plan INL Document PLN-2803, Rev. 1, 2010

Considerations for SA 508 and SA 533 Pressure Vessel Steels

- Conceptual designs for a 600MW thermal reactor specified SA 508 and/or SA 533B steel and active cooling to maintain the vessel temperature below 370°C
- VHTR designs typically have very low lifetime neutron fluence on the vessel; the US reference design concluded less than 1dpa for a sixty year lifetime
- ASME Code rules exist for welding and inspection of heavy section vessels using these steels
- Properties of these materials are largely insensitive to heat treatment and welding; there is a large base of experience resulting from use in light water reactor systems
- For passive cooling of VHTR systems the emissivity of the vessel needs to be high and stable over long operating periods; the native oxide on these steels has been shown to be adequate for passive cooling from accident conditions that have been considered
- Capacity for forging and rolling required section sizes is available for these steels in Japan, Korea and France
- The gas flow path designed to maintain the vessel temperature in the acceptable range on the internal surface of the vessel is defined by the core barrel; Type 316H stainless steel is adequate for the core barrel application

ASME Boiler and Pressure Vessel Code, 2017 Edition, Section IX Welding and Section XI Non-Destructive Examination

Materials Issues for Steam Generator and Heat Exchanger Applications

- The Fe-Ni-Cr material Alloy 800H is fully Section III Division 5 Code qualified for use up to 750°C and times up to 300,000 hours
- Alloy 800H has adequate properties for proposed VHTR steam generator tubes up to the maximum Code qualification temperature
- Above 750°C for gas-to-gas heat exchangers an additional material – Ni-Cr-Co-Mo Alloy 617 is currently being Code qualified
- The Alloy 617 Code Case is for an upper temperature limit of 950°C and time of 100,000 hours
- Both Alloy 800H and Alloy 617 were extensively characterized for the gas reactor programs in Germany, Japan and the US in the 1970s and 1980s
- Alloy 800H was used in the steam generator of the German pebble bed demonstration reactors and in the US Fort St. Vrain plant
- Additional alloys – Hastelloy X and Haynes 230 have been considered for high temperature structural applications, but neither was judged by the US program to have sufficient technical maturity and creep properties to proceed with Code qualification
- The Japanese demonstration reactor has used a modified Hastelloy X in the cross-duct and heat exchanger; this alloy is little known in the US and is not Code qualified

Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan INL Document PLN-2804, Rev. 1, 2010
J. Wright, 2015, Draft ASME Boiler and Pressure Vessel Code Case for Use of Alloy 617 for Class A Elevated Temperature Service Construction, INL/EXT-15-36305

ASME Code Qualification

- Higher temperature design of VHTR systems might require structural alloys with elevated temperature properties exceeding those of the five Code qualified alloys; new materials would need to be qualified
- Section III Division 5, Appendix HBB-Y, "Guidelines for Design Data Needs for New Materials" describes required properties
  - Technical basis established through DOE Advanced Reactor Technology base program on the Alloy 617 Code Case in support of HTGR/VHTR applications

Required testing to introduce a new structural material into Section III, Division 5, or a Division 5 Code Case

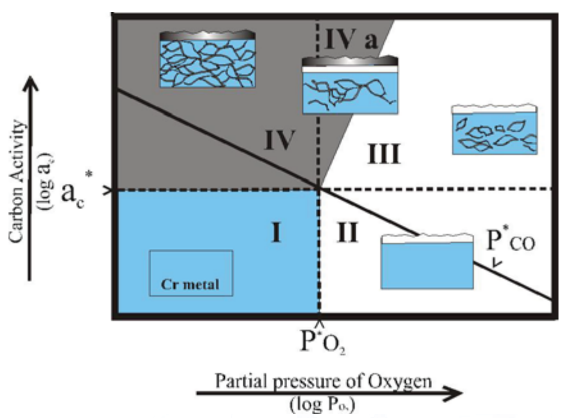
- HBB-Y-2100 Requirement For Time-independent Data
- HBB-Y-2110 Data Requirement for Tensile Reduction Factors for Aging
- HBB-Y-2200 Requirement for Time-Dependent Data
- HBB-Y-2300 Data Requirement for Weldments
- HBB-Y-3100 Data Requirement for Isochronous Stress-Strain Curves
- HBB-Y-3200 Data Requirement for Relaxation Strength
- HBB-Y-3300 Data Requirement for Creep-Fatigue
- HBB-Y-3400 Data Requirement for Creep-Fatigue of Weldments
- HBB-Y-3500 Data Requirement for Cyclic Stress-Strain Curves
- HBB-Y-3600 Data Requirement for Inelastic Constitutive Model
- HBB-Y-3700 Data requirement for Huddleston multiaxial failure criterion
- HBB-Y-3800 Data Requirement for Time-Temperature Limits for External Pressure Charts
- HBB-Y-4100 Data Requirement for Cold Forming Limits
- Validation of Elastic-Perfectly Plastic (EPP) Simplified Design Methods for the new alloy

Welding, Diffusion Bonding, Aging and Cold Work

- Gas-tungsten arc welding (GTAW) and submerged arc welding processes (including weld process qualification and qualified filler metals) and inspection requirements are incorporated in the ASME Code for pressure vessel steels and Alloy 800H
- Only GTAW welding is included in the Alloy 617 Code Case currently in the approval process
- Weld strength reduction factors are specified in Section III Division 5 and are applied to creep rupture properties as specified in appropriate sections of the design rules
- Diffusion bonding has been proposed for fabrication of compact heat exchangers for VHTR use – this process is not approved in Section III Division 5 for nuclear construction, though a DOE-NE IRP project is developing Division 5 construction rules for compact heat exchangers
- Reduction factors on the tensile properties are required for some Section III Division 5 materials to be used in seismic analysis of components after long time aging in service; where those factors are required they are specified in appropriate sections of the design rules
- Since VHTR components are expected to experience long-time, elevated temperature service cold worked materials are generally not allowed for the Section III Division 5 materials
- Up to 5% incidental cold work associated with fit-up strain is typically allowed

ASME Boiler and Pressure Vessel Code, 2017 Edition, Section III Division 1 and Section III Division 5, Section IX Welding and Section XI Non-Destructive Examination

Aging and Environmental Effects



Assessments of Inconel 617 stability at various gas concentrations. Five conditions are represented:

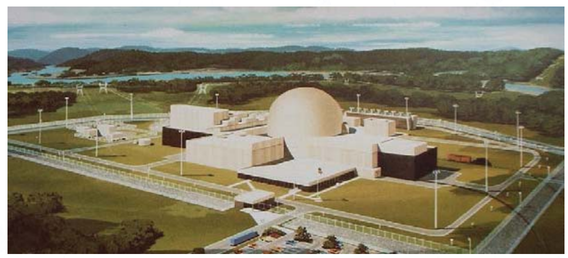
I. Reducing
II. Oxidizing
III. Stable external oxide with stable internal carbides
IV. Strongly carburizing internally and externally
IVa. Strong external carburization with stable oxide layer

- There is no environment that is inert with respect to the alloys; oxidation or carburization will always occur to some extent depending on the coolant gas chemistry and temperature
- Environmental effects maps will help in specification of He impurity content of primary coolant for long-term stability of heat exchangers
- A slightly oxidizing gas chemistry is preferred (region II in the figure); the protective oxide scale prevents either rapid oxidation or carburization
- The large volume of graphite was shown in the German AVR demonstration reactor to provide a chemical buffer on the coolant such that the preferred impurity content was maintained
- The mechanical properties of Alloys 800H and 617 are not significantly affected by long-term exposure to typical VHTR gas chemistry

C. Cabet, A. Mannier, and A. Terlain, 2004, "Corrosion of High Temperature Alloys in the Coolant Helium of a Gas Cooled Reactor," *Materials Science Forum,* Vols. 461-464, pp. 1165-1172.

