Challenges in Regulatory Review of Graphite Core Design Data Extrapolated From MTR Data For Gas Cooled VHTR

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The views presented here are those of the author only and do not necessarily represent those of the US Nuclear Regulatory Commission.

Issues For Assessment

1. Background – NRC and Its Regulatory Role
2. Material Test Reactor (MTR) Data And Their Limitations – Use of "Non-Standard" Sizes and Geometry
3. Data Dispersion For Large Graphite Core Components (Non-irradiated) Using ASTM Standards
4. Methods Used to Extrapolate MTR Data
5. Need For Consistency And Acceptable Methodology For Extrapolation And Interpolation
6. Uncertainties in Data, Assumptions, And Model
7. Verification And Validation Demonstration Using Round Robin Testing on Non-Irradiated Graphite
8. Adoption Of Recommended Practice and Use in ASME Codes and Standards Development

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Mission of the NRC

The Nuclear Regulatory Commission regulates the civilian uses of nuclear materials in the United States to protect public health and safety, the environment, and the common defense and security.

The mission is accomplished through licensing of nuclear facilities which possess, use and dispose nuclear materials; the development and implementation of requirements governing licensed activities; and inspection and enforcement activities to assure compliance with these requirements.

Regulatory Responsibilities

10 CFR Parts 1 -199
Nuclear Regulatory Commission

Staff Reviews Licensee's
Safety Analysis Report (SAR)
Using Standard Review Plan

Staff Uses Guidance Documents,
such as Regulatory Guides,
codes and standards (C&S),
and industry documents,
for guidance on acceptable procedures,
methodology for technical review

Regulatory Requirements for Graphite Components

General Design Criteria (GDC) 1, 2, 4 and 10 and 10 CFR Part 50, §50.55a require that structures and components important to safety shall be constructed and tested to quality standards commensurate with the importance of the safety functions to be performed (GDC 1). and designed with appropriate margins to withstand effects of anticipated operational occurrences (GDC 10), normal plant operation; natural phenomena such as earthquakes (GDC 2); postulated accidents including loss-of-coolant accidents (LOCA), and from events and conditions outside the nuclear power unit (GDC 4).

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Regulatory Requirements for Graphite Components

The application of GDC 1 requirement to the reactor internals provides assurance that established standard design practices of proven or demonstrated effectiveness are used to achieve a high likelihood that these safety functions will be performed.

The application of GDC 2 to the reactor internals provides assurance that they will withstand earthquakes combined with the effects of normal or accident conditions.

The application of GDC 4 to the reactor internals provides assurance that the effects of environmental conditions to which they are exposed over their installed life will not diminish the likelihood of performance of these safety functions under all operating conditions, including accidents. This provides assurance that <u>failures of the reactor internals resulting from environmental service conditions that could cause loss of capability to monitor reactivity, fuel damage resulting from loss of reactivity control</u>, structural damage to fuel cladding, <u>or interference with core cooling are not likely to occur</u>.

Regulatory Requirements for Graphite Components

The application of GDC 10 to the reactor internals provides assurance that they are designed with sufficient margin to ensure their functionality and integrity during any condition of normal operation, including the effects of anticipated operational occurrences, such that a high likelihood of performance of these safety functions is achieved. Assured performance of these safety functions in turn assures that specified acceptable fuel design limits related to reactivity control and core cooling are not exceeded, thus assuring the integrity of the fuel and its cladding.

Regulatory Review of Graphite Component Design

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The regulatory staff reviews, among other information:

(1) The physical or design arrangements of all reactor internals structures, components, assemblies, and systems, including the manner of positioning and securing such items within the reactor pressure vessel, the manner of providing for axial and lateral retention and support of the internals assemblies and components, and the manner of <u>accommodating dimensional changes due to thermal and other effects</u>.

(2) The loading conditions that provide the basis for the design of the reactor internals to sustain normal operation, anticipated operational occurrences, postulated accidents, and seismic events. All combinations of listed design and service loadings (e.g., <u>operating pressure differences</u> and thermal effects, seismic loads, and transient pressure loads associated with postulated loss-of-coolant accidents) that are accounted for in design of the reactor internals.

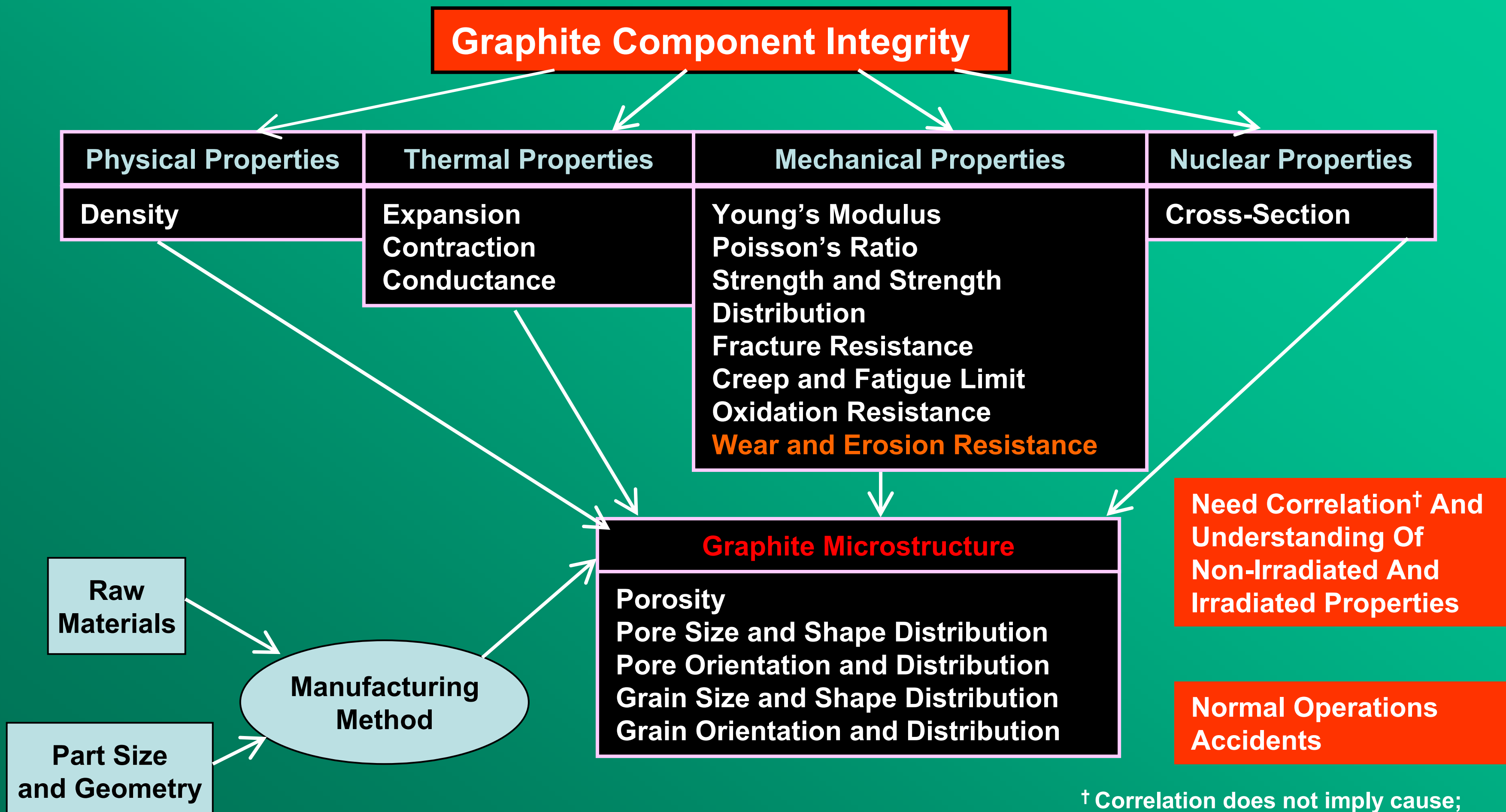
Regulatory Review of Graphite Component Design

The regulatory staff reviews, among other information:

(4) The design bases for the mechanical design of the reactor vessel internals, including allowable limits such as maximum allowable stresses; stability under dynamic loads; deflection, cycling, and fatigue limits; and core mechanical and thermal restraints (positioning and hold-down).

