Validation of Radionuclide Transport Methods

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Nuclear Regulatory Commission Staff
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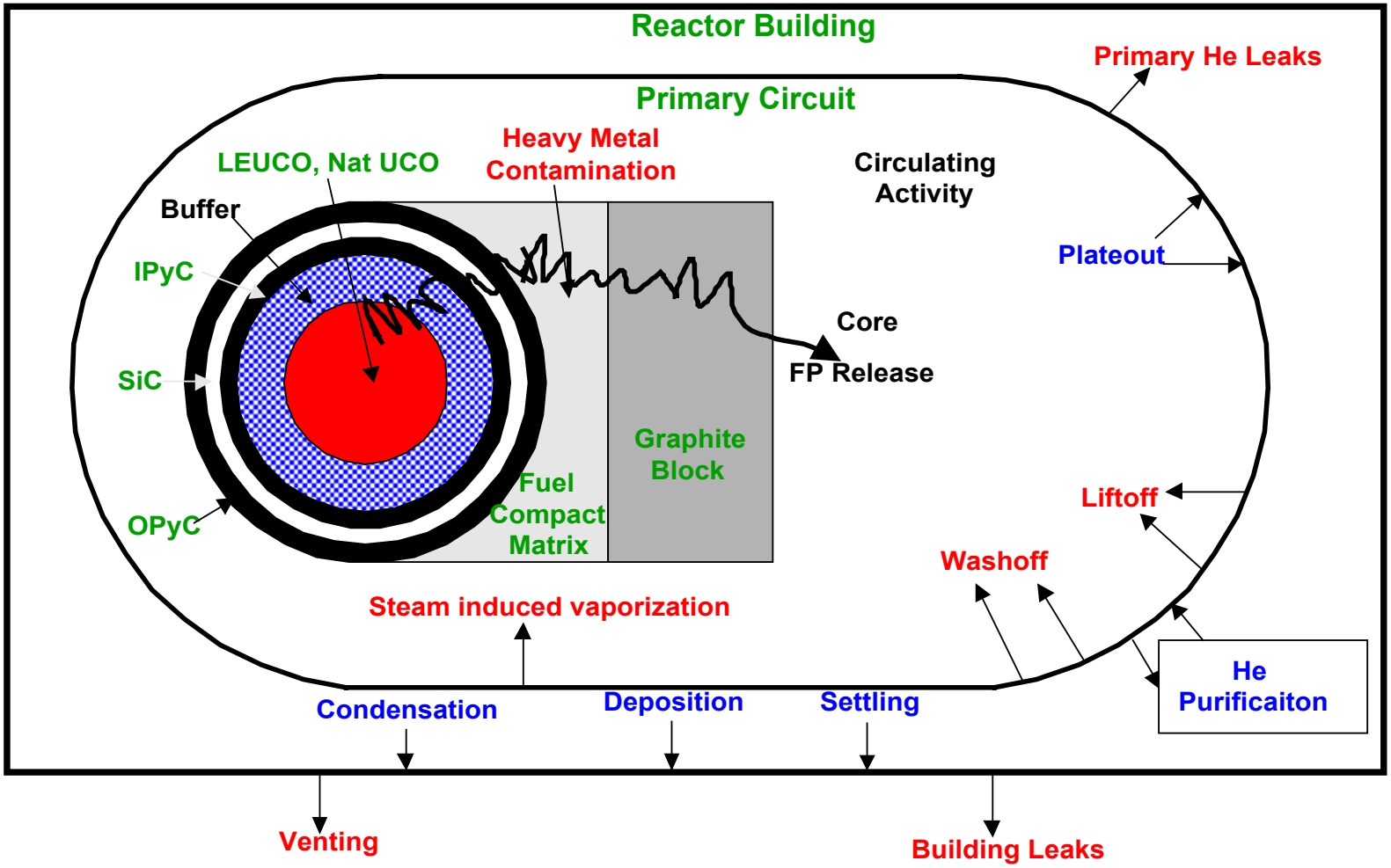
General Atomics

GENERAL ATOMICS

Outline

- RN Release Barriers (Review of 1/28/03)
- Reference Design Methods
- RN Transport Design Data Needs
  - Objective
  - Existing data
  - Additional data needed
  - Planned technology programs

GT-MHR
Radionuclide Containment System



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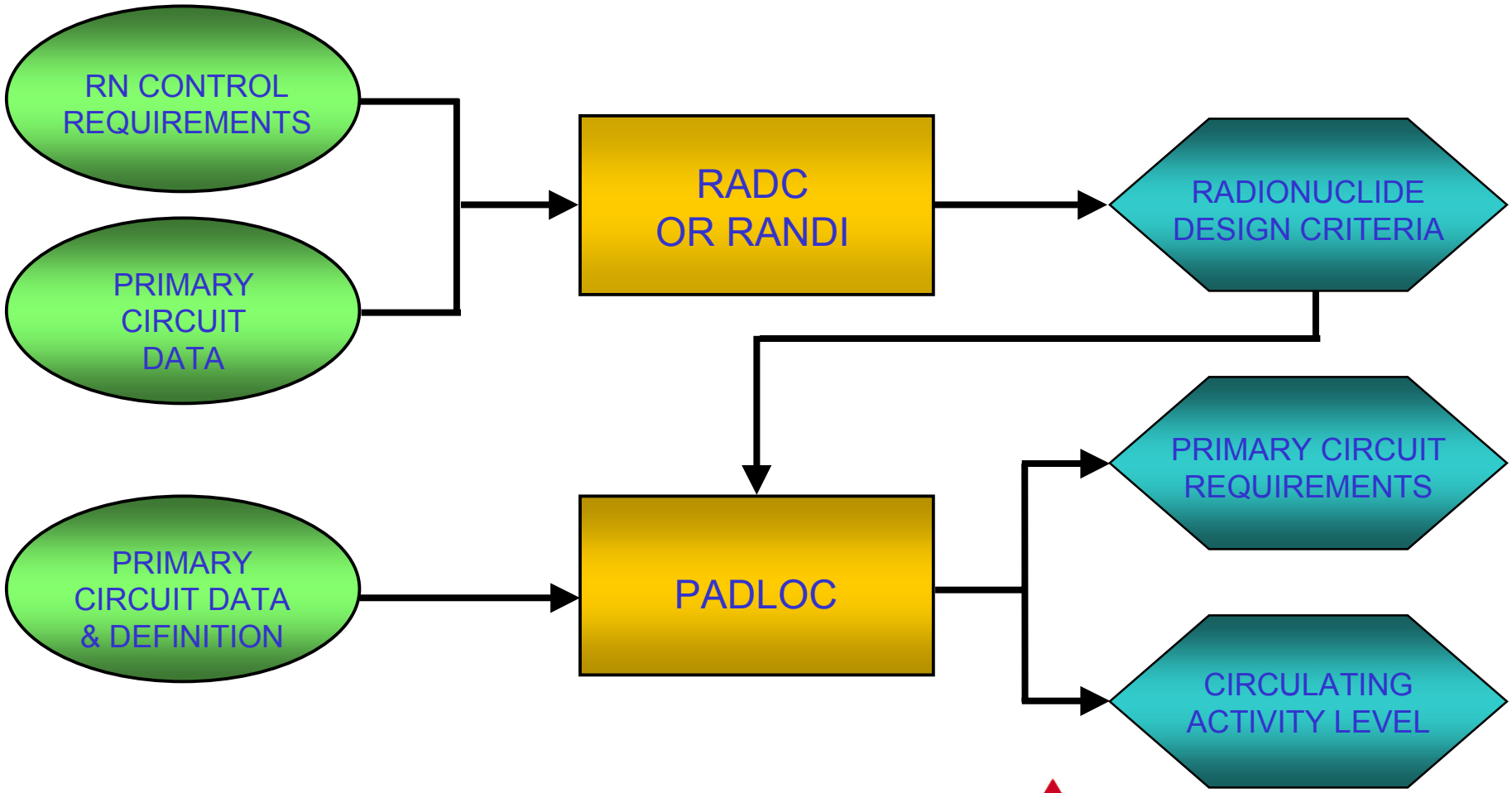
Introduction and Background

- Analytical Tools Available to Predict Performance of Radionuclide Containment System
  - Computer codes
  - Component models (typically phenomenological models)
  - Material property correlations
- Codes Used Extensively for Reactor Design & Analysis, including Fort St. Vrain HTGR
- Codes Not Completely Verified And Validated
- DDNs and Attendant Test Programs Identified
  - Differential test data for improved component models
  - Independent integral data for code validation

GA Fuel/fission Product Codes for Normal Operation

CODE	APPLICATION
SURVEY	Full-Core Fuel & Graphite Temperatures, Fuel Failure, and Fission Gas Release
TRAFIC	Full-Core Fission Metal Release
PADLOC	Plateout Distributions in Primary Circuit
RADC	Overall Plant Mass Balance for Radionuclides
TRITGO	Overall Plant Mass Balance for Tritium