Issues identified in NRC Assessment of the Clinch River Breeder Reactor

- Nine areas of concern were identified in the NRC assessment of the Clinch River Breeder Reactor in the late 70's and early 80's that are still under evaluation for elevated temperature components:
  - Weldment cracking
  - Notch weakening
  - Materials property representation for inelastic analysis
  - Steam generator tubesheet evaluation
  - Elevated temperature seismic effects
  - Elastic follow-up in piping
  - Creep-fatigue evaluation
  - Plastic strain concentration factors
  - Intermediate piping transition weld



NUREG-0968, Vol. 1 Main Report, "Safety Evaluation Report Related to the Construction of the Clinch River Breeder Reactor Plant," March 1983, ML082380939

NRC Evaluation of High Temperature Power Reactors

- In the 90's, the NRC sponsored a reevaluation of the design issues for high temperature reactors
- 23 issues needed to be resolved, most importantly
  - Lack of material property allowable design data/curves for 60 year design life
  - Degradation of material properties at high temperatures due to long-term irradiation
  - Degradation of material properties due to corrosion phenomena
  - Lack of validated thermal striping materials and design methodology
  - Lack of reliable creep-fatigue design rules
  - Lack of validated weldment design methodology
  - Lack of flaw assessment procedures
  - Lack of understanding/validation of notch weakening effects
  - Lack of validated rules/guidelines to account for seismic effects at elevated temperatures
  - Lack of inelastic design procedures for piping

W. O'Donnell, A. Hull, S. Malik, 2008, "Structural Integrity Code and Regulatory Issues in the Design of High Temperature Reactors," Proceedings of the 4th International Topical Meeting on High Temperature Reactor Technology

Further Review of High Temperature Reactor Regulator Requirements

- Mid 2000's, NRC updated the licensing needs for next generation power plants
  - General issues related to high temperature stability
  - Ability to withstand service conditions
  - Long-term thermal aging
  - Environmental degradation (impure helium)
  - Issues associated with fabrication and heavy-section properties
  - Further development of Section III of the ASME code needed (for higher temperatures – up to at least 900oC), including Alloy 617 and Hastelloy X
  - Creep behavior models and constitutive relations are needed for cyclic creep loading
  - Models must account for the interaction between the time independent and time dependent material response

W. O'Donnell, A. Hull, S. Malik, 2008, "Structural Integrity Code and Regulatory Issues in the Design of High Temperature Reactors," Proceedings of the 4th International Topical Meeting on High Temperature Reactor Technology

Phenomena Identification and Ranking Tables (PIRT)

- Safety relevant phenomena were considered for potential degradation concerns and ranked according to importance and current state of knowledge
- High temperature structural materials issues were evaluated for major structural components such as the reactor pressure vessel, control rods, reactor internals, primary circuit components, heat exchangers, etc.
- The PIRT was created as there are major design changes for high temperature reactors from the current LWR reactors and both the industry and NRC have very little experience with HTGRs (there is very little existing data)
- 58 phenomena were identified, with 17 of high importance and low/medium state of knowledge

Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs}, Volume 4: High-Temperature Materials PIRTs, NUREG/CR-6944, Vol. 4, ORNL/TM-2007/147, Vol. 4, March 2008

Suggested Reading

- ASME Boiler and Pressure Vessel Code, 2017 Edition, Section III Division 1 and Section III Division 5, Section IX Welding and Section XI Non-Destructive Examination.
- C. Cabet, A. Mannier, and A. Terlain, 2004, “Corrosion of High Temperature Alloys in the Coolant Helium of a Gas Cooled Reactor,” *Materials Science Forum,* Vols. 461-464, pp. 1165-1172.
- Next Generation Nuclear Plant Reactor Pressure Vessel Materials Research and Development Plan INL Document PLN-2803, Rev. 1, 2010.
- Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan INL Document PLN-2804, Rev. 1, 2010.
- R. Wright, 2014, “Creep of A508/533 Pressure Vessel Steel,” INL External Report 14-32811, Rev. 0
- X. Yan, 2016, “Very high-temperature reactor”, Handbook of Generation IV Nuclear Reactors, pp. 5-90
- NGNP High Temperature Materials White Paper INL/EXT-09-17187, Rev. 1, 2012
- W. O’Donnell, A. Hull, S. Malik, 2008, “Structural Integrity Code and Regulatory Issues in the Design of High Temperature Reactors,” Proceedings of the 4th International Topical Meeting on High Temperature Reactor Technology
- W. O’Donnell, D. Griffin, 2007, “Regulatory Safety Issues in the Structural Design Criteria of ASME Section III Subsection NH and for Very High Temperatures for VHTR & GEN IV,” Final Report for ASME Gen IV Materials Project
- NUREG/CR-6944, “Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs), Vol. 4, High Temperature Materials PIRTs,” 53 pp, (March 2008).
- J. Wright, 2015 “Draft ASME Boiler and Pressure Vessel Code Case for Use of Alloy 617 for Class A Elevated Temperature Service Construction”, INL/EXT-15-36305
- J. Wright et al, 2016 “Determination of the Creep-Fatigue Interaction Diagram for Alloy 617,” Proceedings of the ASME 2016 Pressure Vessels and Piping Conference

Comments to Address Issues from NRC Review

- Design issues, including thermal stresses, are outside of the scope of this discussion
- Inspection issues for Division 5 components are covered by existing Sections V and XI
- Properties are weldments for elevated temperature design are contained in Section III Division 5. Weld process qualification requirements are identical with Section IX
- Allowed materials for use in Section III Division 5 are currently included in Section II. New materials for Section III Division 5 may be added by a Code Case without inclusion in Section II

INL

Idaho National Laboratory

Nuclear Graphite Components

William Windes

Nuclear Graphite Components

Advanced Reactor Technologies

Idaho National Laboratory

Richard Wright, Emeritus Laboratory Fellow

Will Windes, ART Graphite R&D Technical Lead

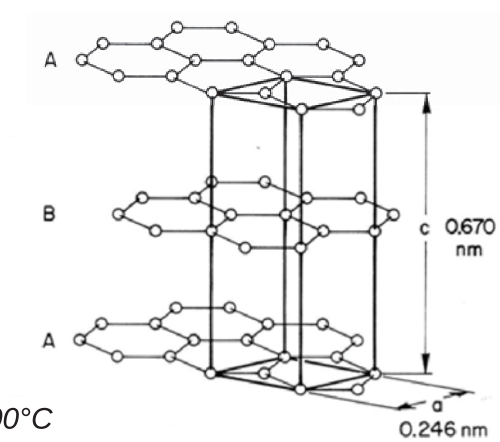
NRC HTGR Training July 16-17, 2019

Idaho National Laboratory

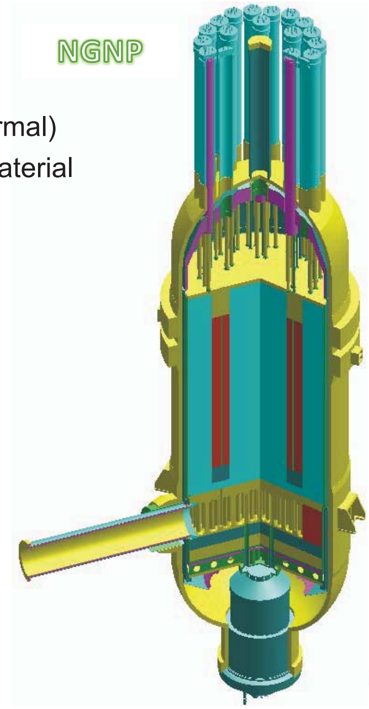
ART ADVANCED REACTOR TECHNOLOGIES

Graphite Outline

- Functions and Requirements
  - Normal and off-normal component functions
  - Key safety requirements of core components
- Graphite Manufacture
  - Unique material properties of graphite
  - Ideal unirradiated material properties – it's not metal
- Environmental effects on nuclear graphite
  - Effects of oxidation
    - ***It doesn't burn!***
  - Effects of irradiation of graphite
    - ***No Wigner*** *(stored) energy if operated above 300°C*
    - *Physical, thermal, and mechanical properties*
    - *Turnaround and creep significance explained*
- ASME Code for Graphite Core Components
  - New ASME code: probabilistic (ceramics) vs. deterministic (metals)
  - How environmental effects are accounted for in design requirements
- Operating considerations (prismatic vs. pebble vs. molten salt)
  - Differences between different graphite core designs

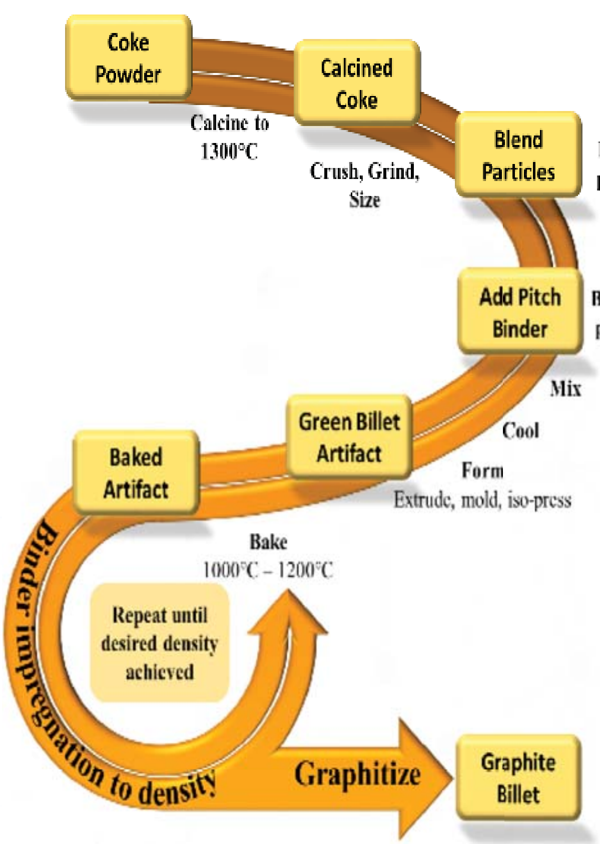


Critical Safety Requirements



- Maintain core geometry and structural integrity
  - Maintain fuel configuration during all operations (normal and off-normal)
  - Maintain undisturbed access for the insertion of reactivity control material
  - Maintain proper core coolant configuration
    - *No blockage of coolant pathway*
    - *No gaps between graphite components*
- Protection of fuel
  - Compacts within the prismatic fuel elements
  - Pebbles within the core center
- Passively remove core heat during off-normal events
  - Rapidly absorb large thermal transients
  - Primarily by radial conduction from the fuel to the core barrel
    - *During off-normal events when forced cooling is not available*
- <u>How does it do this?</u>
  - Graphite does **NOT** melt or burn
  - Graphite **DOES** have high thermal conductivity and thermal stability
  - Relatively strong in compression, weak in tension.

