U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Presentations for August 3, 2017 Public Meeting
Regulatory Improvements for Advanced Reactors

1) NRC Slides
   - Opening
   - Implementation Action Plans
   - Prototype Testing (Draft White Paper)
   - Miscellaneous Topics / Fuel Qualification

2) NEI Slides
   - Industry Views on Draft Prototype Guidance
   - Regulatory Engagement Plans
   - Possible Example for Framework Matrix (Functional Containment)

3) TRISO Particle Fuel Qualification

4) Metallic Fuels Irradiation Database and Data Qualification

5) Fuel Qualification for Molten Salt Reactors

Advanced Reactor Program

Public Meeting on Possible Regulatory Process Improvements for Advanced Reactor Designs

August 3, 2017

Telephone Bridge
(800) 857-9782
Passcode: 2382665

Public Meeting

- Telephone Bridge

  (800) 857-9782

  Passcode: 2382665

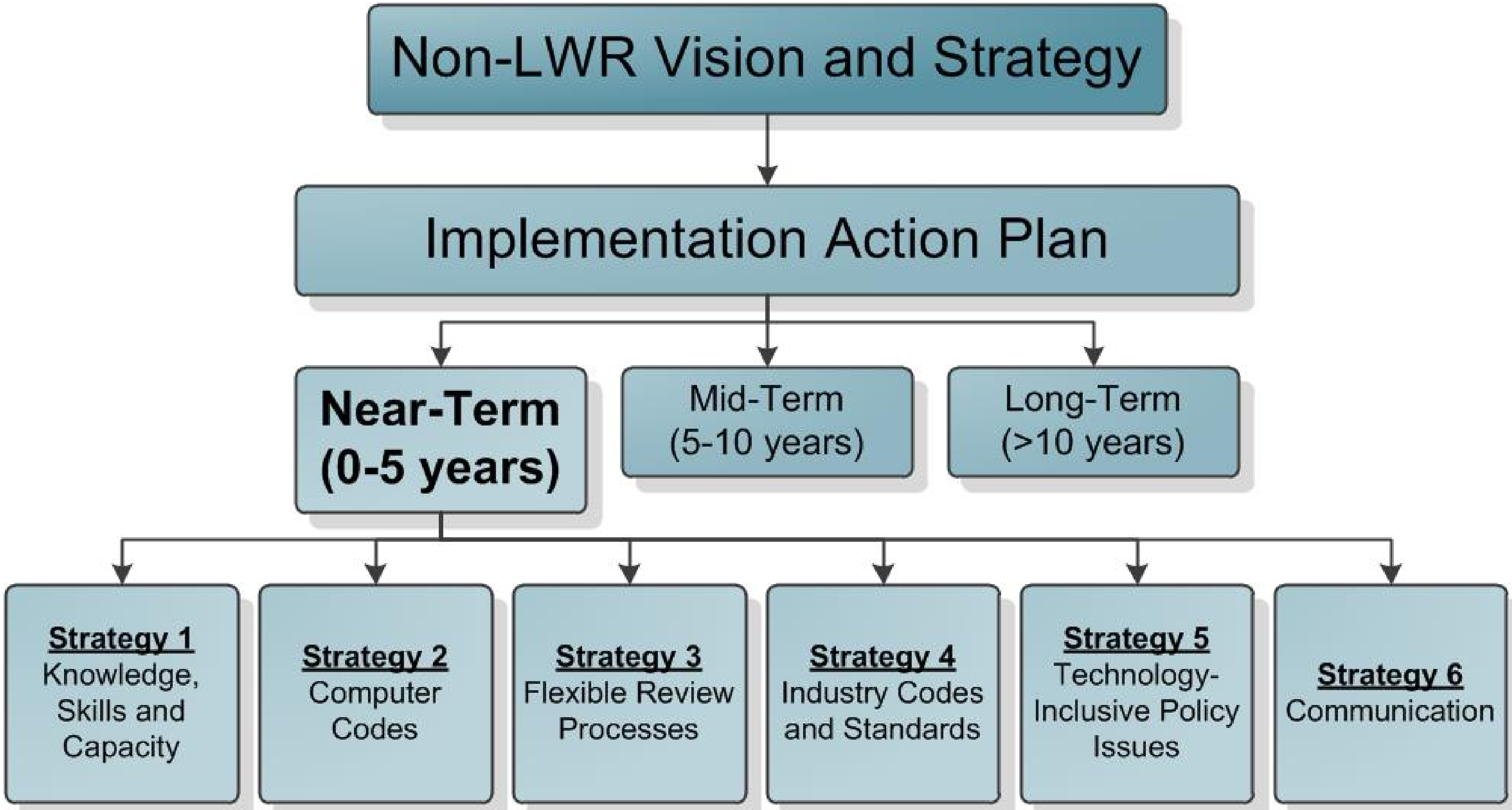
- Opportunities for public comments and questions at designated times

Outline

- Implementation Action Plans
  - Near-Term (ML17165A069) & Mid, Long-Term (ML17164A173)
- Prototype Testing (ML17025A353)
- Regulatory Engagement Plans (NEI)
- Lunch
- Fuel Qualification
  - TRISO Fuel (INL and Industry)
  - Metal Fuel (ANL)
  - Molten Salt (ONRL)
- Miscellaneous Topics
  - Planning for Upcoming Meetings/Interactions