Radionuclide Control Design Methods



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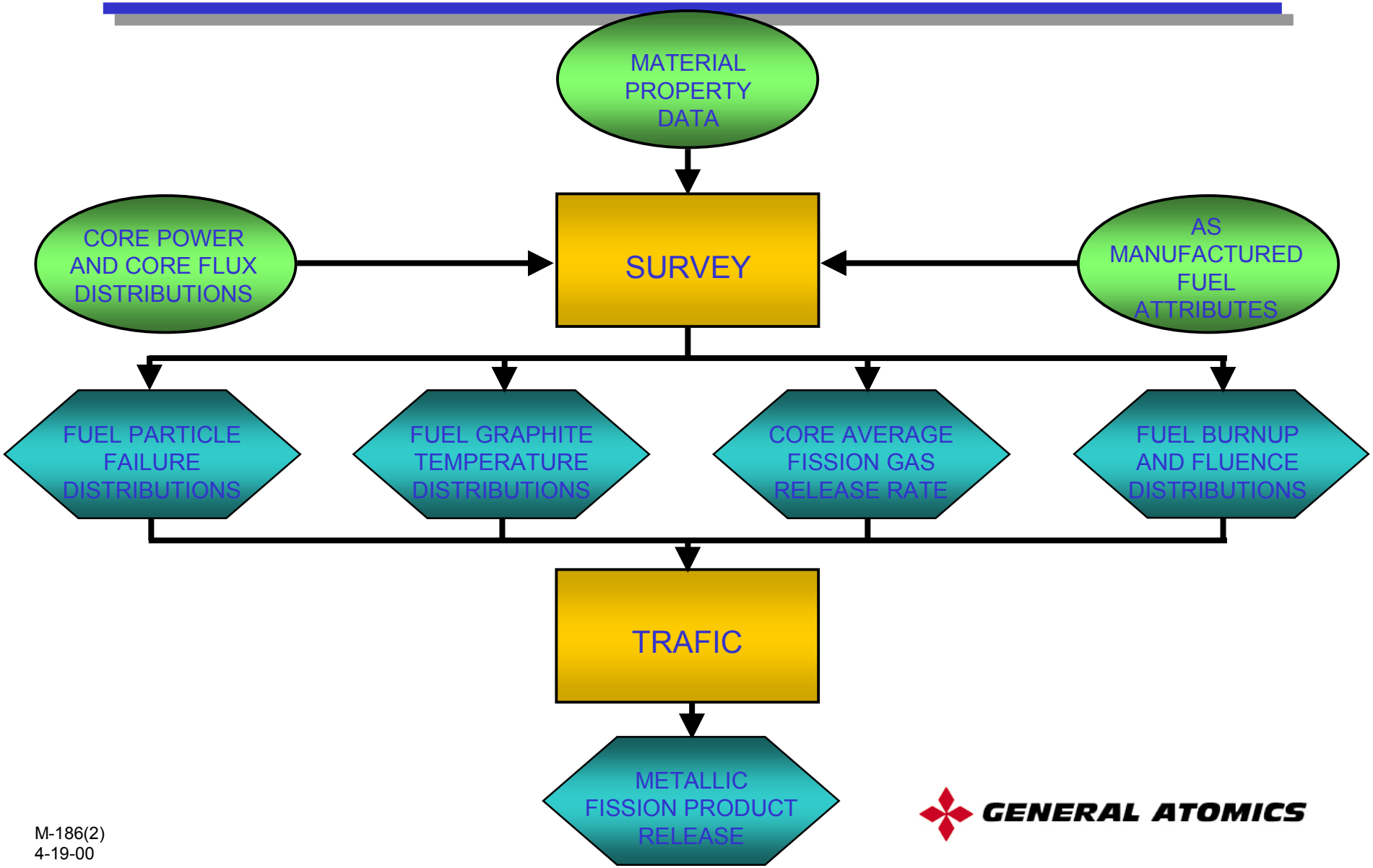
Core Performance Design Methods



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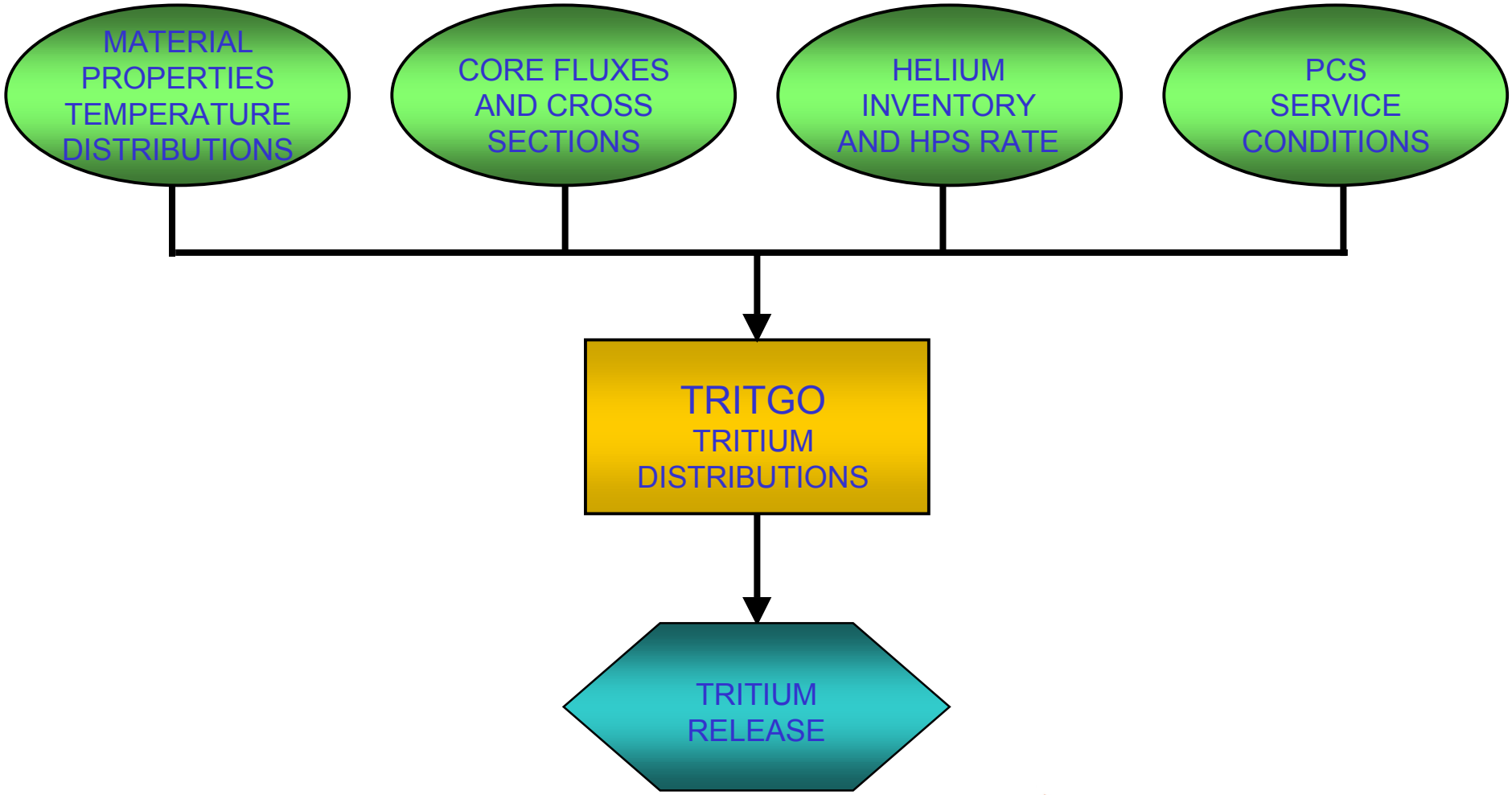
Tritium Control Design Methods



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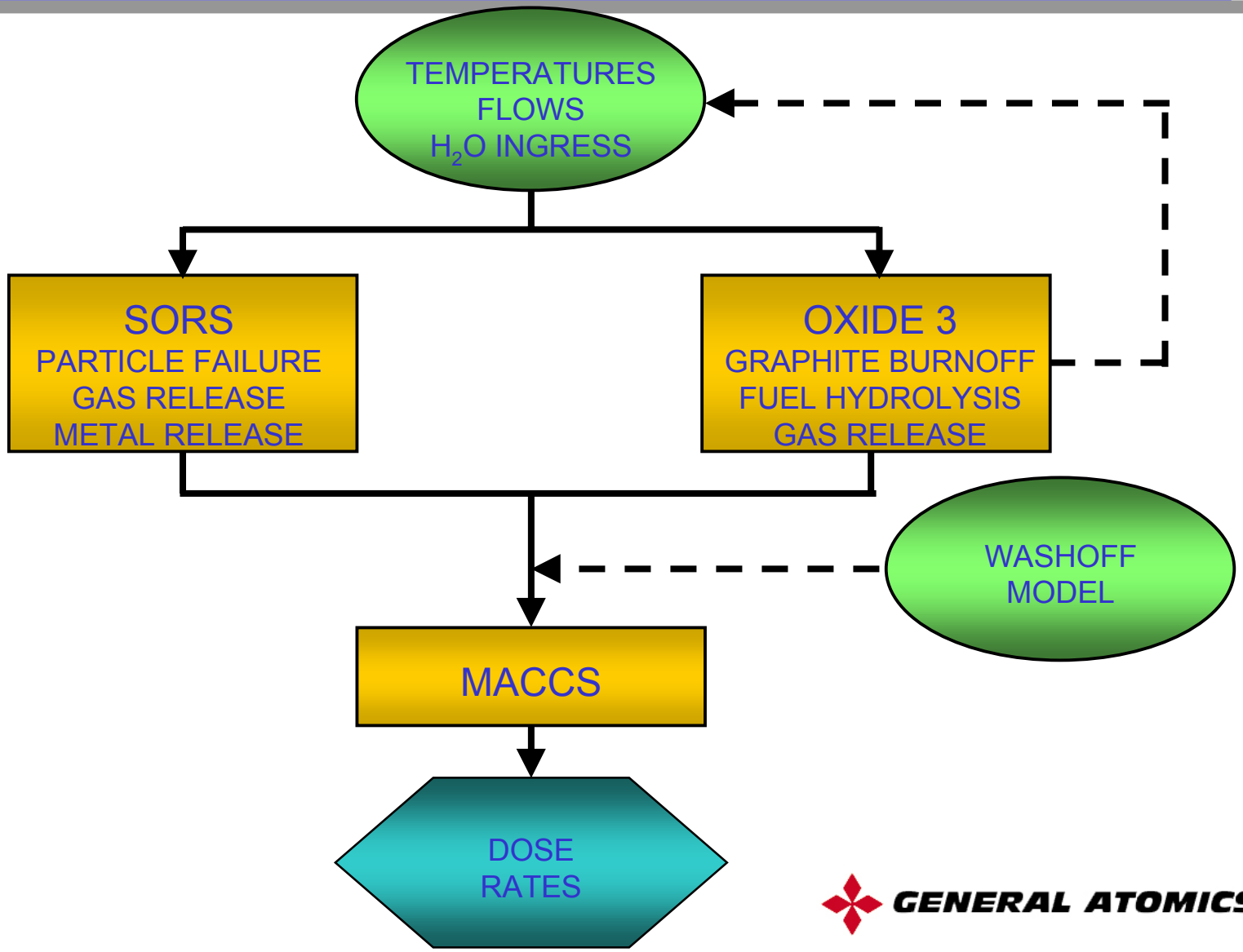
GA Fuel/Fission Product Codes for Accident Conditions