Graphite Manufacture



- All graphite grades **are proprietary**. Only limited/general fabrication data is known
- Unique manufacturing processes for graphite must be understood to appreciate graphite behavior
  - Graphite is a porous material (15-20%) - ***By design***!
  - Porosity provides thermal and irradiation stability
- Graphite is manufactured from calcined coke and a pitch binder.
  - Multiple pitch impregnations to increase density
- Green forming technique influences the final microstructure
  - Desire isotropic (or near isotropic) material response
- Properties and performance of graphite are significantly influenced by both raw materials and processing
  - Nuclear graphite undergoes further purification steps

Graphite Material Properties of Interest

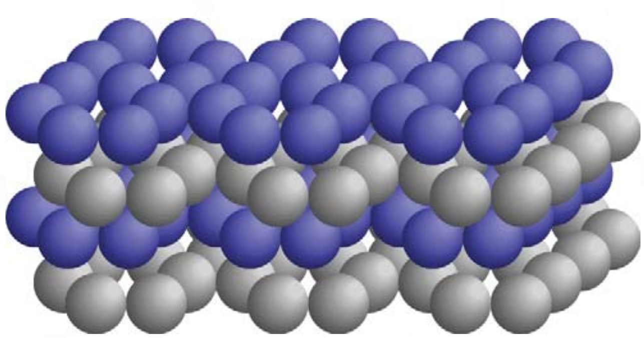
Property	Nominal Range	Performance Attributes
Density	1.7 - 1.9 g/cm^3	Neutron efficiency, Structural integrity, Thermal efficiency
Thermal Conductivity (at Room Temperature)	> 90 W/m/K	Heat transport
Purity (Total Ash Content)	< 300 ppm	Reduced component activity levels during replacement and/or disposal Reduced graphite oxidation under normal and accident conditions.
Tensile Strength	> 15 MPa	Structural integrity
Compressive Strength	> 45 MPa	Structural integrity
Flexural Strength	> 20 MPa	Structural integrity
CTE (20°C to 500°C)	3.5 to 5.5 x 10^{-6} K^{-1}	High value is indication of isotropy = dimensional stability under irradiation Lower value potentially beneficial in terms of thermal stress
CTE Isotropy Ratio	< 1.10	Irradiation dimensional stability Structural integrity
Dynamic Elastic Modulus	8 – 15 GPa	Structural integrity Irradiation creep
Dimensional Changes with Irradiation	Minimal shrinkage Minimal differences in with-grain and against-grain directions	Structural integrity (lower internal stresses)

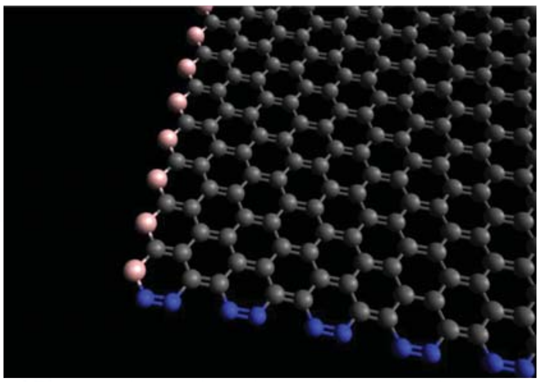
From ASTM D7219 : *Standard Specification for Isotropic and Near-isotropic Nuclear Graphites*

- **Density**
  - Higher = Stronger
  - Lower = Better irradiation performance
- **Conductivity**
  - Nearly a 70% drop almost immediately after reactor startup
- **CTE (Coefficient of Thermal Expansion)**
  - Indicates isotropy and needed for gas gap analysis
- **Purity**
  - Requires additional heat treatment
- **Dimensional changes**
  - Affects structural integrity
  - If internal stress exceeds inherent strength of graphite = *cracks*

Graphite “Burning” and dust “Explosions”

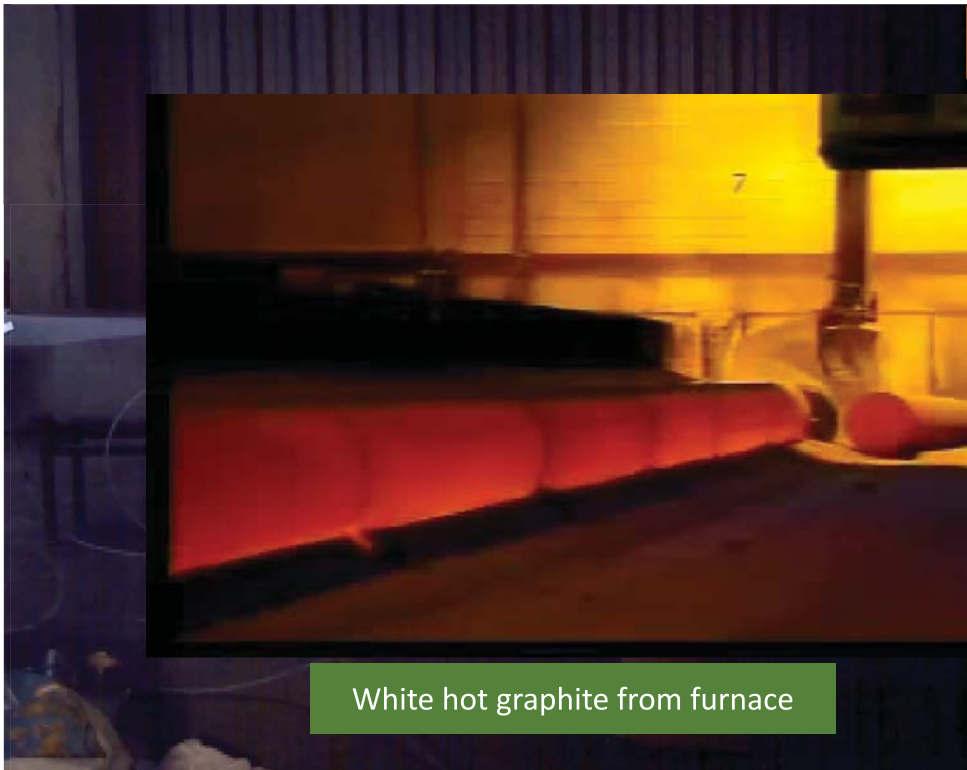
- Graphite **can not** burn – just physically can not sustain self oxidation
  - Fire needs → Heat, fuel, and oxygen
  - Fuel (carbon) is restricted to only the edges. Oxygen is restricted by the crystallography.
  - Self-sustained oxidation (better definition than simple burning) can not be sustained.





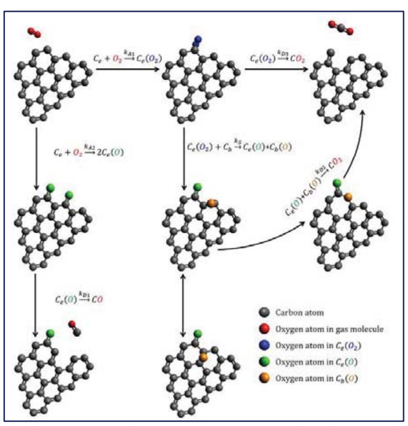
- Graphite dust **can not** explode
  - It does rapidly react but it self-suppresses. Similar mechanisms for “burning”
  - Initial flare up of surface layer on dust particles – but then nothing.
    - *No chain reaction*

Graphite “Burning” and dust “Explosions”

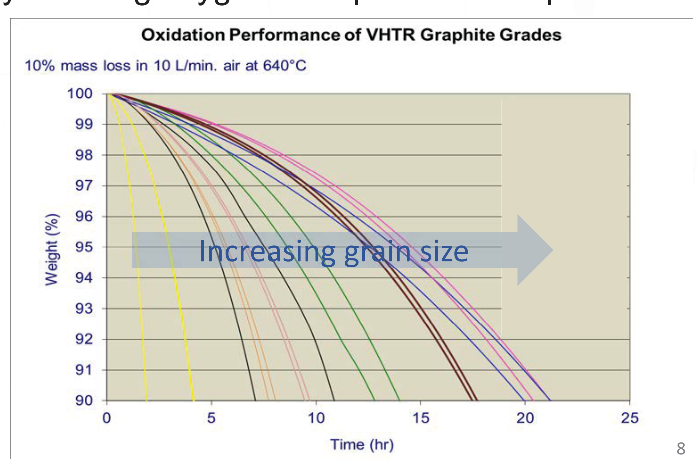


Graphite Oxidation and “Burning”