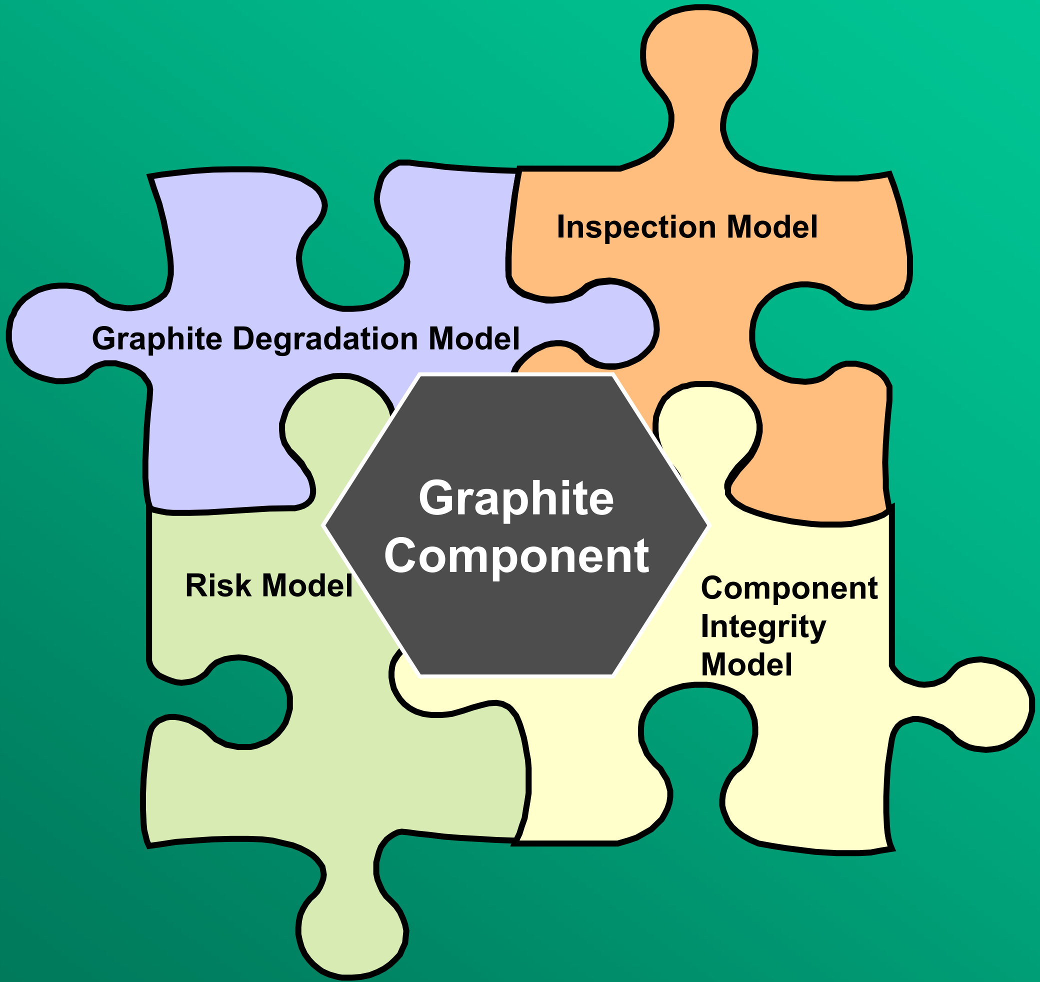
(5) Each combination of design and service loadings, categorized with respect to the allowable design or service limits (defined in the ASME Code), and the stipulated associated stress intensity or deformation limits. Design or service loadings should include safe shutdown earthquake (SSE) and operating basis earthquake (OBE) loads as appropriate.