CODE	APPLICATION
SORS	Transient, Full-Core, Fuel Failure and Fission Product Release (Gases and Metals)
OXIDE3	Transient, Full-Core, Graphite Corrosion and Fuel Hydrolysis for Large H_2O Ingress
POLO	Transient, Fission Product Release from Primary Circuit to Reactor Building
MACCS	Fission Product Transport in Reactor Building and Environment; Offsite Dose

Core Heatup with H_2O Ingress Design Methods



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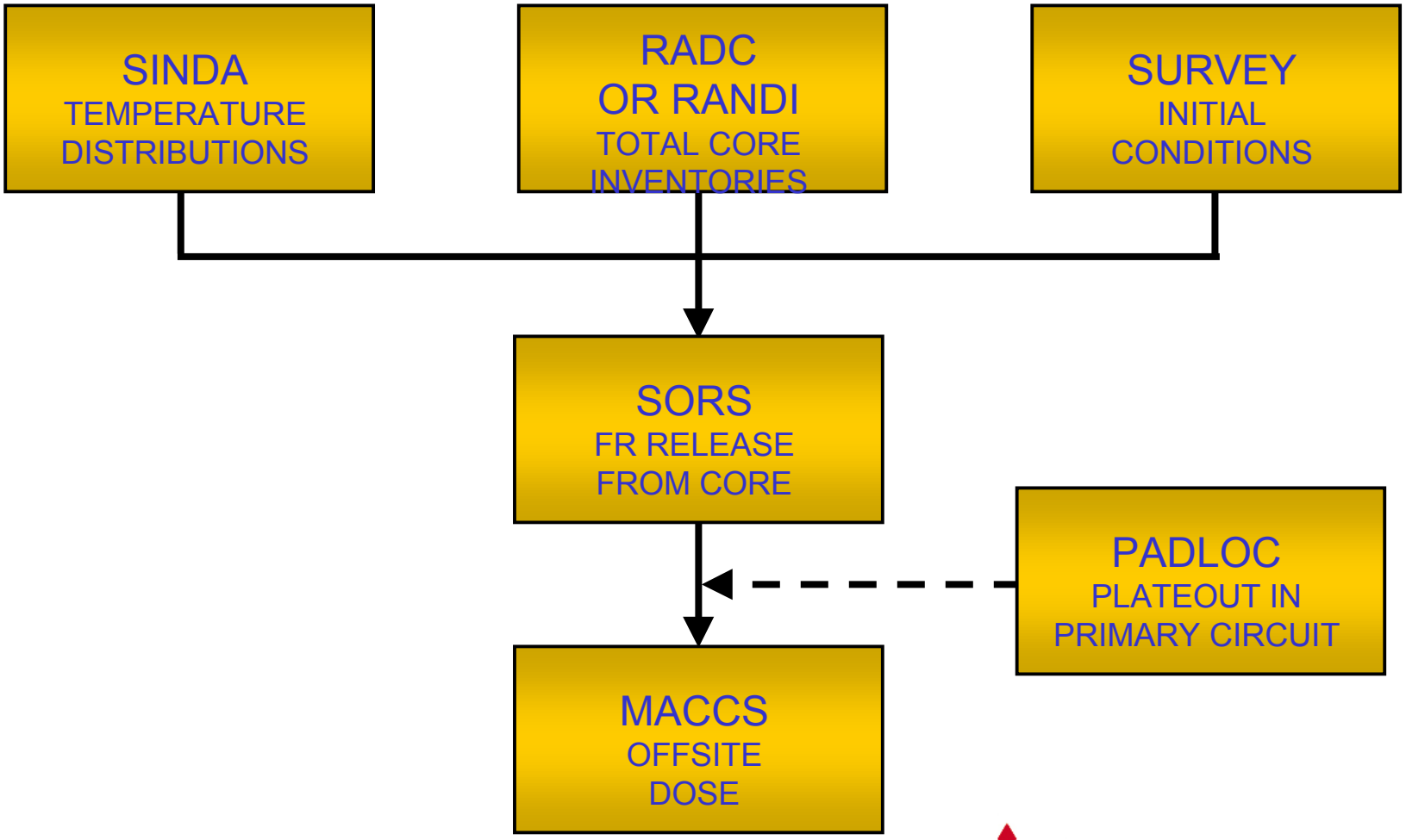
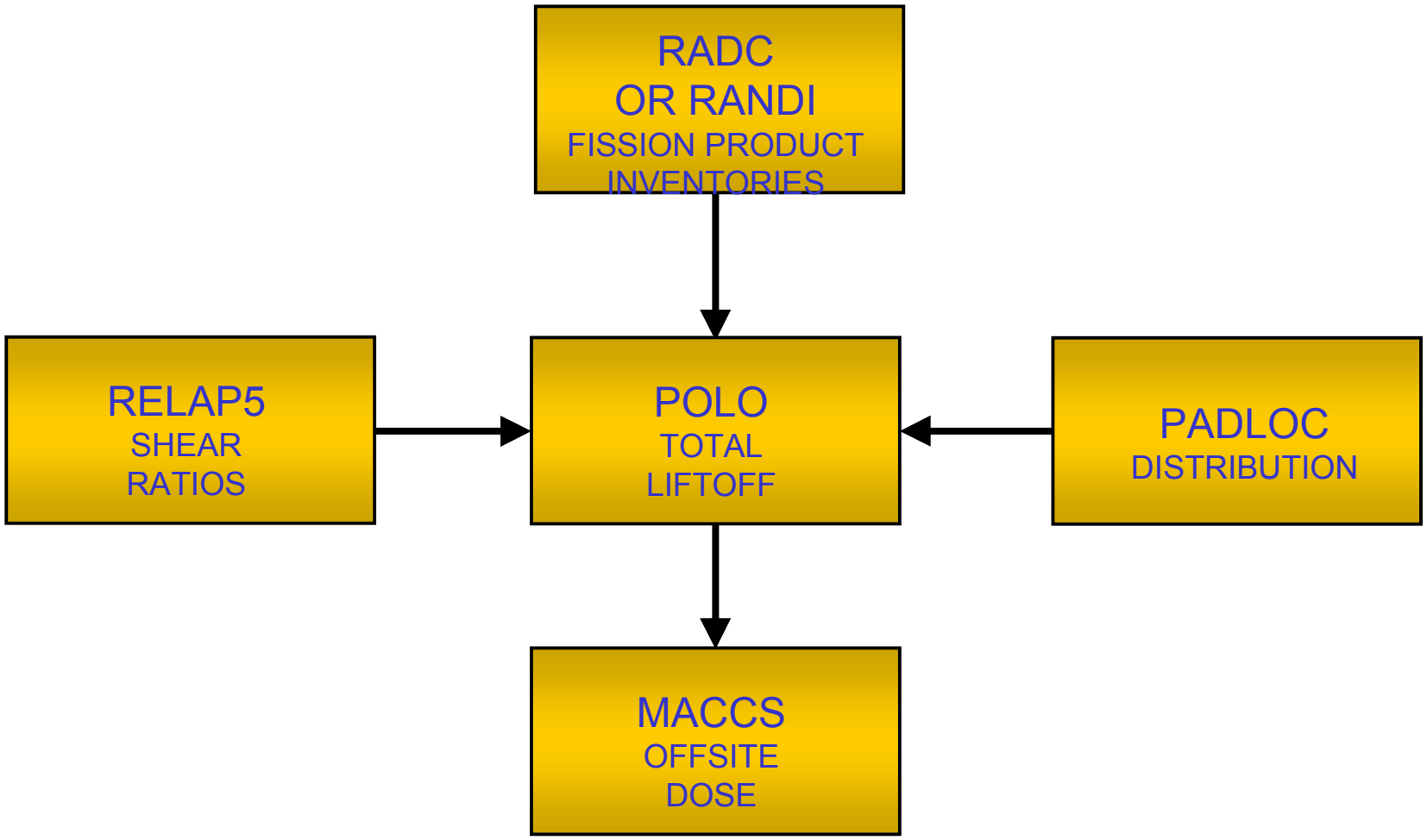
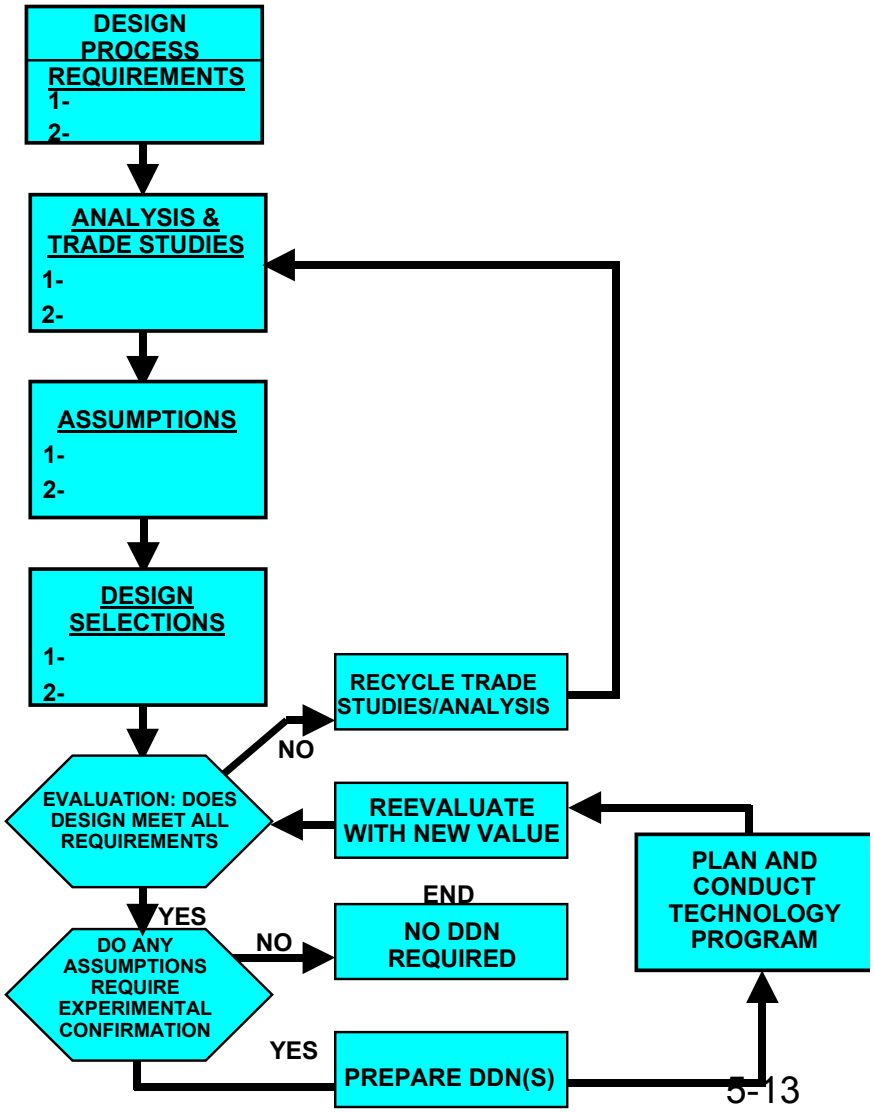
Core Heatup Design Methods