- Graphite **can** and **does** oxidize – high temperatures



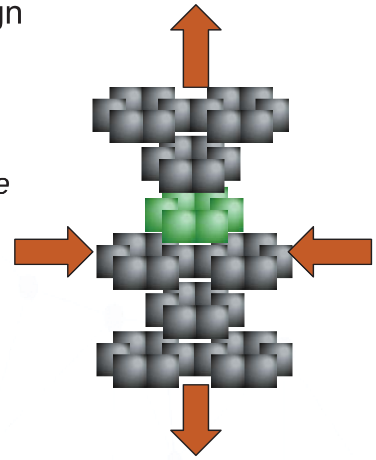
- Needs continuous oxygen and temperatures above 200°C – 300°C
  - Temperatures > 400°C needed for more rapid acute oxidation (accidents)
  - Temperatures < 400°C can still oxidize but at very slow rates (chronic oxidation)
- Oxidation still restricted to edges of crystallites with porosity dictating oxygen transport into component



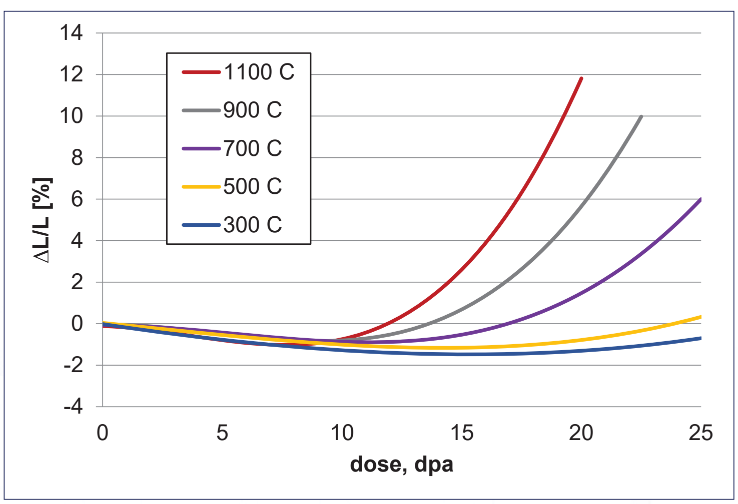
- Oxidation rates of different grades can be compared using ASTM D7542 standard, “Air Oxidation of Manufactured Carbon and Graphite in Kinetic Regime”
  - Grain size dependent
  - Oxidation of small grain grade >> than large grain size

Irradiation Effects on Graphite Properties

- Irradiation induced changes **must** be considered in design
- Significant changes occur during normal operation in:
  - Component dimensions
    - *Components actually shrink …*
    - *Until **Turnaround** when they begin to expand until failure*
  - Density
    - *Components become more dense …*
    - *After **Turnaround** dose they decrease in density*
  - Strength and modulus
    - *Graphite gets stronger with irradiation …*
    - *Until **Turnaround** dose is achieved. It then decreases*
  - Thermal conductivity
    - *Decreases almost immediately to ~30% of unirradiated values*
  - Coefficient of thermal expansion
    - *Initially increases but then reduces after **Turnaround** until saturation*
- Significant changes do not typically occur in the following properties:
  - Oxidation rate, neutron moderation, specific heat capacity, emissivity
- No Wigner energy release **if** components irradiated above 300°C.



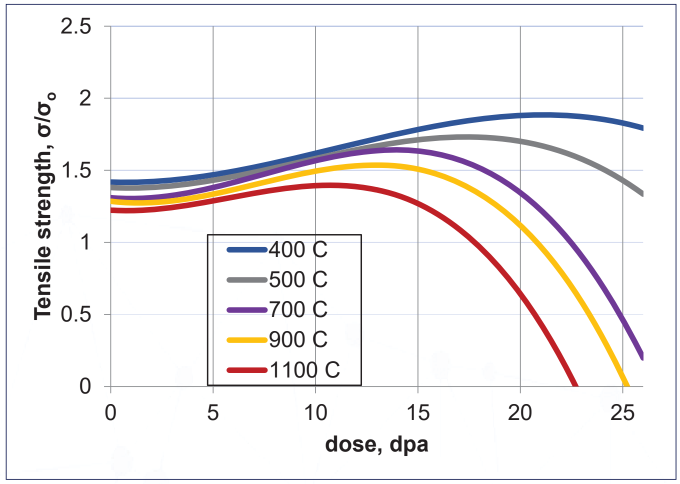
Irradiation-Induced Dimensional Changes

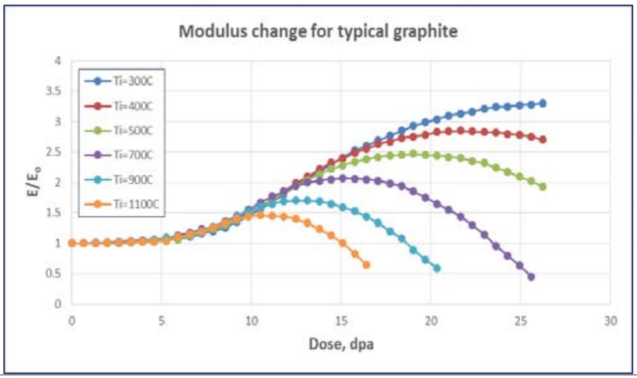


- Under neutron irradiation graphite components shrink (densify) – stop at **Turnaround** – then begin to expand (crack formation)
  - Change is dose dependent: Higher doses = larger change
  - Rate of change is highly temperature dependent
  - Rate and amount of change is grade specific
- Results in tremendous internal stresses formed within graphite
  - Crack formation and component failure – usually after Turnaround
  - Isotropic response is desired to assist in prediction of stresses and dimensional changes

Irradiation-Induced Strength/Modulus Changes

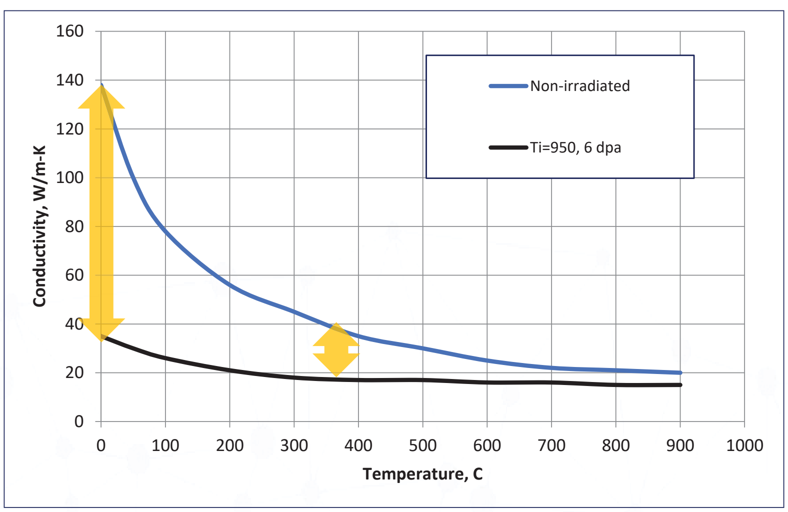
- Changes in strength and modulus somewhat parallel dimensional changes
- Strength/modulus initially increase
  - Maximum value is reached at approximately the **Turnaround** dose





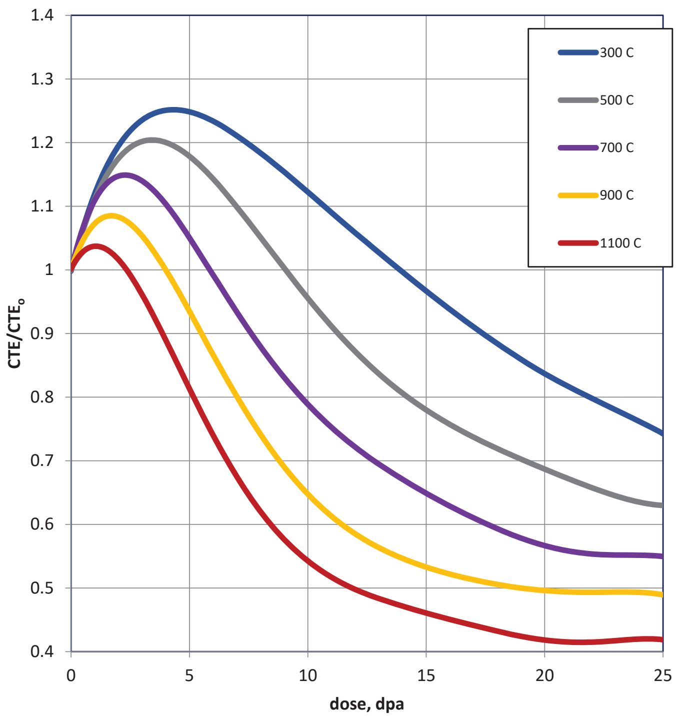
- **After Turnaround** pores start to form in microstructure
  - As porosity forms, strength and modulus fall at increasing rate
- As with dimensional changes, strong dependence on irradiation temperature