Graphite Component Integrity

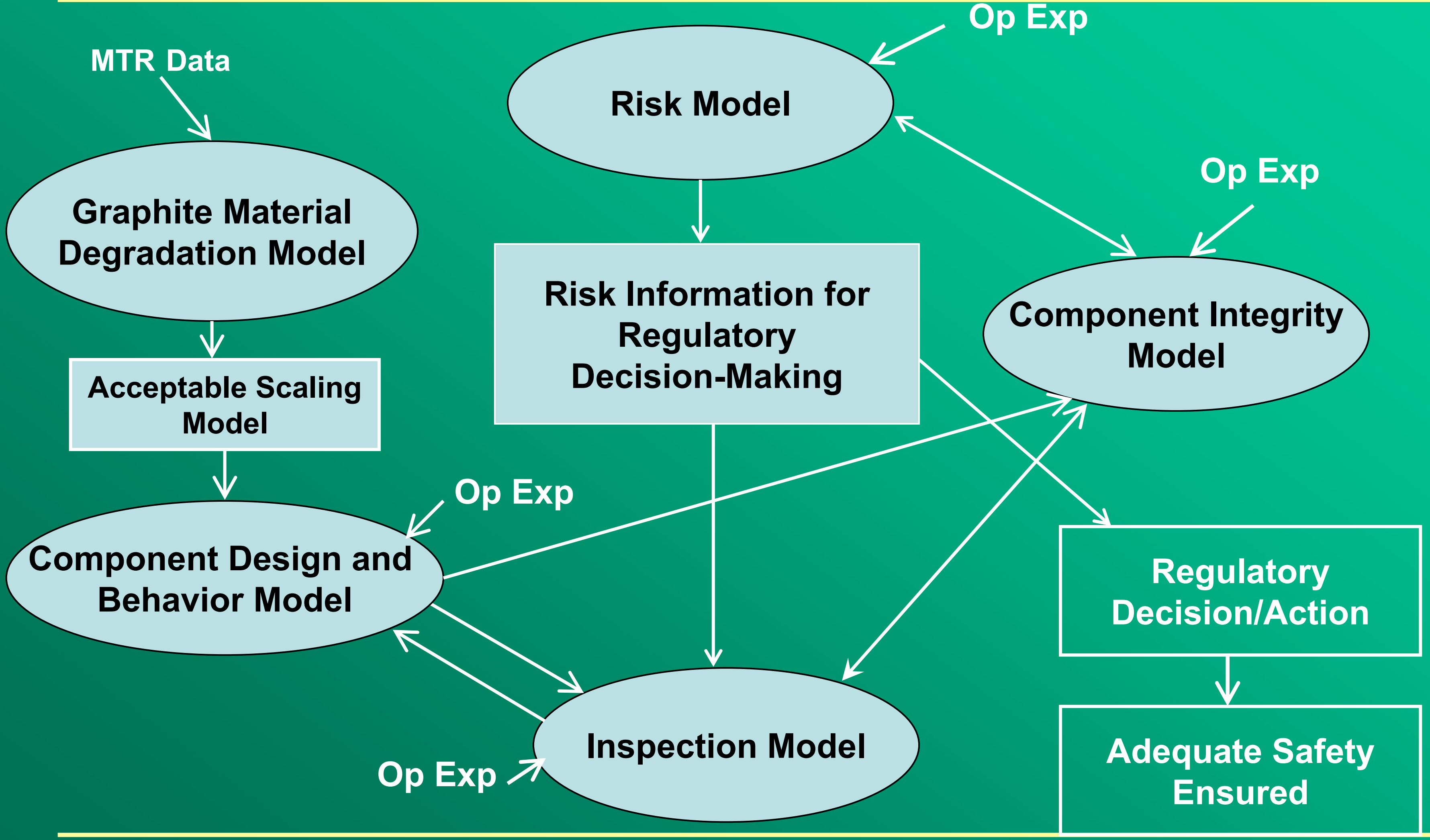


† Correlation does not imply cause; PIE and analysis may shed light.

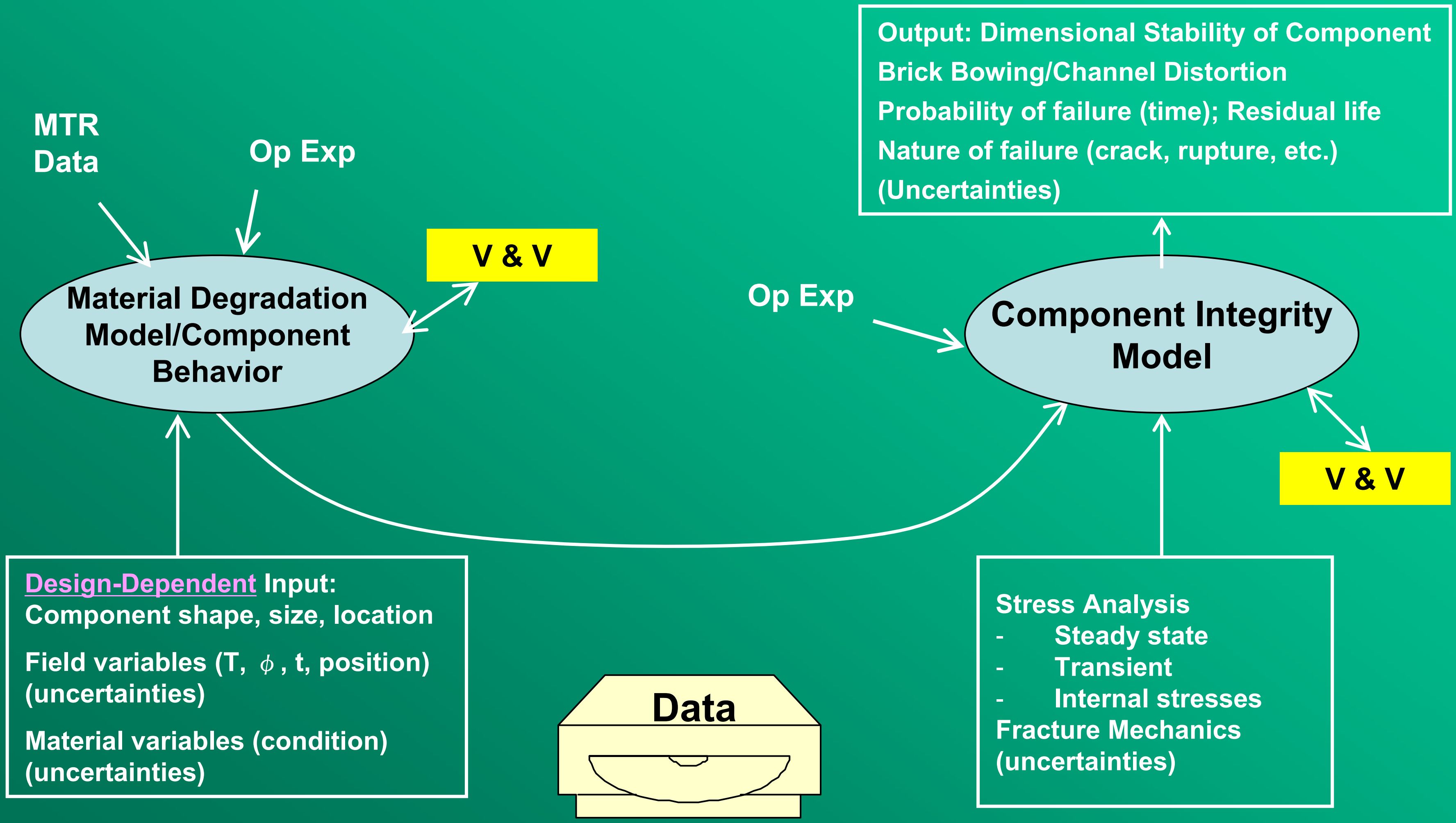
Development of NGNP- Specific PRA Tools for Graphite Components