Depressurization Transient Design Methods



Methodology for Identifying Design Data Needs



- DDNs Are Identified as Part of the Functional Analysis Process
- Assumptions Are Made when Making Design Selections to Satisfy Requirements
- Certain Assumptions Require Technology Development before their Validity Can Be Determined

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GT-MHR Design Data Needs Radionuclide Transport

DDN No.	DDN Title	Priority
C.07.03.01	**Fission Gas Release from Core Materials**	**High**
C.07.03.02	Fission Metal Diffusivities in Fuel Kernels	Medium
C.07.03.03	**Fission Product Diffusivities in Particle Coatings**	**High**
C.07.03.04	**Fission Product Diffusivities/Sorptivities in Graphite**	**High**
C.07.03.05	Tritium Permeation in Heat Exchanger Tubes	Low
C.07.03.06	Tritium Transport in Core Materials	Low
C.07.03.07	**RN Deposition Characteristics for Structural Materials**	**High**
C.07.03.08	Decontamination Protocols for Turbine Alloys	Medium
C.07.03.09	**RN Reentrainment Characteristics for Dry Depressurization**	**High**

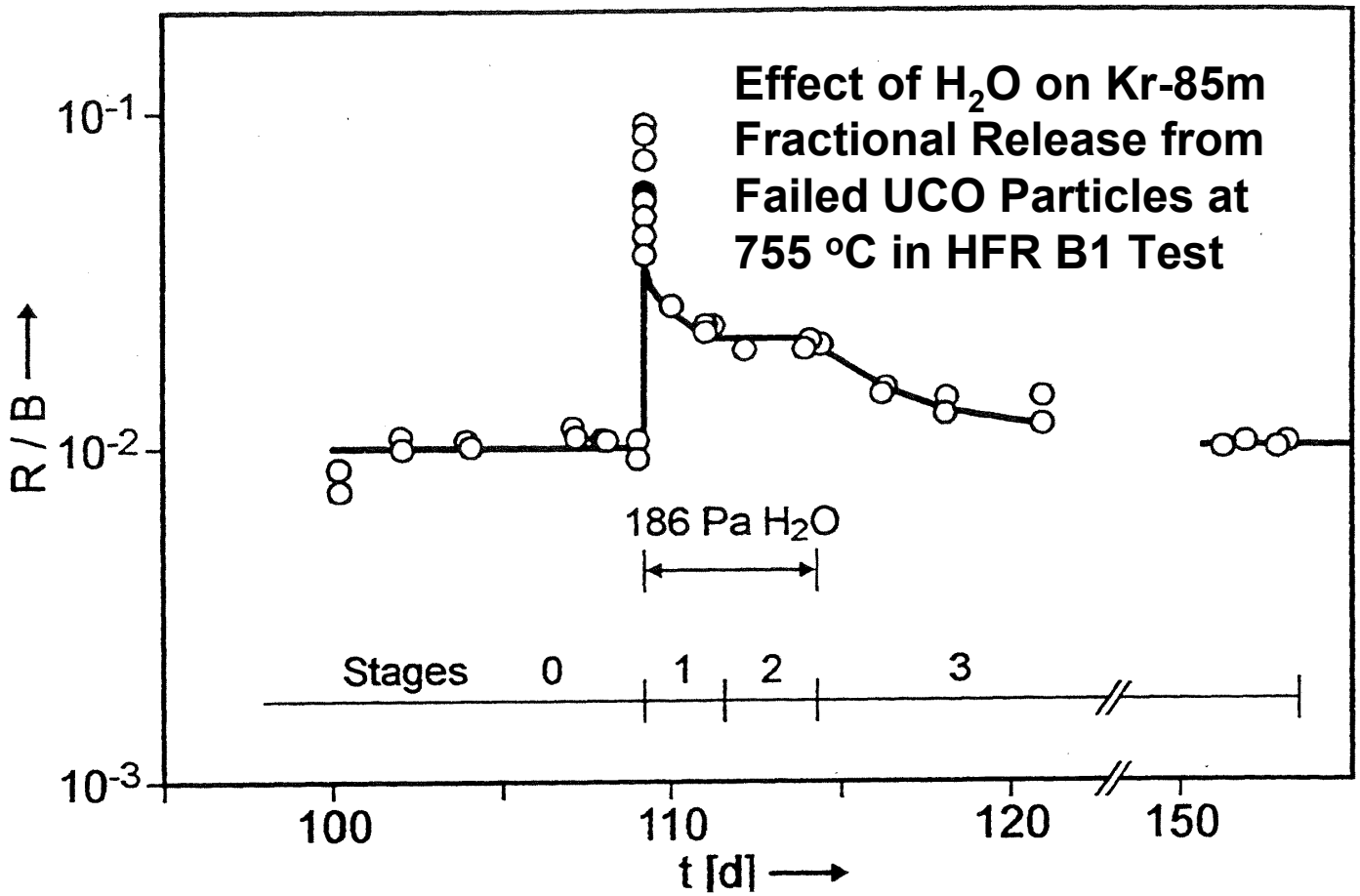
GT-MHR Design Data Needs RN Transport - Continued

DDN No.	DDN Title	Priority
C.07.03.10	RN Reentrainment Characteristics for Wet Depressurization	Low
C.07.03.11	Characterization of the Effects of Dust on RN Transport	Medium
C.07.03.12	**Fission Product Transport in Vented Low-Pressure Containment**	**High**
C.07.03.13	Decontamination Efficiency of Depressurization Train Filter	Medium
C.07.03.14	**Fission Gas Release Validation Data**	**High**
C.07.03.15	**Fission Metal Release Validation Data**	**High**
C.07.03.16	**Plateout Distribution Validation Data**	**High**
C.07.03.17	**Radionuclide "Liftoff" Validation Data**	**High**
C.07.03.18	Radionuclide "Washoff" Validation Data	Medium

C.07.03.01 Fission Gas Release from Core Materials

- **OBJECTIVE**
  - Increase Data Base for Fission Gas Release From Heavy-Metal Contamination and Exposed Fuel Kernels
  - Reduce Uncertainty in Models To <4x ($R_{95\%}/R_{50\%} < 4$)
- **EXISTING DATA BASE**
  - Gas Release from Heavy-Metal Contamination as Function of Element, Half Life, Temperature and Time
  - Gas Release from Exposed Kernels as Function of Element, Half Life, Temperature, Burnup, Hydrolysis State, and Time
  - Out-of-pile Measurements of Hydrolysis Effects
  - In-pile Measurements of Hydrolysis Effects