Irradiation-Induced Thermal Conductivity Changes



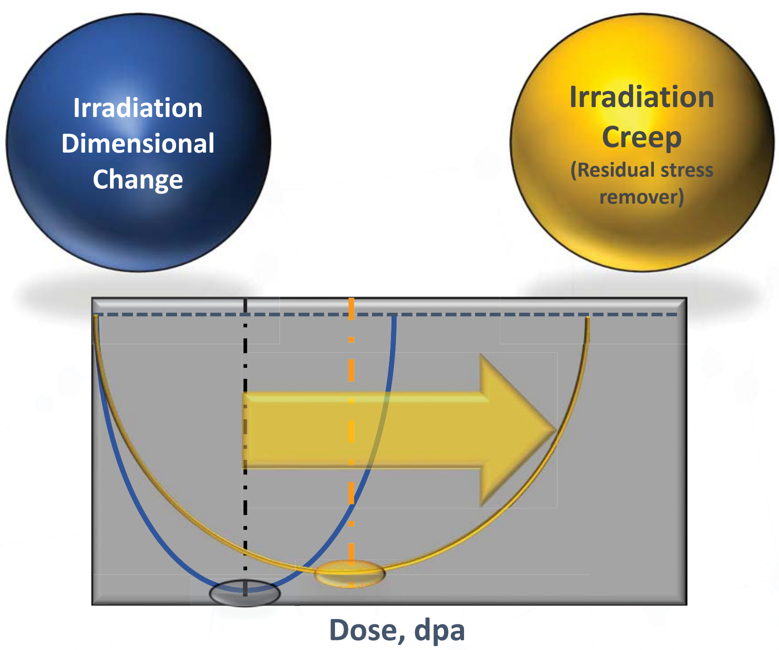
- Initial steep drop in conductivity followed by a saturation level
  - Point defects interrupt thermal diffusivity/conductance
  - Efficiency of recombination rate of point defects is dependent upon irradiation temperature = saturation
  - Further degradation of conductivity due to larger microstructure defects
    - Pore generation after turnaround
- At high operating temperatures irradiated and non-irradiated thermal diffusivity differences are small

Irradiation-Induced CTE Changes



- Overall, graphite CTE is low compared to other structural materials, e.g., metals
  - Implies excellent shock resistance
- Along with dimensional changes, must be accounted for in the design
- Initial increase with dose as manufacturing-related microcracks are closed
  - Limited dependence on **Turnaround**
- Subsequent reduction of CTE at increased dose rate

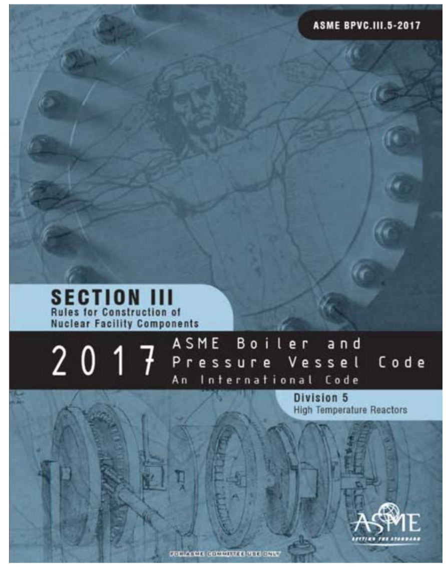
Irradiation Creep – Life Limiting Mechanism



- **Reduces internal stresses** resulting from dimensional changes
- 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, this effect becomes increasingly important in attaining acceptable design lifetimes.

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
  - First published in 2012 under 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
  - Applies probabilistic design methods
  - 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
  - Allowance of cracks in graphite components, provided that safety functions are retained



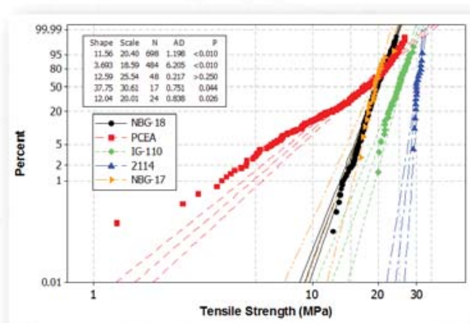
ASME Code for Graphite Core Components

Three methods are provided for assessing structural integrity

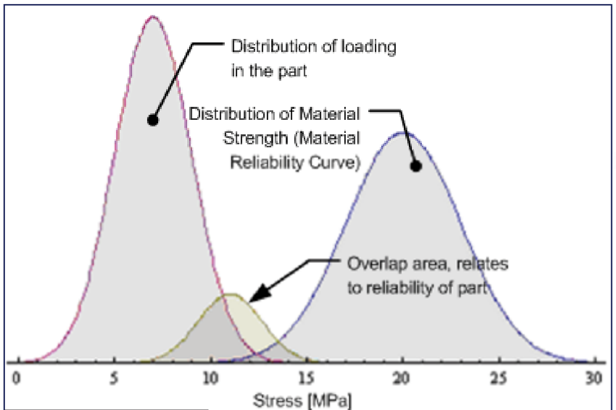
1. Deterministic
   - Simplified conservative method based on ultimate strength derived from Weibull statistics
2. Full Analysis Method
   - Detailed structural analysis taking into account loads, temperatures and irradiation history
   - Weibull statistics used to predict probability of failure
   -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

All methods must consider changes from irradiation and oxidation

Structural Reliability Class	Maximum Probability of Failure
SRC-1	1.00E-04
SRC-2	1.00E-02
SRC-3	1.00E-01



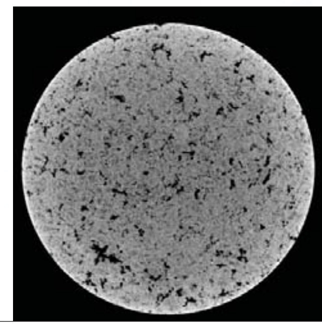
ASME Code for Graphite Core Components



From Dr. Mark Mitchell – PBMR Inc.

- New grades (third generation) are consistent and ready for codification
  - Lack of quantitative data on graphite behavior at higher temperature and dose applications
  - Test data is needed to define how precursor material changes, fabrication, and microstructure changes will affect performance
- **Probablistic** verses **deterministic** design approach
  - Deterministic is too limiting for a brittle material
  - A distribution of possible strengths in a material is needed for quasi-brittle materials (i.e., flaw size for graphite)

- Some amount of failure (i.e., a crack) is certain – graphite is porous
  - The core needs to be designed to accept some amount of failure
  - Probability of failure based upon overlap of applied stresses and graphite strength
    - Irradiation and oxidation effects must be addressed



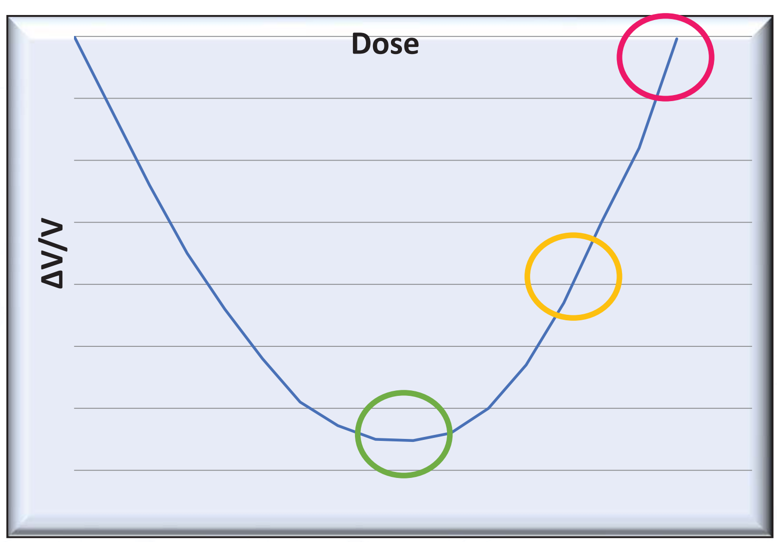
Operational Considerations – Operational Life

When do you replace the graphite?

Most Conservative Dose Level

More Risk but some Rx do operate here

Highest Risk



Operational Lifetime Considerations

Pebble Bed

- Highest component lifetime dose
  - What is expected lifetime dose?
  - Turnaround dose? After Turnaround?
- Continuous operation
  - Inspection of components is problematic
  - Component replacement is difficult
- Components in high-fluence regions should be designed for replacement
  - Will require shutdown and de-fueling of pebbles from core
- Large grain grades are possible
  - Higher Turnaround dose than fine grain
  - Lower oxidation rates than fine grain
- Irradiated test data validating models will be required
  - Currently only limited irradiation data for newer nuclear grades
  - Design life to be appropriately adjusted as data become available.
- Dust?