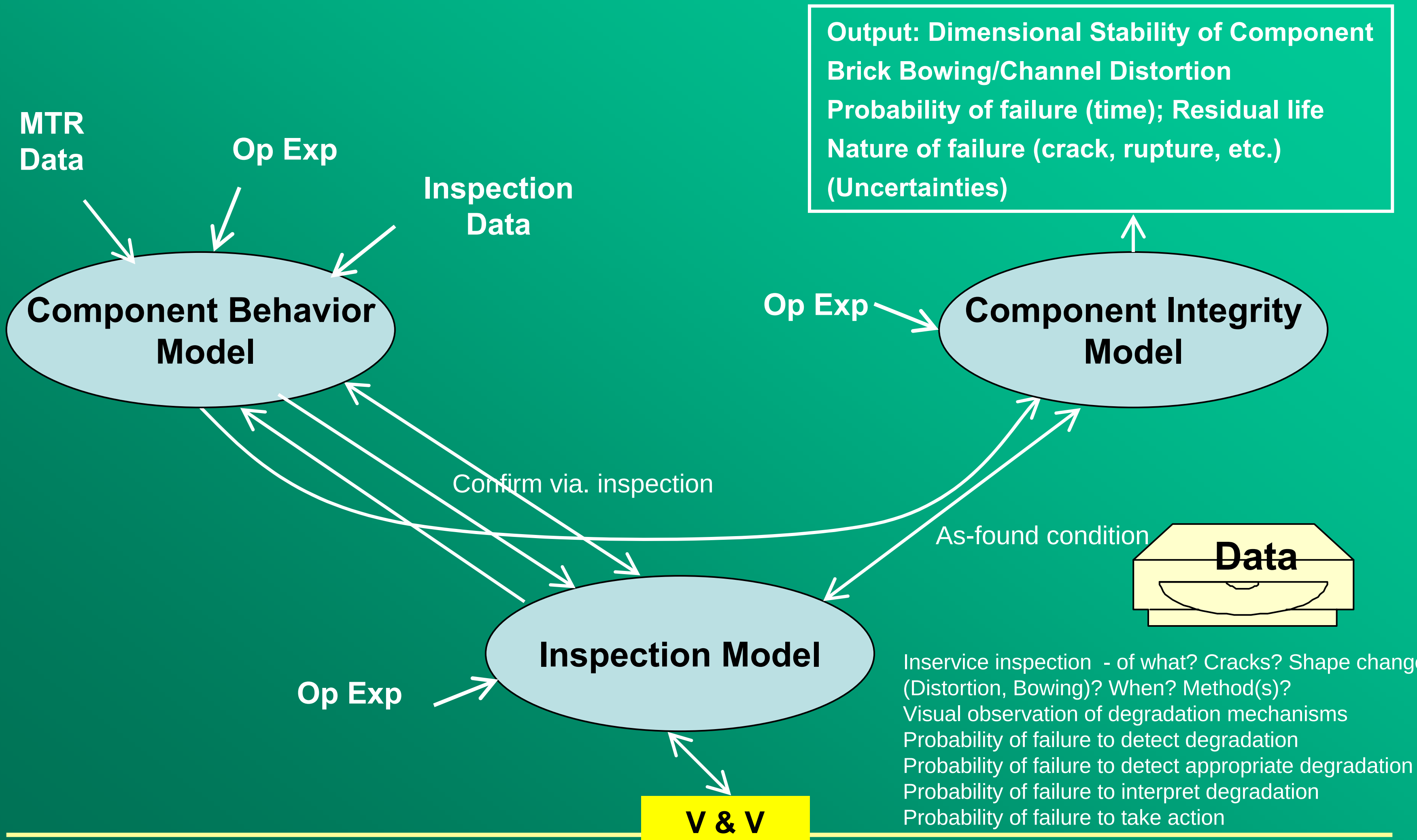
Influence of Graphite Behavior on Risk Assessment



Analysis of Graphite Component Degradation for Risk-Informed PRA



Role of Graphite Inspection in Component Integrity Evaluation for Risk-Informed PRA



Evaluation of the Risk-Informed PRA for Graphite Components

MTR Data

Op Exp

Material Degradation Model

Specific to Each Degradation

Graphite Aging Effects – CTE, Creep, Thermal Conductivity, Dimensions, Elastic Modulus, and Strength.

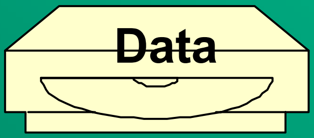
F(time (t), time at temperature (t_T), fluence (ϕ), temperature (T), and atmosphere

Issues: Generic degradation mechanism – however, extent of degradation may be component and environment (material, stress, temperature, atmosphere) specific

Unknowns (changes in environment)

Output: Change in Property as a f(time)
Degradation rate – highly variable (includes modeled and not-modeled variables)

Considers and Quantifies Uncertainties



Rationale for Irradiation Specimen Size and Geometry & Challenges

- Irradiation volume is limited and expensive
- Often several properties are measured on a single sample
- For screening or qualification studies many data points are required
- Dose gradients in the specimen have to be as small as possible
- Smaller samples become less activated – for graphite this enables handling in glove boxes instead of lead shielded cells
- Less radioactive waste after experiment

Ref: O. Wouters, A. Vreeling, A. de Jong, A. Fok and C. Berre, "Property characterisation of small graphite samples in irradiation experiments ", 8th International Nuclear Graphite Specialists Meeting, INGSM-8, Sun City, South Africa, Sep 10-12 (2007),

- Lack of specific procedures or consensus methods to estimate properties for standard ASTM sizes from very-small sized MTR specimen properties
- Limited population of MTR specimens per condition; lack of procedures to incorporate data uncertainty in results

Properties Correlation Needs

The methods development (for MTR sizes to ASTM standard sisze and geometry) would be needed for:

a) physical properties (density, permeability, and microstructural feature statistical distribution data);

b) mechanical properties (Young's modulus, Poisson's ratio, strength, and fracture toughness (via., perhaps, some indentation method);

c) thermal properties (specific heat, conductivity, coefficient of thermal expansion); and,

d) chemical (oxidation and other chemical attack due to impurities in the helium coolant).

Earlier American Irradiation Program at ORNL

Shape	Dimensions	Type of Test	Technique
Cylinder	5.1 mm dia. x 11.4 mm long	Dimensional Thermal expansivity	Optical gauge Silica dilatometer
Cylinder	6.3 mm dia. x 22.9 mm long	Dimensional Thermal expansivity Tensile test	Optical gauge Silica dilatometer Uniaxial tension
Disc	10.2 mm dia. x 1.3 mm thick	Thermal diffusivity	Heat pulse

R.J. Price and L.A. Beavan, “Final report on graphite irradiation test OG-3”, 1977. Report can be obtained from DOE OTIS.

Specimen Sizes and Geometry Used for Graphite Irradiation

ORNL Irradiation Program

- Specimen size: 50.8 x 6.3 x 2.9 mm (T. Burchell and L. Snead, The irradiation behavior of VHTR candidate graphite grade NBG-10", 6th Internaitonal Nuclear Graphite Specialists Meeting (INGSM-6), Chamonix, France, Sep 18 – 22 (2005) .
- (HTN Seires capusles) H-451 graphite: ≈ 12 mm outer dia, 3 mm inside dia, and 6 mm length. (Burchell, T. D., Neutron Irradiation Damage in Graphite and its Effects on Properties, in Proceedings Carbon '02, Beijing, China, September 15–19, 2002).
- IG-110 Graphite: Rings of (1) 38.1 mm OD, 27.9 mm ID, and 6.4 mm thickness; (2) 19.1 mm OD, 8.9 mm ID, and 6.4 mm thickness; 6.4 mm OD, 3.1 mm IDn and 6.4 mm thickness. Disks of 6.4 mm diameter with 2 mm thickness. (S. Ishiyama, T.D. Burchell, J.P. Strizak and M. Eto, "The effect of high fluence neutron irradiation on the properties of a fine-graiend isotropic nuclear graphite", J. Nucl. Matls. 230, 1-7 (1996).
- HFIR Irradiation, HTV-1 and HTV-2, graphite test specimen sizes:
  (1) 7.62 mm diameter X 5.33 mm length;
  (2) 8.89 mm diameter X 5.33 mm length; and
  (3) 10.16 mm diameter X 5.33 mm length.