Fission Gas Release from Failed UCO
Effect of H_2O on Release



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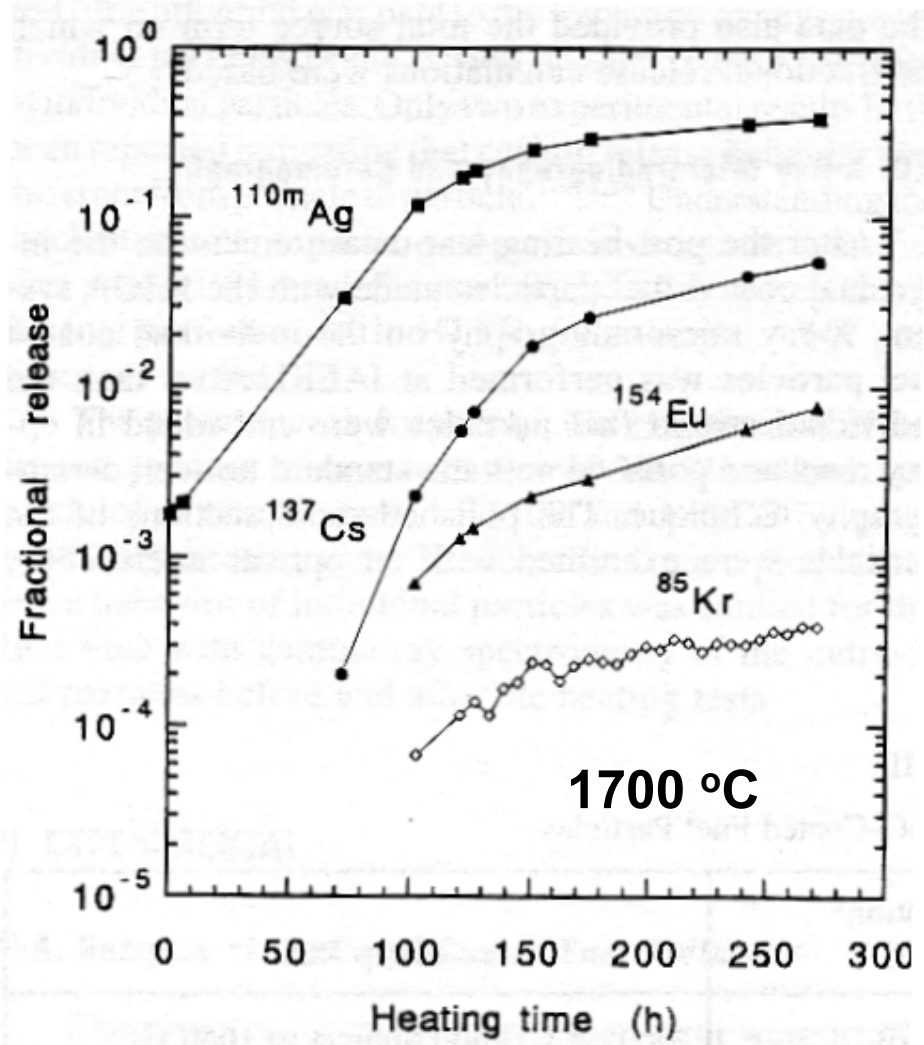
C.07.03.01 Fission Gas Release from Core Materials - Continued

- ADDITIONAL DATA NEEDED
  - Confirmation of Temperature and Burnup Dependence of Gas Release for Failed UCO TRISO Fuel
  - Extension of Hydrolysis Data Base to 1200 ^{o}C
  - Confirmation that I Isotopes Behave like Xe Isotopes
- PLANNED TECHNOLOGY PROGRAMS
  - In-pile Measurements of Gas Release from Fuel Compacts with Known Failure Fraction ("Designed-to-Fail Particles")
  - Measurement of Gas Release during Post-Irradiation Heating of Reference UCO Fuel
  - Direct Measurements of I Release to Confirm Analogy between I and Xe Isotopes

C.07.03.03 Fission Product Diffusivities in Particle Coatings

- **OBJECTIVE**
  - Increase Data Base for Transport of Fission Metals in SiC and of Fission Gases in PyC Coatings under Normal Operating and Core Heatup Conditions
  - Reduce Uncertainty In Correlations To <10x ($D_{95\%}/D_{50\%} < 4$)
- **EXISTING DATA BASE**
  - Cs and Sr Diffusivities in Pyrocarbon As Function of Temperature
  - Cs, Sr and Ag Diffusivities in SiC as Function of Temperature
  - Permeabilities of Fission Gases in PyC as Function of Temperature

Fission Product Release during Accident Simulation Test



- LEU UO_2 TRISO from HRB-22 (Japanese Fuel Test at ORNL)
- Fuel Compacts Irradiated in HFIR: Burnup ≤9% FIMA; T_{fuel} ≤1350 °C
- Postirradiation Heating Test at 1700 °C in High Temperature Furnace in Hot Cell
- Solid and Gaseous Fission Product Release vs. Time
- Release Profiles Indicate:
  - Ag diffusively released from intact TRISO
  - Kr retained by PyC coatings
  - Cs slowly released as SiC degrades

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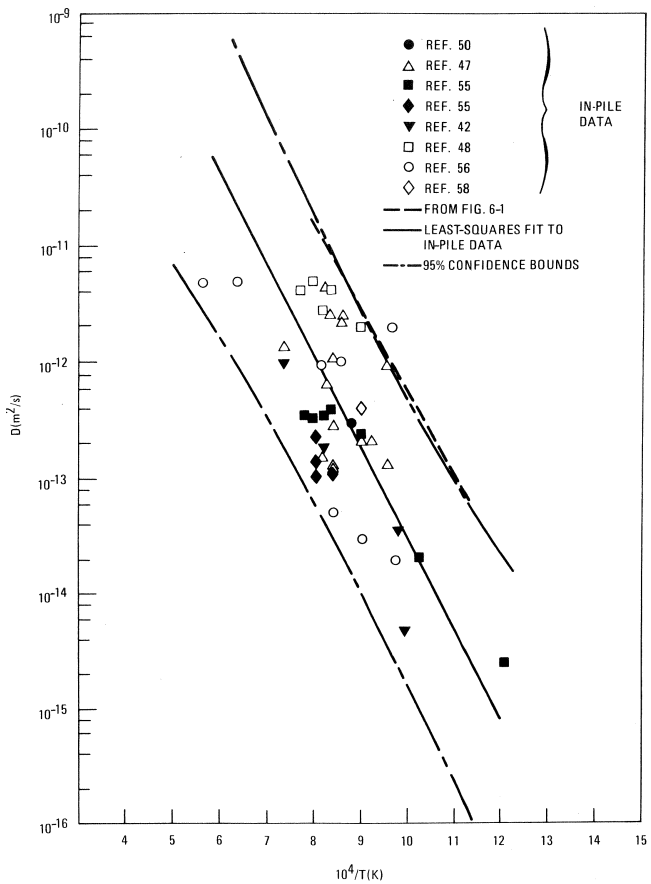
C.07.03.03 FP Diffusivities in Particle Coatings - Continued

- **ADDITIONAL DATA NEEDED**
  - Data for Ag, Cs and Sr in SiC under Normal Conditions and Core Heatup Conditions
  - Data for I, Xe, and Kr in PyC under Core Heatup Conditions
- **PLANNED TECHNOLOGY PROGRAMS**
  - Determination of Ag, Cs and Sr Diffusivities in SiC during Irradiation by PIE of Intact TRISO Particles with Different Irradiation Histories
  - Determination of Diffusivities of Ag, Cs and Sr in SiC and of Kr, Xe and I in PyC under Core Heatup Conditions by Post-Irradiation Heating of Intact And "Partially Failed" TRISO Particles (Failed SiC, Intact PyC)

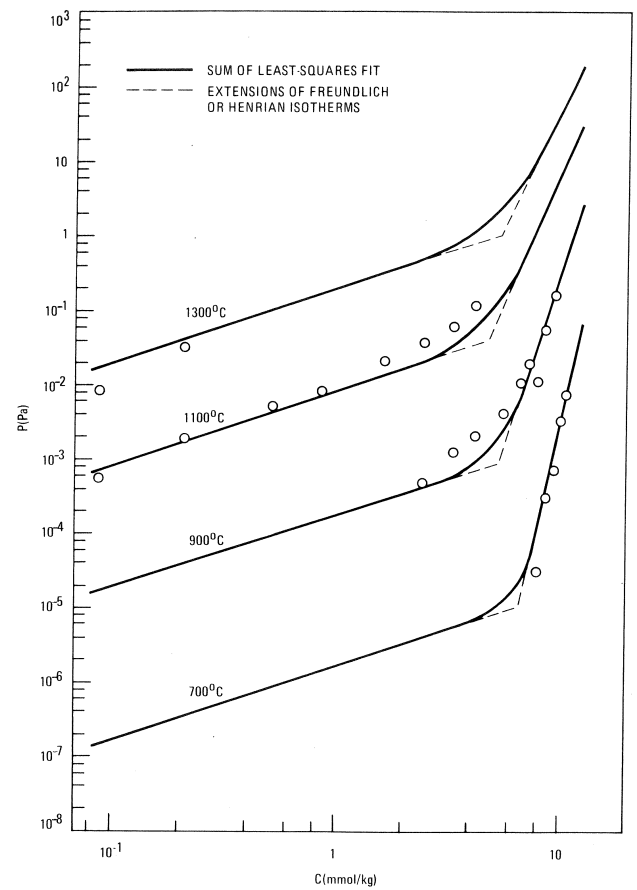
C.07.03.04 Fission Product Diffusivities/Sorptivities in Graphite