Prismatic

- Lower component lifetime dose
  - Still need expected lifetime dose
- Periodic shutdown
  - Much easier to inspect components
  - Components in high-fluence regions can be replaced **or** shuffled
- Finer grain grades required
  - Webbing between fuel/coolant channels requires smaller grain size
  - Slightly lower Turnaround dose
  - Higher oxidation rate
- Still requires irradiated test data to validate operational models
  - Currently only limited irradiation data for newer nuclear grades
  - Design life to be appropriately adjusted as data become available

Conclusions

- All graphite nuclear grades are proprietary
  - Graphite is porous – by design
  - Compressive applications only ($\sigma_c >> \sigma_t$)
- Irradiation behavior is required for design
  - Dimensional change and creep is life limiting mechanism
  - Strength/internal stress is dose dependent
- Degradation/Oxidation of graphite
  - Graphite does not burn (but it does oxidize at high temperatures)
  - Oxidation limited to 10% mass loss. Then replace the component
- In-service Inspection
  - Easy for Prismatic designs. More difficult for Pebble designs
    - Visual and physical inspection of accessible areas during refueling or maintenance
    - In-situ Measurements (primarily interest to pebble reactors)
- ASME Code
  - Probabilistic design calculations
    - Some amount of failure (i.e., a crack) is nearly certain over time
- Operational considerations – Pebble and prismatic
  - What is the lifetime dose of component?
    - Is this after Turnaround dose?
  - Can core be inspected? How are components to be replaced if required?
  - Oxidation rates of graphite (small versus larger grain grade)

Suggested Reading

- Manufacturing
  - ASTM D7219, Standard Specification for Isotropic and Near-isotropic Nuclear Graphites.
  - Kelly, B. T., 1981, Physics of Graphite, Applied Sciences Publishers LTD, London U.K. and New Jersey USA, 1981.
  - R.E. Nightingale, 1962, Nuclear Graphite, Academic Press, ISBN: 978-1-4832-2854-9.
- Oxidation
  - International Atomic Energy Agency, 1992, INSAG 7, "The Chernobyl Accident: Updating of INSAG 1, International Nuclear Safety Advisory Group, IAEA, Safety Series No. 75 INSAG 7, (ISBN: 9201046928), 1992.
  - Cristian I. Contescu, Robert W Mee, Yoonjo (Jo Jo) Lee, Jose D Arregui-Mena, Nidia C Gallego, Timothy D Burchell, Joshua J Kane, William E Windes, "Beyond the Classical Kinetic Model for Chronic Graphite Oxidation by Moisture in High Temperature Gas-Cooled Reactors", Carbon, 127 (2018) 158-169.
  - Joshua J. Kane, Cristian I. Contescu, Rebecca E. Smith, Gerhard Strydom, and William E. Windes, "Understanding the Reaction of Nuclear Graphite with Molecular Oxygen: Kinetics, Transport, and Structural Evolution", Journal of Nuclear Materials, Volume 493, September 2017, Pages 343–367.
  - Windes, W., G. Strydom, R. Smith, and J. Kane, 2014, "Role of Nuclear Grade Graphite in Controlling Oxidation in Modular HTGRs," INL/EXT-14-31720, Rev. 0, November 2014.
  - Laine, N. R., Vastola, F. J., and Walker Jr., P.L., 1963, "The importance of active surface area in the carbon oxygen reaction," J Phys Chem, Vol. 67, pp. 2030–2034.
  - Walker Jr., P. L., Rusinko, F., and Austin, L. G., 1958, "Gas Reactions of Carbon," Advances in Catalysis, Vol. 11, pp. 133–221.
  - Bradbury, D., Wickham, A., Graphite Decommissioning: Options for graphite treatment, recycling, or disposal, including a discussion of safety related issues, EPRI Technical Report 1013091, March 2006.
  - Walker Jr., P. L., 1990, "Carbon: An old but new material revisited," Carbon, Vol. 28, pp. 261–279.
  - Walker Jr., P. L., Taylor, T. L., and Ranish, J. M., 1991, "An update on the graphite-oxygen reaction," Carbon, Vol. 29, pp. 411–421.

Suggested Reading (cont.)

- Dust
  - Bradbury, D., Wickham, A., Graphite Decommissioning: Options for graphite treatment, recycling, or disposal, including a discussion of safety related issues, EPRI Technical Report 1013091, March 2006.
  - P. W. Humrickhouse, HTGR Dust Safety Issues and Needs for Research and Development, INL/EXT-11-21097, June 2011.
  - A. Bentaib & J. Vendel, ITER Project: Dust Mobilization and Explosion, Introductory Meeting on the Planned PSI Research Project on HTR Graphite Dust Issues, PSI, Villigen, 26-27 November 2009.
- Irradiation Effects
  - J. H. W. Simmons, Radiation Damage in Graphite: International Series of Monographs in Nuclear Energy, Elsevier publications, 2013, ISBN: 1483186490
  - Kelly, B. T., 1981, Physics of Graphite, Applied Sciences Publishers LTD, London U.K. and New Jersey, USA, 1981.
  - R.E. Nightingale, 1962, Nuclear Graphite, Academic Press, ISBN: 978-1-4832-2854-9.
  - Idaho National Laboratory, NGNP High Temperature Materials White Paper, INL/EXT-09-17187 R1, August 2012.
  - N. C. Gallego and T. D. Burchell, A Review of Stored Energy Release of Irradiated Graphite, ORNL/TM-2011/378, September 2011.
- ASME Code and Licensing
  - 2017 ASME Boiler and Pressure Vessel Code: An International Code, SECTION III: Rules for Construction of Nuclear Facility Components, Division 5: High Temperature Reactors, ASME BPVC.III.5-2017.
  - G. Longoni, R.O. Gates, B.K. Mcdowell, High Temperature Gas Reactors: Assessment of Applicable Codes and Standards, PNNL-20869 Rev. 1, October 2015.
  - Mitch Plummer and Andrea Mack, Graphite Characterization: Baseline Variability Analysis Report, INL/EXT 18 45315, June 2018.
  - Timothy D Burchell, Rob Bratton, Barry Marsden, Makuteswara Srinivasan, Scott Penfield, Mark Mitchell, Will Windes, Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs) Volume 5: Graphite PIRTs, NUREG/CR-6944, Vol. 5, 2007, Washington, DC USA.
  - AB Hull, S. Malik, M. Srinivasan, and R. Tregoning, Survey of NRC's Materials Research Associated with Advanced Reactors, Proceedings from Corrosion 2012 Research Topical Symposium on Corrosion Degradation of Materials in Nuclear Power Reactors – Lessons Learned and Future Challenges, Salt Lake City, Utah, March 11-15, 2012.

Suggested Reading (cont.)

- ASME Code and Licensing
  - Noel N. Nemetha and Robert L. Bratton, Overview of statistical models of fracture for nonirradiated nuclear-graphite component, Nuclear Engineering and Design, 2010, Vol 240, pp. 1-29.
  - S.F. Duffy and A. Parikh, Quality Control Using Inferential Statistics in Weibull-Based Reliability Analysis, ASTM International Technical Papers: Graphite Testing for Nuclear Applications, STP 1578, pp. 105-122.
  - M. Srinivasan, The Use of Small Graphite Specimen Test Data for Large Core Components for HTGR, ASTM International Technical Papers: Graphite Testing for Nuclear Applications, STP 1578, pp. 30-64.
  - Xiang Fang, Haitao Wang, and Suyuan Yu, The Stress and Reliability Analysis of HTR's Graphite Component, Science and Technology of Nuclear Installations, Volume 2014, http://dx.doi.org/10.1155/2014/964848.
  - D. Kanse, I. A. Khan, V. Bhasin, and R. K. Singh, Interpretation of ASME Code Rules for Assessment of Graphite Components, SMiRT-23 Division II Paper ID 346, Manchester, United Kingdom - August 10-14, 2015.