  T. Burchell, D. Heatherly J. McDuffee, D. Sparks and K. Thoms, "Experimental Plan and Final Design Report for HFIR High Temperature Graphite Irradiation Capsules HTV-1 and - 2", ORNL-GEN4/LTR-06-019, August 31 (2006)

Current INL Irradiation Programs

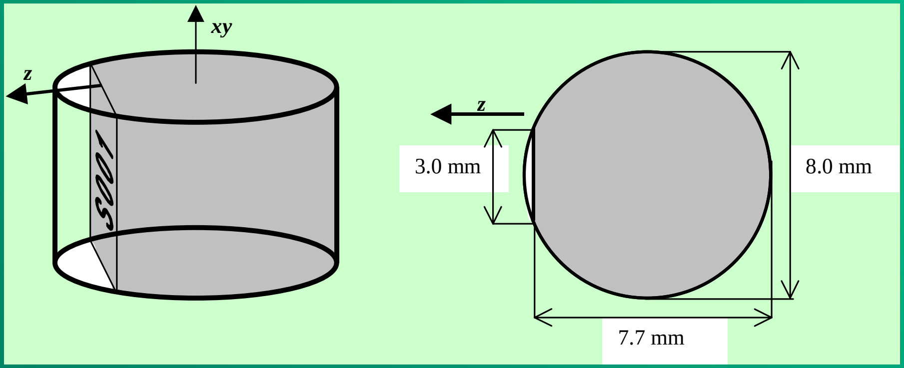
- Idaho National Laboratory, AGC Irradiation graphite test specimen size: 12.7 mm diameter X 25. 4 mm length

  (R. Bratton, "NGNP AGC-1 Graphite Irradiation Capsule", 6^{th} International Nuclear Graphite Specialists Meeting, INGSM-6, Chamonix, Sep 18 – 21, France (2005).

Petten Pre-& Post-Irradiation Examinations†

- Dimensions, Density , R.T. *(ASTM C781/E1461)*
- Dynamic Young's modulus, R.T *(ASTM C769, time of flight)*
- Coefficient of Thermal Expansion, R.T-T_{irr} *(ASTM C781/E228)*
- Coefficient of Diffusivity, R.T-T_{irr} *(ASTM C781/E1461/C714, laser flash)*

Cylinder Height = 6 or 12 mm

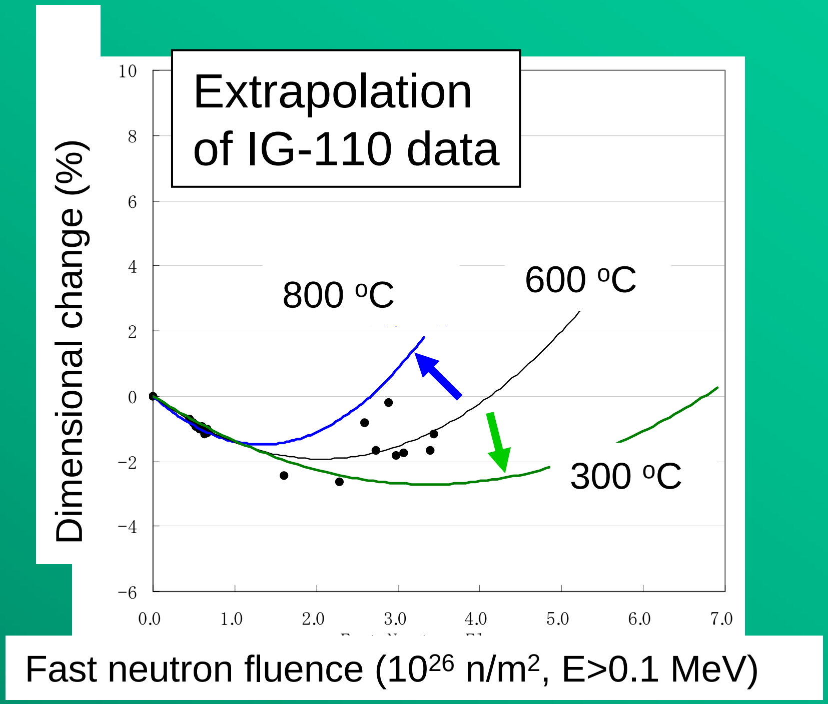


†J.G. van der Laan, J.A. Vreeling and J.B.J. Hegeman, "EU graphites irradiation programme at HFR Petten", 6th International Nuclear Graphite Specialists Meeting, Chamonix, France, Sep 18 – 21 (2005)

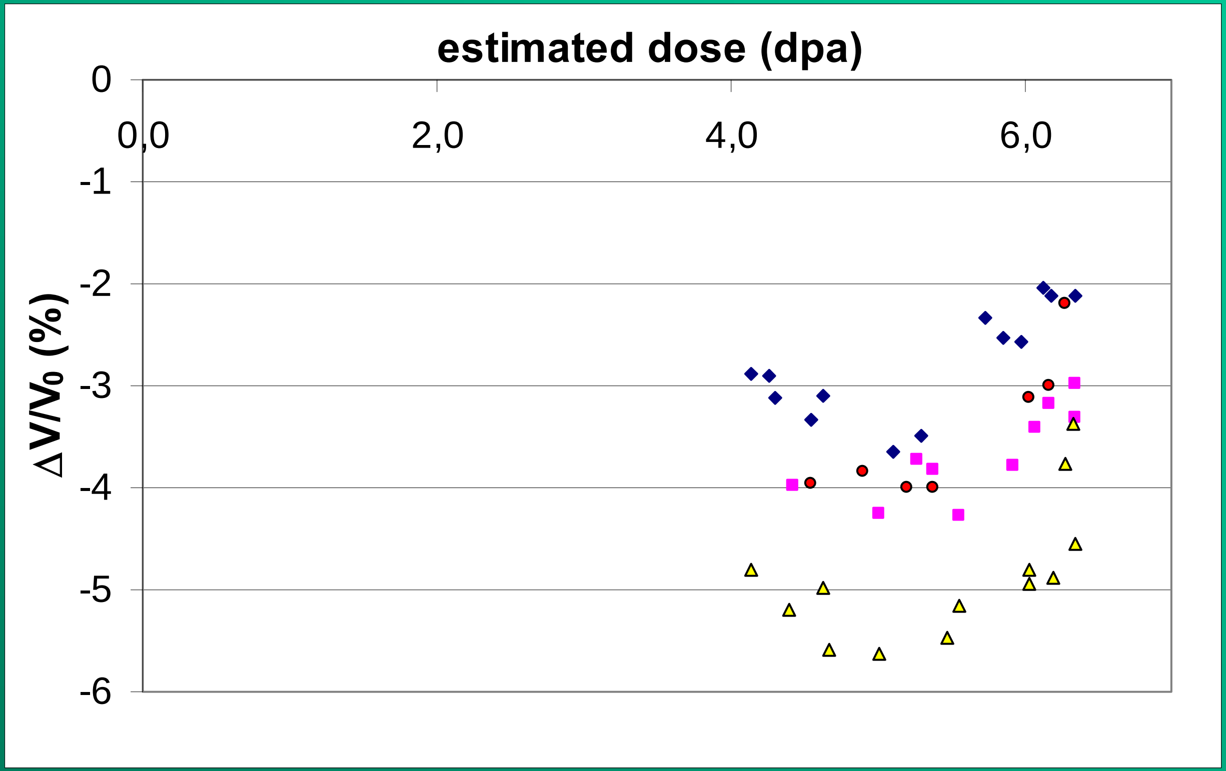
Examples of the Nature of Dispersion of Graphite Irradiation Data

Methods are being developed by the Atomic Energy Society of Japan.

T. Shibata, E. Kunimoto, J. Sumita, M. Yamaji, T. Konishi and K. Sawa, "R&Ds for application of IG-110 graphite to VHTR in-core components", 9th International Nuclear Graphite Specialists Meeting (INGSM-9) The Netherlands, September 15-17, 2008.