- **OBJECTIVE**
  - Increase Data Base for Diffusivities and Sorptivities of Cs, Sr and Ag in Reference Graphite and Fuel-Compact Matrix under GT-MHR Specific Conditions
  - Reduce Uncertainty in Models to <10x
- **EXISTING DATA BASE**
  - Diffusivities of Cs and Sr in Unirradiated & Irradiated Graphite as Function of Temperature
  - Sorptivities of Cs and Sr on Unirradiated and Irradiated Fuel-Compact Matrix as Function of Temperature, Partial Pressure, and Fluence
  - FRG and British Data on Ag Transport in Nonreference Graphites

Cs Transport in Nuclear Graphite Has Been Characterized



Cs Diffusion in Nuclear Graphites



Cs Sorption on H-451 Graphite

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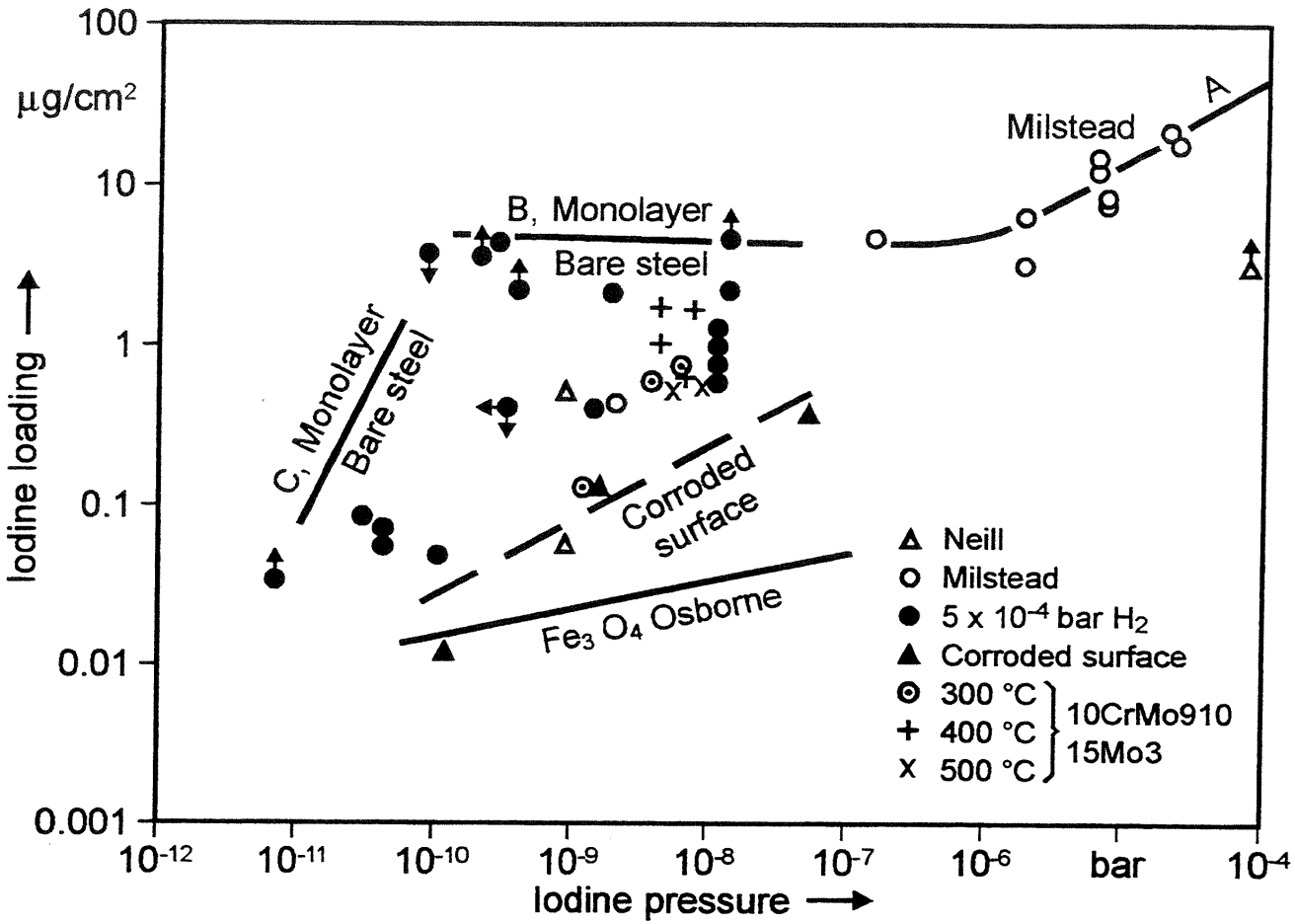
C.07.03.04 Fission Product Diffusivities/-Sorptivities in Graphite - Continued

- **ADDITIONAL DATA NEEDED**
  - Temperature and Fluence Dependence of Graphite Diffusivities and Sorptivities between 300 - 1800 ^{o}C
  - Effect of Burnoff on Diffusion
  - Effect of Coolant Chemistry on FP Transport in Graphite (H_2O & CO)
- **PLANNED TECHNOLOGY PROGRAMS**
  - Laboratory Sorptivity Measurements at Representative Partial Pressures with Reference Matrix and Graphite
  - Inpile Cs and Sr Transport Experiments with Reference Matrix and Graphite
  - Post-Irradiation Heating of Graphite Bodies

C.07.03.07 RN Deposition Characteristics for Structural Materials

- **OBJECTIVE**
  - Measure Sorptivity of Volatile Radionuclides (Ag, Cs, I, Te) on Structural Metals under GT-MHR Specific Conditions
  - Measure "In-Diffusion" of Ag and Cs into Turbine Blade Alloys under GT-MHR Service Conditions
  - Reduce Uncertainties in Correlations to <10x ($S_{95\%}/S_{50\%} < 10$)
- **EXISTING DATA BASE**
  - Sorptivities for Cs, Ag And I on Selected Circuit Alloys as Function of Temperature, Partial Pressure and Oxidation State
  - FRG Data on Diffusion of Plateout Activity into Structural Metals at High Temperatures (>600 ^{o}C)

Iodine Sorption on Low-Alloy Steel at 400 °C



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C.07.03.07 RN Deposition Characteristics for Structural Materials - Continued

- **ADDITIONAL DATA NEEDED**
  - Sorption Isotherms for Ag, Cs, I and Te on Reference Alloys at Representative Partial Pressures and Temperatures
  - Diffusivities for Ag and Cs in Reference Turbine Blade Alloys
  - Determination of Effects of Dust on Plateout (DDN C.07.03.11)
- **PLANNED TECHNOLOGY PROGRAMS**
  - Laboratory Sorptivity Measurements for Ag, Cs, I and Te at Partial Pressures $<<10^{-10}$ atm
  - Measurements of Ag and Cs Diffusion Profiles in Metal Specimens from In-Pile Loop Tests
  - Single-Effects Measurements of Plateout as Function of System Parameters in Out-of-Pile Loop
  - Measurements of Dust Effects in Out-of-Pile Loop (DDN C.0703.11)

C.07.03.09 RN Reentrainment Characteristics for Dry Depressurization

- **OBJECTIVE**
  - Obtain Additional In Situ Data for Fission Product Liftoff (I, Cs, Te, and Ag) during Depressurization Transients
  - Reduce Uncertainty in Liftoff Model to <10x
- **EXISTING DATA BASE**
  - Liftoff of Sr, I, Cs, and Ce vs. Shear Ratio ($T_{W,BD}$/ $T_{W,NOC}$) from Ex Situ Blowdown Tests with Specimens
    - Gail IV Loop
    - GA Deposition Loop
    - Peach Bottom Steam Generator Tube Specimens
    - CPL 2 Inpile Loop (4 Tests)
  - Japanese Ex-Situ Liftoff Data to Very High Shear Ratios (>100)

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C.07.03.09 RN Reentrainment Characteristics for Dry Depressurization - Continued

- **ADDITIONAL DATA NEEDED**
  - An Extended Data Base for the Fractional Liftoff of I, Sr, and Cs as Function of Shear Ratio, Temperature, and Time
  - Determination of Effects of Dust on Liftoff (DDN C.07.03.11)
- **PLANNED TECHNOLOGY PROGRAMS**
  - Single-Effects In Situ Measurements of Liftoff as Function of System Parameters in Out-of-Pile Loop
  - Measurements of the Effects of Dust on Liftoff in Out-of-Pile Loop (DDN C.07.03.11)

C.07.03.12 Radionuclide Transport in Vented Low-Pressure Containment

- **OBJECTIVE**
  - Increase Data Base for Describing Radionuclide Transport in Reactor Building under Core Conduction Cooldown Conditions
  - Reduce Uncertainty in Models for Transport in VLPC to <10x
- **EXISTING DATA BASE**
  - LWR Data Base, including Integral Data (e.g., DEMONA Tests in FRG, MARVIKEN Tests In Sweden, LACE in the USA, etc) on RN Transport in LWR Containment Buildings
  - Partition Coefficients for Iodine In Steam/Liquid H_2O Systems
  - US and FRG Data on Aerosols in Operating HTRS
  - Iodine and Cesium Sorptivities on Low-Alloy Steels as Function of Temperature and Partial Pressure