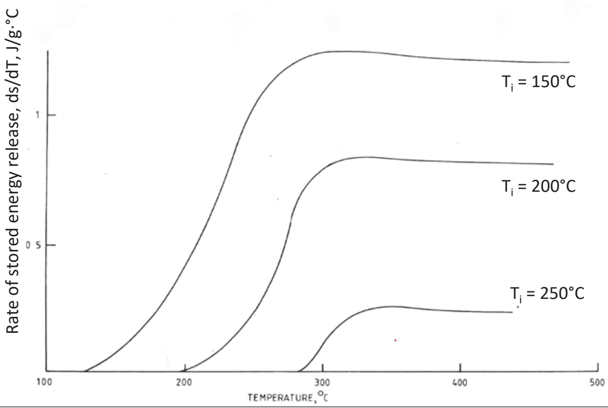
Source-dependence on graphite properties

- There is no generic "nuclear grade" graphite that can be made by all vendors
  - All nuclear graphite grades **are proprietary**. How they are made is secret to the individual vendor
    - *Completely different than metals. There is no fabrication information available for any grade.*
  - Graphite users must select the grades that match their specific requirements
  - And no, vendors wont give up their recipes. There is no customer base asking for it
- As discussed in fabrication slide the unique graphite manufacturing processes dictate the graphite behavior – both unirradiated and irradiated
  - Main fabrication parameters are:
    - *coke source: petroleum or coal-based coke source*
    - *grain size: coke particles (grains) range in size from 1800 μm to 15 μm*
    - *fabrication method: iso-static molded, vibration molded, or extruded fabrication*
    - *Grain-binder ratio: the amount of carbonaceous binder added to the grain particles*
  - Modifying these parameters can dramatically alter the unirradiated material properties and irradiation performance

Parameter	Unirradiated Behavior	Irradiated Behavior
Increased Density	Increased strength and modulus Higher fracture strength	A general decrease in Turnaround dose • Shorter component lifetime
Isostatic fabrication	Higher isotropy (than extruded) Higher cost material	Better, more predictable, irradiation performance.
Smaller grain size	More uniform, finer microstructure • Especially when isostatic molded Higher oxidation rate than larger grained	Super-fine grades may have lower Turnaround dose

Minimal effects to graphite from irradiation

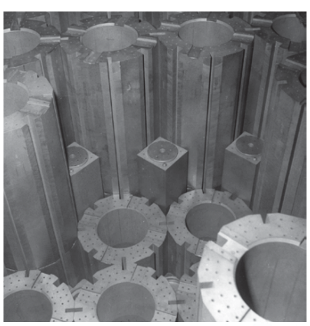
- No significant changes occur in:
  - Neutron moderation – Carbon atoms not removed
  - Specific heat capacity – Crystal structure remains intact
  - Oxidation rate - Minimal changes if any due to densification during irradiation.
  - Molten salt interaction – Graphite behavior (unirr. and irr.) similar to gas-cooled
    - Physical damage possible from salt intrusion into pores in graphite components
  - Emissivity:
    - Unaffected by irradiation but oxidation may leave impurity oxides on outer surface.



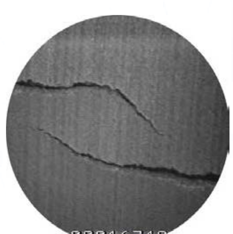
- Minimal Wigner energy release if components irradiated above 300°C.
  - Annealing of point defects in graphite is rapid above 250°C
  - Minimal accumulation of stored energy
- Need **high dose** & **low T_i**
  - Low dose/low T_i components have reduced risk

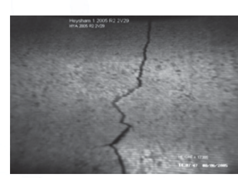
Graphite Component Failure

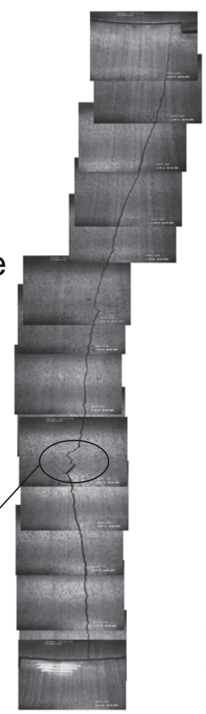
- What do we mean by structural integrity
  - U.K.'s AGR bricks – ***Now past Turnaround dose***
    - Example of graphite component failure.
  - Both axial as well as radial cracking in components
- Lifetime is completely dependent upon graphite core now
  - Not fuel design/performance, metallic internals, or secondary systems

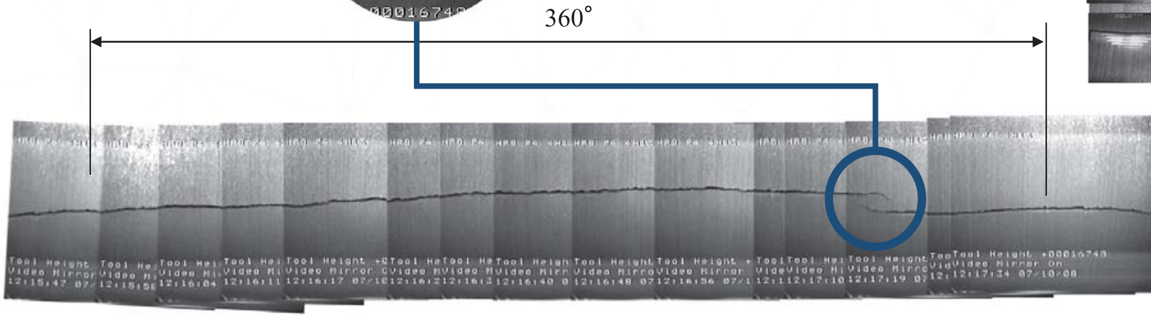


AGR Core components

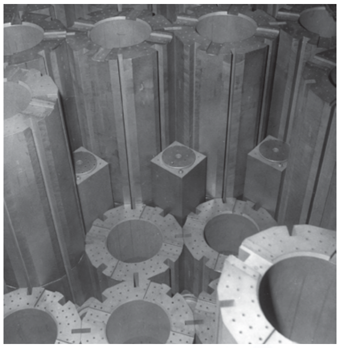








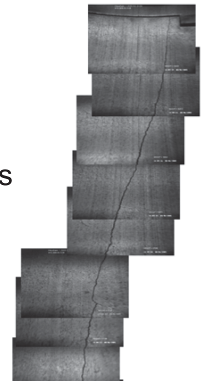
Graphite Component Failure

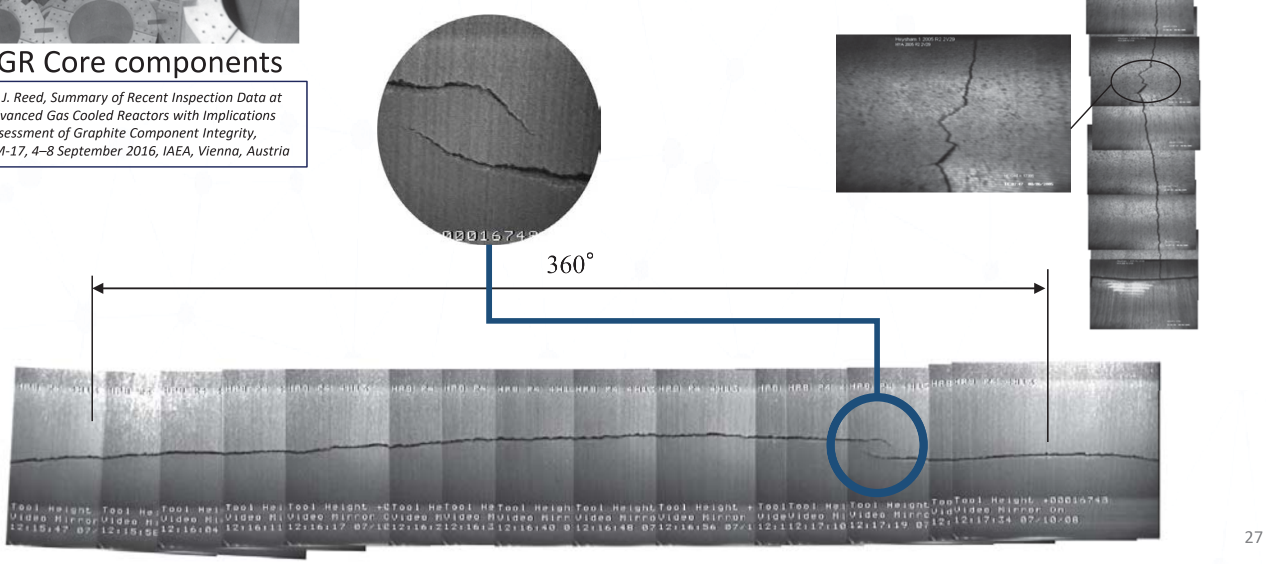


AGR Core components

From: *J. Reed, Summary of Recent Inspection Data at UK Advanced Gas Cooled Reactors with Implications for Assessment of Graphite Component Integrity, INGSM-17, 4–8 September 2016, IAEA, Vienna, Austria*

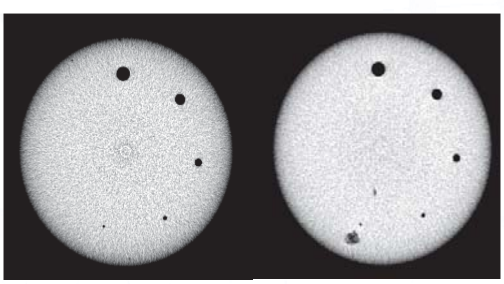
- **CAUTION!**
  - U.K. AGR uses CO_2 for coolant
  - Radiolytic oxidation exacerbates all strength changes
- Inert gas cooled designs will be more robust
  - Component strength, internal stresses, and POF will be much different than CO_2 cooled AGRs