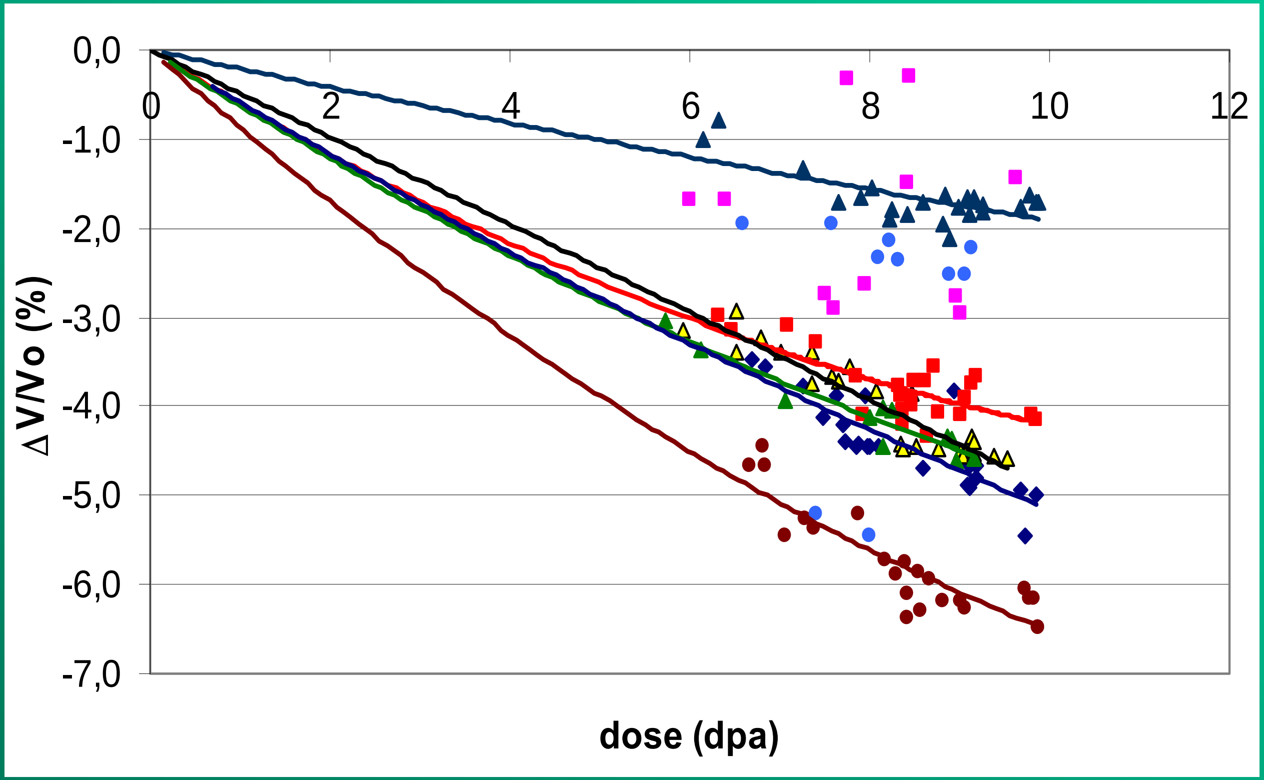


Examples of the Nature of Dispersion of Graphite Irradiation Data - Continued



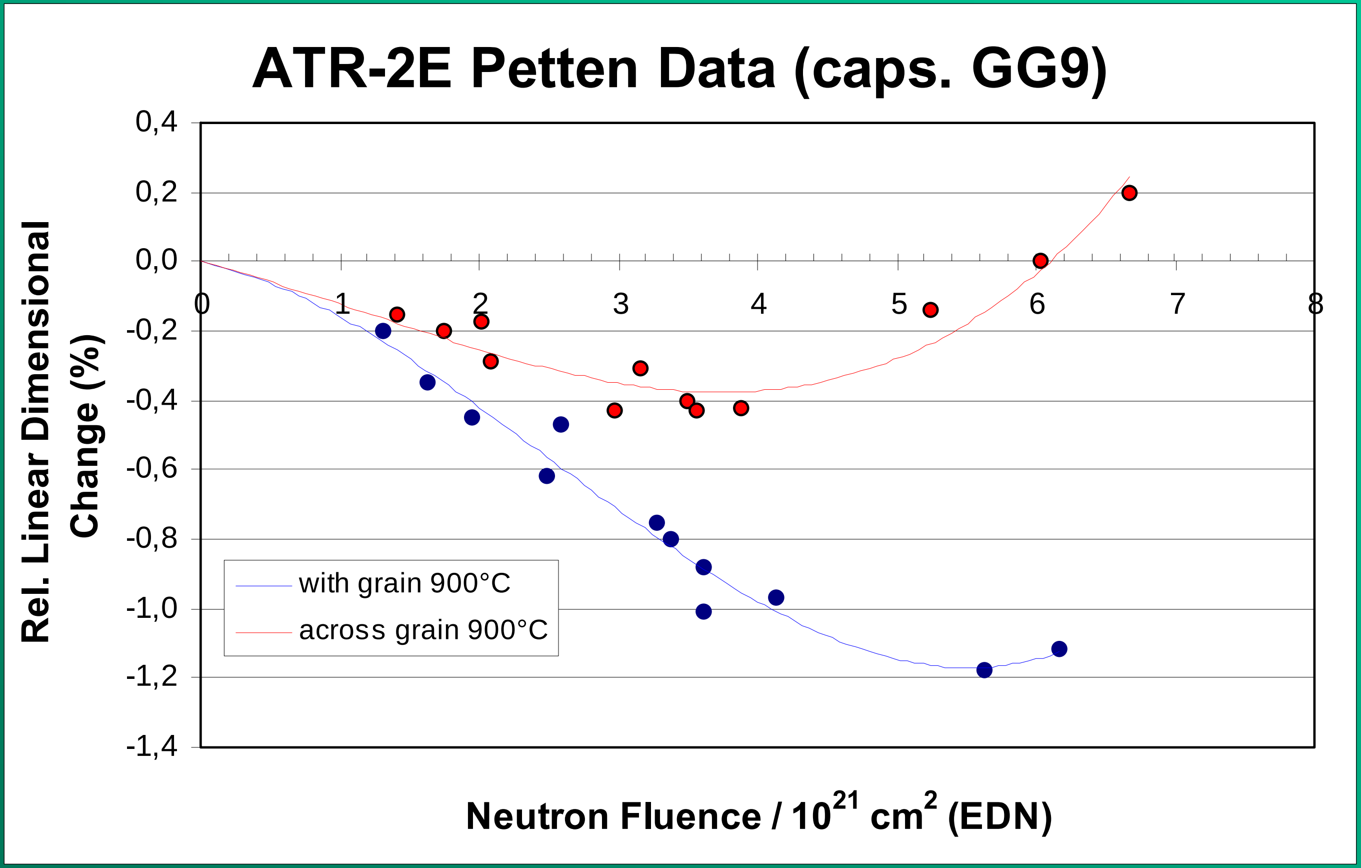
J.A. Vreeling, O. O. Wouters, J.G. van der Laan, D. Buckthorpe and M. Davies, "Graphite irradiation testing for HTR technology – an update", 9th International Nuclear Graphite Specialists Meeting, INGSM-9, Egmond aan Zee, The Netherlands, Sep 14-19 (2008),