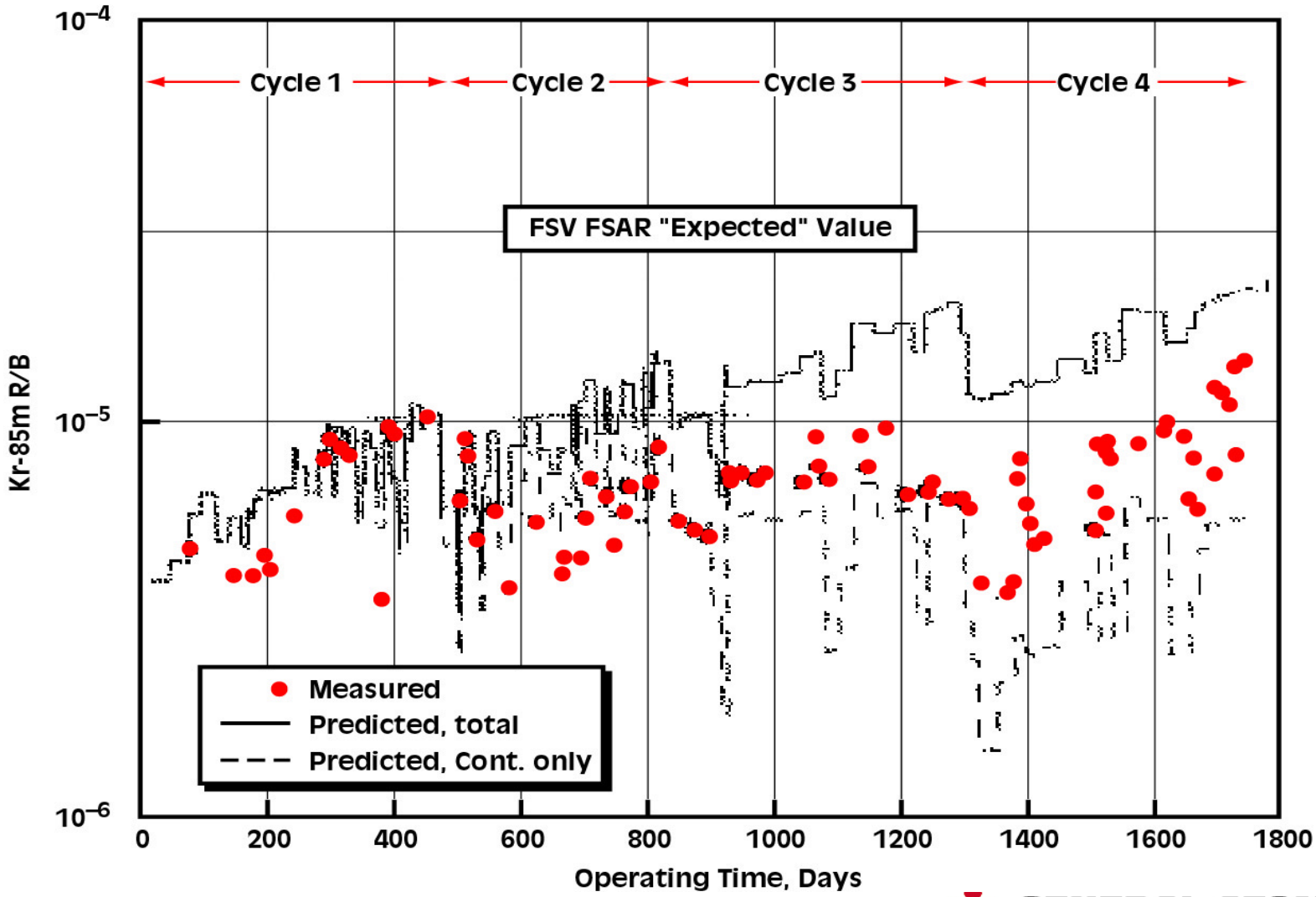
C.07.03.12 RN Transport in Vented Low-Pressure Containment - Continued

- ADDITIONAL DATA NEEDED
  - Data Describing Fission Product Transport in Reactor Building, including Plateout, Condensation & Settling under GT-MHR Specific Conditions
  - Independent Data Base to Validate the Design Methods for Predicting Radionuclide Transport in VLPC
- PLANNED TECHNOLOGY PROGRAMS
  - Determination of Applicability of Selected LWR Data to GT-MHR
  - Measurement of Iodine Sorptivities on Steels and Painted Surfaces between 30 and 360 ^{o}C
  - GT-MHR Specific Test Programs as Required

C.07.03.14 Fission Gas Release Validation Data

- **OBJECTIVE**
  - Obtain Data to Validate that Design Methods for Predicting Fission Gas Release from Core during Normal Operation ("SURVEY") and LBEs ("SORS") Are Accurate to within 4x ($R_{95\%}/R_{50\%} \leq 4$)
- **EXISTING DATA BASE**
  - Gas Release from Fort St. Vrain (TRISO Fuel)
  - Extensive International Data on Release from TRISO Oxide Fuels
  - In-Pile Gas Release from LEU UCO Fuel under Dry Conditions to 1250 ^{o}C from Irradiation Capsules
  - Gas Release from LEU UCO in In-Pile CEA COMEDIE Loop
  - Gas Release From LEU UCO Fuel as Function of Temperature, Burnup and Time during Post-Irradiation Heating Tests

Comparison of FSV Predicted and Measured Kr-85m Release



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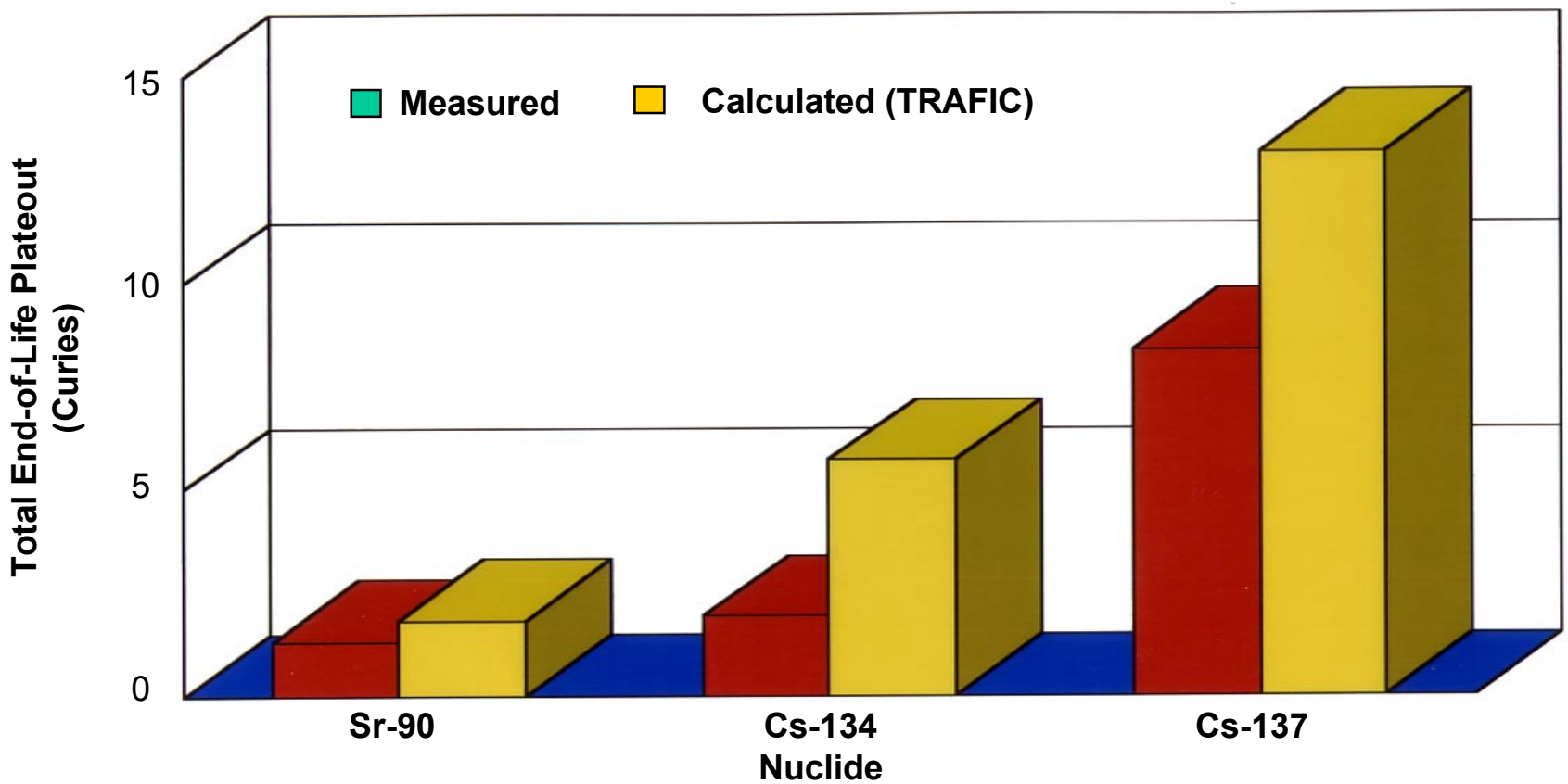
C.07.03.14 Fission Gas Release Validation Data - Continued

- ADDITIONAL DATA NEEDED
  - An Independent Data Base to Confirm that Gas Release Models and Codes Are Accurate to within 4x
- PLANNED TECHNOLOGY PROGRAMS
  - Irradiation Capsules with Reference Fuel with Known Failure Fraction
  - In-Pile Loop Tests at High Pressure and High Flow
  - Post-Irradiation Heating of Fuel with known Failure Fraction