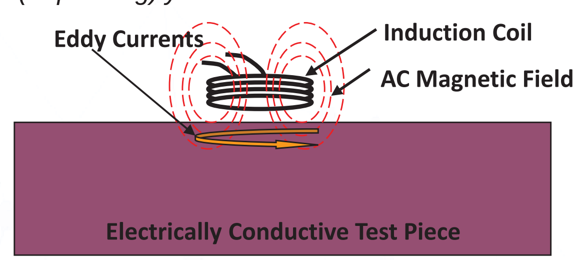


Component inspection (NDE techniques)

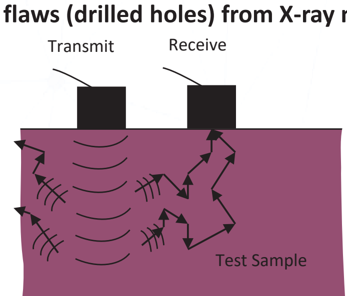
- Visual inspection, Eddy current, Ultrasonic, and X-ray inspection is possible
  - Thick graphite components are difficult to inspect
    - Flaw size resolution (i.e., cracks) are difficult to resolve in thick components
  - Visual, Eddy current, and small sample trepanning are <u>current</u> methods used
    - *U.K.'s AGR inspection program*
    - *No good technique exists. Destructive analysis (trepanning) yields most information*



Detection of flaws (drilled holes) from X-ray method



Measure Induction Coil Impedance to Detect Defects



Ultrasonic method to detect defects

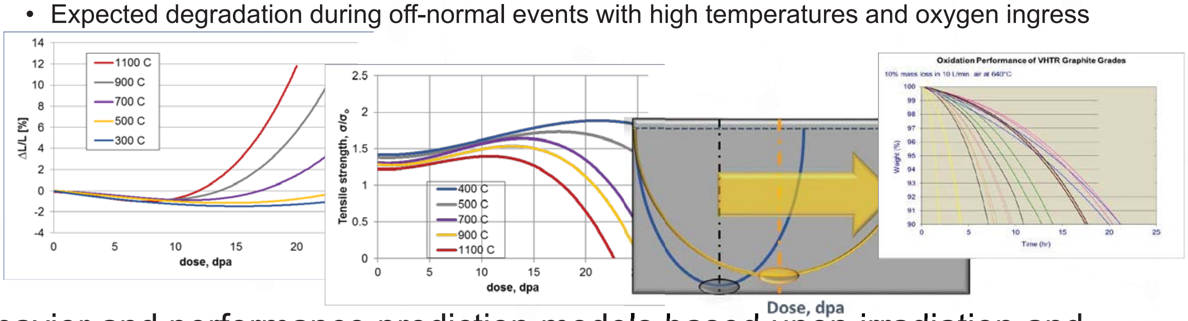
- ASTM D8093 Standard Guide for Nondestructive Evaluation of Nuclear Grade Graphite
  - Guideline on how to use various Non-Destructive Examination (NDE) techniques to graphite core components.

ASME code methodology for graphite - 1

- Two key points to keep in mind:
  1. All nuclear graphite is proprietary – Specific fabrication recipes are unknown
     - The properties for each grade are highly dependent on the recipe and **are optimized** (altered) to suit each users requirements
  2. Graphite is brittle (quasi-brittle)
     - Metals are ductile giving them the ability to fail in a predictable manner
     - Graphite fails much like ceramic – probability of failure (POF) due to flaw size distributions
     - Weibull analysis historically used to predict the probability of failure and characteristic strength of brittle and flaw dependent materials
- Consequently, there are no “standard” specifications such as metals have
  - ASTM D7219 specifies impurity levels only. Other properties are *desired* ranges
  - It’s like specifying “Stainless steel” for a component (not 304, 316, or 316L)
    - The selected grade is then fabricated to the specific requirements of component
    - **However**, not much variation over all the grades. Not like metals
      - K_{Ic} ~ 0.5 – 1.5 Pa·√m, σ_t = 15-30 MPa, 4.5 – 5.5 x 10^{-6}, etc.
- Thus, graphite code is a “process” vs just picking a preapproved material
  - The reactor applicant must demonstrate the graphite grade selected will <u>consistently</u> meet the component requirements
    - Requires property testing and analysis of the material properties **before** is durability as a nuclear component is analyzed
      - Achieved through the “Material Data Sheets” required in Code
    - Weibull parameters from strength tests used to predict the probability of failure of graphite
      - Data used in both “simple” (deterministic) and “full” (probabilistic) determination

ASME code methodology for graphite -2

- Fundamental material properties change with irradiation/oxidation
  - Code must assess changes to design of component due to these changes
    - Irradiation: changes to density, strength, dimension, CTE, thermal conductivity
    - Oxidation: changes in density, strength, CTE, and thermal conductivity
  - Code must also address these changes to in service and inspection
    - NDE and ISI are still outstanding issues that need to be addressed for graphite
- Material testing and analysis must be performed to determine changes
  - Property changes and irradiation creep to maximum expected dose levels
  - Oxidation rates, property changes, and strength assessment to maximum expected oxidation levels
    - Expected degradation during off-normal events with high temperatures and oxygen ingress



- Behavior and performance prediction models based upon irradiation and oxidation experimental results
  - Property degradation due to oxidation, irradiation, and dimensional stress buildup.
  - Fracture behavior and structural integrity = Primary

Summary of Simplified Graphite Assessment

Simple Assessment: 2 parameter Weibull (Deterministic Analysis)

Estimate the scale and shape of a 2 parameter Weibull using a linear fit to **measured property data**

m^* and S_c^*

(ref. HHA-II-3100 pg. 414)

→

Using **m^*** and **S_c^*** determine the Weibull parameters corresponding to a 95% confidence interval

$m_{95\%}$ and $S_{c95\%}$

(ref. eq.6 and eq.7 pg. 417)

→

Using **$m_{95\%}$** and **$S_{c95\%}$** determine the "design allowable stress" as a function of POF = 10^{-4}, 10^{-3}, 10^{-2} and $5x10^{-2}$ from SRCs

$S_g(P)$

(ref. HHA-II-3300 pg. 418)

→

Calculate the ratio of flexural to tensile strength

R_{tf}

(Ref.HHA-II-2000 pg. 412)

→

$S_g(P)$, R_{tf}

Perform a stress analysis of the graphite component

(ref. HHA-3215 pg. 392 and HHA-3216 pg. 393)

→

C_m = Combined Membrane Stress
C_b = Combined Bending Stress
F = Peak Stress
R_{tf} = ratio of flexural to tensile strength

→

Evaluate the acceptability of the design

- $C_m < S_g(P)$
- $C_m + C_b + F < R_{tf} * S_g(P)$

(ref. HHA-3220 pg. 394)

Summary of Full Graphite Assessment

Full Assessment: 3 parameter Weibull (Probabilistic Analysis)

Define the "**Material Reliability Curve**" by fitting a 3 parameter Weibull model to the measurement data.

(ref. HHA-II-3200 pg. 417)

→

Estimate 3 parameter Weibull parameters using MLE's .

(S_o, $m_{095\%}$ and $S_{c095\%}$)

(ref. HHA-3217 pg. 393)

→

Calculate the POF of the graphite core component using the "**Material Reliability Curve**" and stress distribution in the component.

↓

$POF_{component}$

↓

Determine the allowable POF from the **Structural Reliability Class (SRC)**, and **Service Level Design Loading**.

(ref. HHA-3230 thru HHA-3237 pg. 397)

→

$POF_{allowable}$ $\leq 10^{-2}$ and $\geq 10^{-4}$

→

Evaluate the acceptability of the design

$POF_{component} < POF_{allowable}$

ASME Code modifications (Roadmap)

- Corrosion rate variability within a nuclear grade
  - Oxidation test specimens should require testing specimen be selected at different locations within a billet, over multiple billets, and over multiple batches
    - *This will provide the oxidation rate variability across the entire specific grade*
  - Currently the oxidation mass loss for a component is limited to 10 wt%
    - *After 10 wt% the component is recommended to be replaced*
    - *Code needs to provide guidance on how the oxidation mass loss is applied*
      - **Averaged over entire core? Only in central core region? Or only for select components?**
- High temperature mechanical testing isn't really necessary for graphite
  - As noted mechanical strength and modulus increase with increasing temperature
  - Room temperature results are conservative for graphite
  - No elevated temperature testing standards exist to support this current requirement
    - *(i.e., no ASTM standards)*
  - How is elevated temperature testing of irradiated material to be conducted?
    - *Testing temperatures at (or above) T_{irr} will anneal out irradiation effects*
- Mechanical testing of irradiated material is unnecessary up to Turnaround
  - As noted mechanical strength and modulus increase with increasing dose – until Turnaround dose has been reached
  - Room Temperature/unirradiated mechanical testing is conservative until Turnaround dose has been achieved
  - If components will be used to dose levels above Turnaround (i.e., high dose levels) extensive testing will be required