Examples of the Nature of Dispersion of Graphite Irradiation Data - Continued

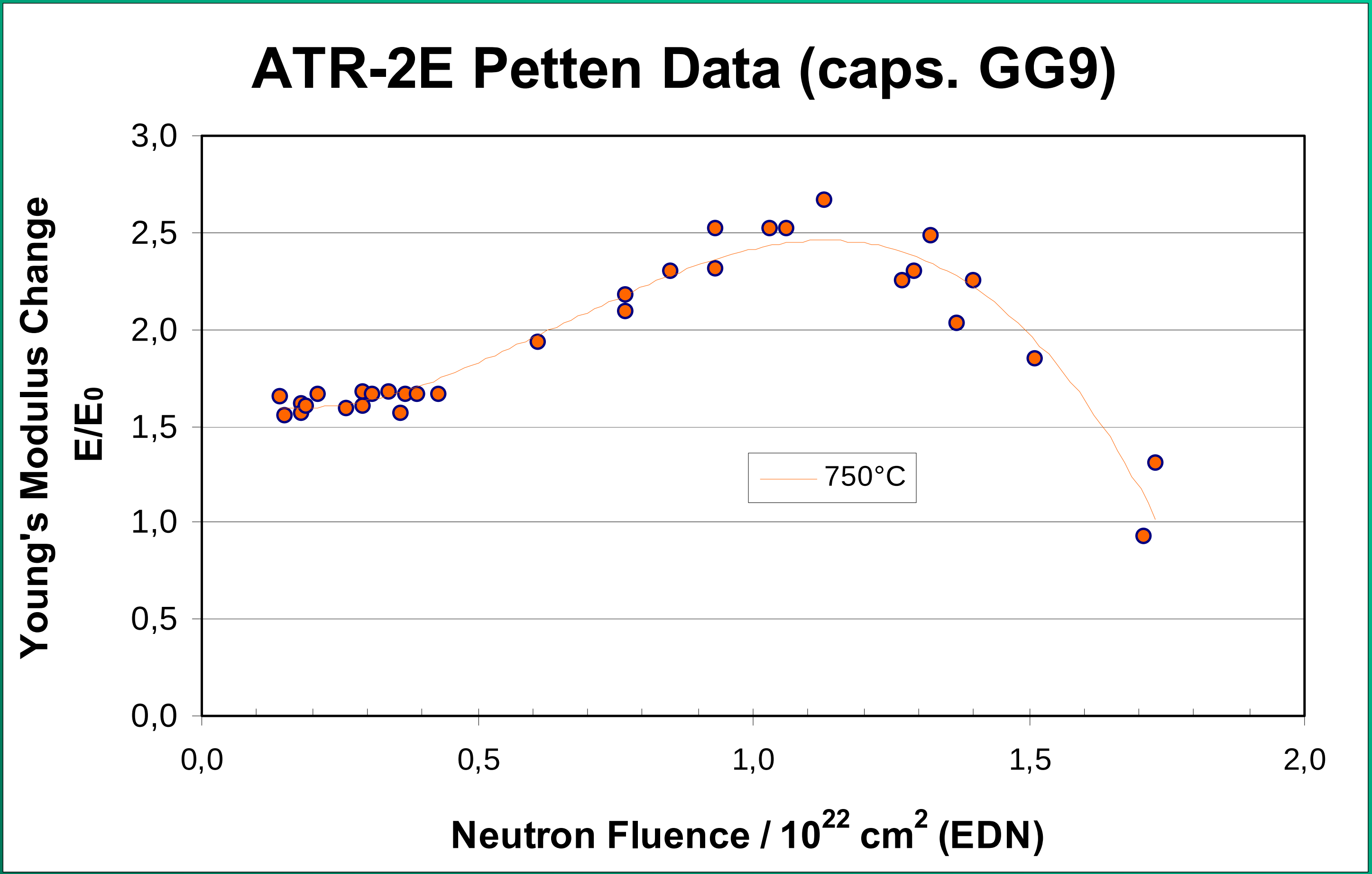


J.A. Vreeling, O. O. Wouters, J.G. van der Laan, D. Buckthorpe and M. Davies, "Graphite irradiation testing for HTR technology at the High Flux Reactor in Petten", 8th International Nuclear Graphite Specialists Meeting, INGSM-8, Sun City, South Africa, Sep 10-12 (2007),

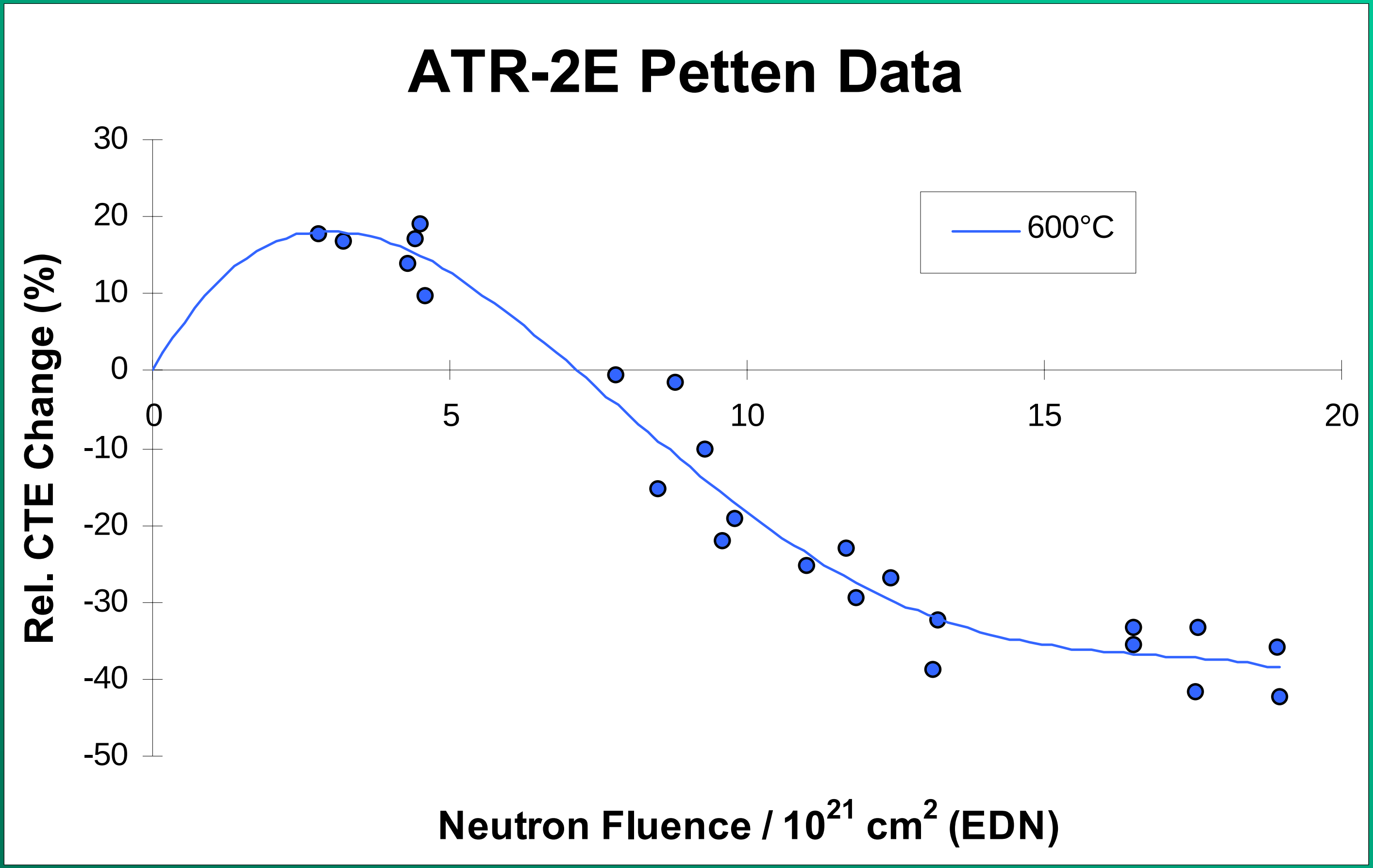
Examples of the Nature of Dispersion of Graphite Irradiation Data - Continued