C.07.03.15 Fission Metal Release Validation Data

- **OBJECTIVE**
  - Obtain Data to Validate that Design Methods for Predicting Fission Metal Release from Core during Normal Operation (”TRAFIC") and LBEs ("SORS") Are Accurate to within 10x
- **EXISTING DATA BASE**
  - Metal Release from Peach Bottom Core 2
  - Metal Release from FSV (Two Plateout Probes)
  - International Data on Metal Release from Oxide Fuels, especially in Fuel Spheres
  - Metal Release Data from Post-Irradiation Heating Tests
  - Metal Release from LEU UCO in In-Pile CEA COMEDIE Loop

FSV Fission Metal Plateout Predictions



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

. . . Predictions of Metallic Plateout are Accurate and Conservative

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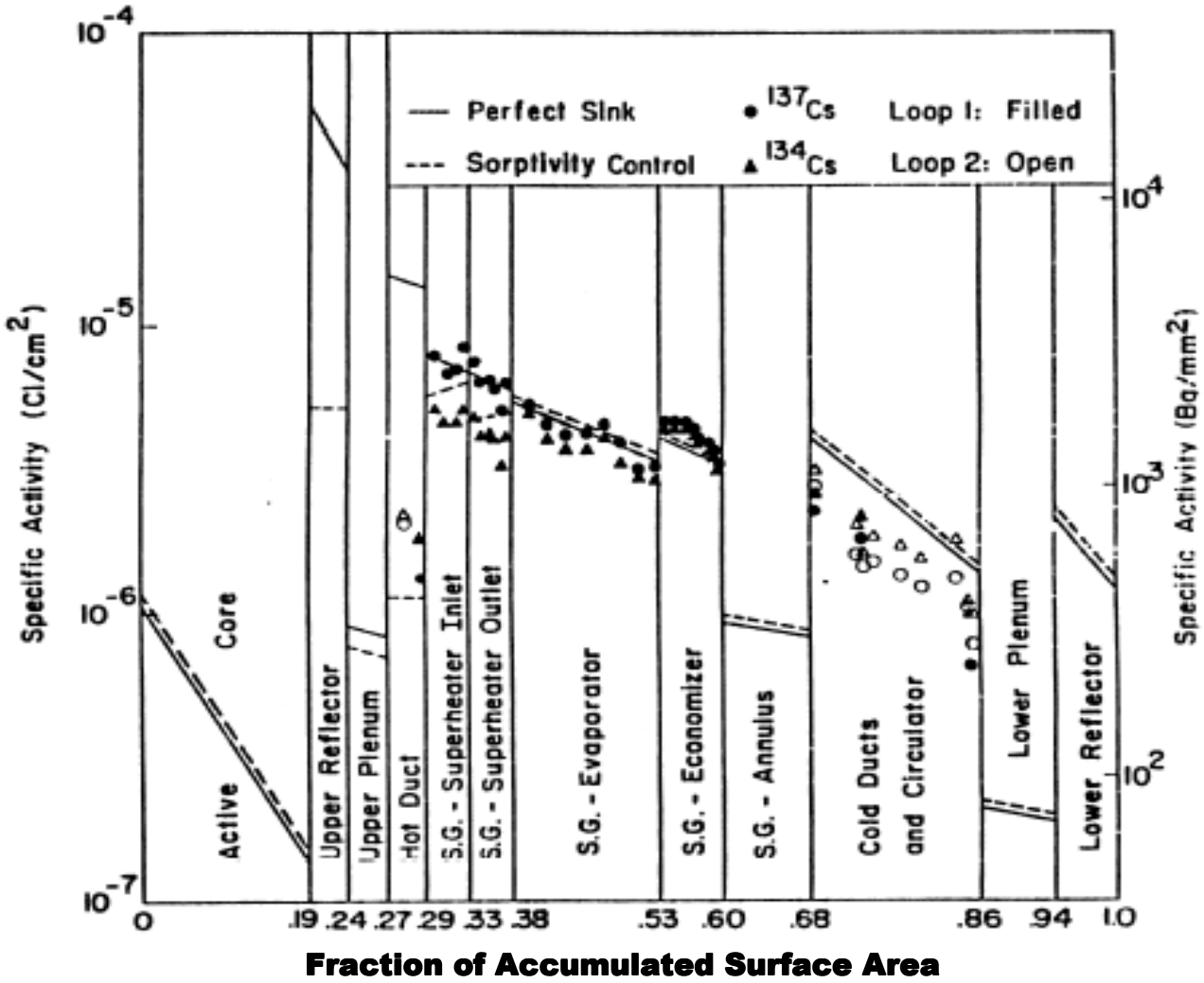
C.07.03.15 Fission Metal Release Validation Data - Continued

- ADDITIONAL DATA NEEDED
  - An Independent Data Base to Confirm that Metal Release Models and Codes Are Accurate to within 10x
- PLANNED TECHNOLOGY PROGRAMS
  - Irradiation Capsules with Reference Fuel with Known Failure Fraction
  - In-Pile Loop Tests at High Pressure and High Flow
  - Post-Irradiation Heating of Fuel with known Failure Fraction

C.07.03.16 Plateout Distribution Validation Data

- **OBJECTIVE**
  - Obtain Experimental Data To Validate That the Plateout Distribution Code "PADLOC" Is Accurate to within 10x
- **EXISTING DATA BASE**
  - Cs and Sr Plateout Distributions in Peach Bottom
  - FSV Plateout Probes and Circulator Gamma Scan Data
  - Plateout Distributions of I, Cs and Sr in CPL 2 Loop Tests (four)
  - FRG Plateout Data from In-Pile and Out-of-Pile Loop Tests
  - Plateout Distributions of Ag, Cs, I, and Sr SR1 and BD1 Tests in the CEA COMEDIE Loop
  - Japanese Data From the OGL-1 Loop

Measured and Predicted Cs Plateout Distributions in Peach Bottom HTGR



C.07.03.16 Plateout Distribution Validation Data - Continued

- ADDITIONAL DATA NEEDED
  - An Independent Data Base to Confirm that the PADLOC Code Is Accurate to within 10x, including "In-diffusion" Effects
  - Confirmation that Dust Effects Are Insignificant in GT-MHR
- PLANNED TECHNOLOGY PROGRAMS
  - Three Tests in a High-Pressure, High-Flow Inpile Loop followed by PIE to Determine Plateout Distributions under Representative GT-MHR Conditions
  - Measurement of Effects of Dust in these Tests, including Injection of Representative Dust in One Test
  - Measurement of the Amount of Indiffusion

C.07.03.17 Radionuclide "Liftoff" Validation Data

- **OBJECTIVE**
  - Obtain Experimental Data to Validate that Design Methods ("POLO") for Predicting Radionuclide Liftoff during Dry Depressurization Transients Are Accurate to within 10x
- **EXISTING DATA BASE**
  - Liftoff Of Sr, I, Cs and Ag from In Situ Blowdown of CPL 2/4 Inpile Loop (Shear Ratio = 1.02)
  - Liftoff Of Sr, I, Cs and Ag from Four In Situ Blowdowns of the COMEDIE BD-1 Inpile Loop (Shear Ratio = 0.7, 1.7, 2.8, 5.6)

Measured and Predicted Liftoff in COMEDIE BD-1 Loop Test

Nuclide		Cumulative Liftoff Fraction (%)			
		SR = 0.7	SR = 1.7	SR = 2.8	SR = 5.6
I-131	**Meas.**	**0.077**	**0.10**	**0.11**	**0.13**
	Pred.	0.15	0.16	0.53	2.1
Cs-137	**Meas.**	**0.014**	**0.021**	**0.030**	**0.11**
	Pred.	0.19	0.29	0.48	1.1
Cs-134	**Meas.**	**0.015**	**0.020**	**0.028**	**0.096**
	Pred.	0.19	0.29	0.48	1.1
Ag-110m	**Meas.**	**0.015**	**0.019**	**0.043**	**0.23**
	Pred.	0.010	0.32	0.90	2.8
Sr-90	**Meas.**	**0.16**	**0.36**	**0.56**	**0.74**
	Pred.	0.54	0.56	1.2	4.2

C.07.03. 17 Radionuclide "Liftoff" Validation Data - Continued

- **ADDITIONAL DATA NEEDED**
  - An Independent Data Base to Confirm that POLO Code Is Accurate to within 10x
  - Confirmation that Dust Effects Are Insignificant in GT-MHR
- **PLANNED TECHNOLOGY PROGRAMS**
  - Three Inpile Loop Tests including Multiple In Situ Blowdowns of Loop at Successively Higher Shear Ratios
  - Measurement of the Effects of Dust in these Tests, including Injection of Representative Dust in at least One Test

Conclusions

- Analytical Tools Available to Predict Radionuclide Transport from Fuel Kernel to Site Boundary
- Codes Used Extensively for Reactor Design & Analysis, including Fort St. Vrain HTGR
- DDNs to Validate Design Methods for Predicting Source Terms Have Been Systematically Identified
  - Differential Test Data for Improved Component Models
  - Independent Integral Data for Code Validation

Outcome Objectives

- NRC concurs with or feedback on the adequacy of Radionuclide Transport DDNs to validate methods for quantifying source terms

Summary of Objectives

- Acceptance and/or feedback on the logic and adequacy of the approach described
  - Quantifying and validating source term (defining Design Data Needs - DDNs)
  - Determining fuel requirements
- Agreement or feedback on adequacy of the defined DDNs supporting source term including
  - Coated particle fuel quality and performance
  - Radionuclide transport mechanisms across barriers