Examples of the Nature of Dispersion of Graphite Irradiation Data - Continued



Examples of the Nature of Dispersion of Graphite Irradiation Data - Continued



Classes of Irradiation Data (Model) Uncertainty

- *Epistemic*: lack-of knowledge uncertainties arising because our scientific understanding of irradiation data is imperfect for the present, but are of a character that in principle are reducible through further research and gathering of more and better irradiation data.
- *Aleatory* - "random" (stochastic) in character; uncertainties in models which for all practical purposes cannot be known in detail or cannot be reduced.

Even under "perfect information“, i.e., when the model has been validated and the numerical values of its parameters are known, these aleatory uncertainties are still present (for a given model).

- Approximations (predictions deviate by a fixed but unknown amount from observed values of the predicted value).
- uncertainties about the numerical values of the parameters of a given model

Adapted from: Ref: R. J. Budnitz G. Apostolakis, D. M. Boore, L. S. Cluff, K. J. Coppersmith, C. A. Cornell, and P. A. Morris “Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts”, NUREG/CR-6372 (1997).

Development Needs for Using Irradiation Data for Codes and Standards

- Development of procedures to interpret MTR irradiation data, consistent with ASTM properties determination specifications
- Development of procedures to interpolate and extrapolate limited MTR irradiation data, quantifying the uncertainties in the data and model
- Development of methods to check for consistency in irradiation data using “non-standard” specimen size and geometry
- Validation and verification of the properties behavior model
- Formulation of consensus standard procedure to use MTR irradiation data for large specimen sizes (components)
- Incorporation of the standard in ASME graphite design codes

Considerations for Next Step

- Demonstrate proof-of-concept using non-irradiated nuclear graphite specimens
- Fabricate specimens conforming to:
  - Existing ASTM standard size and geometry for various significant physical and mechanical properties
  - Several selected size and geometry being used for current NGNP irradiation programs
- Determine various physical and mechanical properties
- Compare and interpret data for consistency and conformance to material microstructural sampling and representation, including quantification of uncertainties
- Develop procedures to extrapolate information from MTR data to properties prediction for large graphite blocks with typical non-homogeneity in microstructure
- Recommend range of allowable specimen sizes and geometry for future graphite irradiation.

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Summary

1. Logistic, operational, and economic considerations limit the size and geometry of graphite material test reactor (MTR) specimens to deviate from recommended ASTM standard sizes and geometry.
2. Recommended procedures need to be developed to translate MTR properties to those of ASTM standard sizes and geometry, where applicable.
3. Development needs exist to interpolate and extrapolate irradiation data to "large" graphite component sizes, considering the epistemic and aleatory uncertainties, including their quantification.
4. A round-robin (non-irradiated) properties determination exercise to examine the consistency and conformance of physical and mechanical properties data, obtained from MTR-type specimen size and geometry to current ASTM specimen size and geometry may be considered as a first step towards drafting consensus standards for irradiation testing and data interpretation.

ABBREVIATIONS

ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing Materials
CFR	Code of Federal Regulations
CTE	Coefficient of Thermal Expansion
C&S	Codes and Standards
GDC	General Design Criteria
LOCA	Loss of Coolant Accident
MTR	Material Test Reactor
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
OpE	Operating Experience
PRA	Probabilistic Risk Assessment
SAR	Safety Analysis Report
SRP	Standard Review Plan
SSE	Safe Shutdown Earthquake
V&V	Validation and Verification

Inter-MTR Irradiation Data Comparison for ATR-2E Graphite

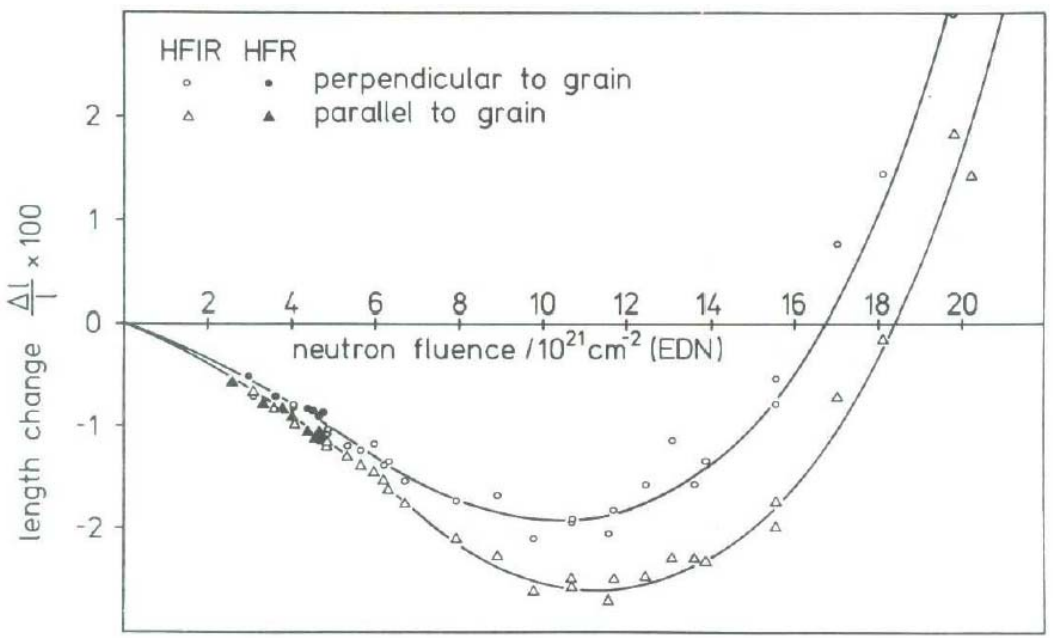

Fig. 12 Dimensional changes of ATR-2E graphite in HFIR Oak Ridge and HFR Petten at 600°C

G. Haag, "Properties of ATR-2E Graphite and Property Changes due to Fast Neutron Irradiation", Berichte des Forschungszentrums Jülich, Report 4183. October 2005.