Public Comment Period

Implementation Action Plans (IAPs)



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Near-Term IAP Strategies

Strategy	Activities/Goal	Interactions
1	• Core Team Approach • MSR Training • Knowledge Management • Ongoing Gap Assessments	• DOE and National Labs • Technology working groups • Pre-application • International
2	• Assess current capabilities • Acquire/develop computer codes and tools • Identify experimental data needs	• DOE and National Labs • International (IAEA, NEA) • EPRI (workshops) • Technology working groups • Pre-application

Advanced Reactor Program

Implementing IAP Strategies

Strategy	Activities/Goal	Interactions
3	• Flexible Framework o ARDC (Aug 24) o Functional Containment o ACRS (early 2018) • Roadmap • Prototype • Regulatory engagement plans	• Licensing Modernization Project • DOE and National Labs • NEI (white papers, guidance document) • NIA (SDA white paper) • Pre-applications
4	• ASME • ANS • Non-LWR PRA Standard • Standards Forum	• SDOs • DOE and National Labs • NEI, EPRI • Technology working groups • Pre-applications

Implementing IAP Strategies

Strategy	Activities/Goal	Interactions
5	• Policy Issue Table • Rulemaking actions • White papers	• DOE and National Labs • NEI (white papers) • Technology working groups • Pre-applications
6	• Develop timely, clear requirements, guidance • Consistent messaging • Promote exchange of experience, information o Stakeholder meetings	• DOE and National Labs • International (IAEA, NEA) • NEI, NIA, NIC • Technology working groups • Pre-applications

Mid- and Long-Term IAPs

Strategy		Near-Term	Mid-Term	Long-Term
1	NRC Knowledge, Skills and Capabilities	→	→	
2	NRC Computer Codes and Tools	→	→	
3	Flexible Review Process	→	→	→
4	Industry Codes and Standards	→	→	
5	Policy Issues	→	→	
6	Communication	→	→	→

Prototype Testing

Regulatory Engagement Plans

Morning Wrap Up

- Public Comments / Questions
- Lunch
- Afternoon Session
  - Fuel Qualification
    - TRISO
    - Metal
    - Molten Salt
  - Miscellaneous Topics
    - Planning – Activities & Future Meetings

Miscellaneous Topics

- GAIN Website (regulatory questions)
- Siting (proximity to population centers)
- Licensing Modernization Project
  - PRA Approaches White Paper
  - Framework (Table/Matrix)
- Physical Security White Paper
- Enrichments
- Insurance/Liability (Tentative - Nov 2)
- International Activities

Fuel Qualification

Fuel qualification is a process which provides high confidence that physical and chemical behavior of fuel is sufficiently understood so that it can be adequately modeled for both normal and accident conditions, reflecting the role of the fuel design in the overall safety of the facility. Uncertainties are understood such that any calculated fission product releases include appropriate margin to ensure conservative calculation of radiological dose consequences.

Nuclear Power Reactor Testing Needs and Prototype Plants for Advanced Reactor Designs

George Tartal

Senior Project Manager

NRO/DSRA/ARPB

Background

- NPP Standardization Policy Statement (1987)
  – "For those SSC designs which represent significant deviations from previously-approved LWR designs, prototype testing and/or empirical information may also be required."
  – "When an advanced design concept is sufficiently mature, e.g., through comprehensive, prototypical testing, an application for design certification could be made."
- Advanced Reactor Policy Statement (1986, 1994, 2008)
  – "Applicants are responsible for documentation and research necessary to support a specific application. Research activities would include testing of new safety or security features..."
  – "The testing shall ensure that these new features will perform as predicted, will provide for the collection of sufficient data to validate computer codes, and will show that the effects of system interactions are acceptable."

Background (cont'd)

- Prototype testing regulations in place since 1989
  - Moved from Part 52 to Part 50 in 2007
  - No formal guidance issued
- In 1991, NRC staff developed SECY-91-074
  - Described the process for applicant to decide what types of testing are needed (prototype is last step)
  - Information paper; no Commission SRM or follow-up
- No COL or OL license has ever been issued requiring prototype testing under 50.43(e)(2)

Background (cont'd)

- Was identified as potential policy issue in SECY-10-0034
- Staff concluded in SECY-11-0112 that no rulemaking or policy changes were needed
- Staff determined that guidance may be beneficial to support implementation of the prototype provisions
- Preliminary draft document
  - ADAMS Accession No. ML17025A353
- Objective: Agree on path forward

Regulation – 10 CFR 50.43(e)

- (e) Applications for a design certification, combined license, manufacturing license, or operating license that propose nuclear reactor designs which differ significantly from light-water reactor designs that were licensed before 1997, or use simplified, inherent, passive, or other innovative means to accomplish their safety functions, will be approved only if:
  - (1)(i) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;
  - (ii) Interdependent effects among the safety features of the design are acceptable, as demonstrated by analysis, appropriate test programs, experience, or a combination thereof; and
  - (iii) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions; or
  - (2) There has been acceptable testing of a prototype plant over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If a prototype plant is used to comply with the testing requirements, then the NRC may impose additional requirements on siting, safety features, or operational conditions for the prototype plant to protect the public and the plant staff from the possible consequences of accidents during the testing period.

Terminology

- Terminology Related to Facility Types
  - Prototype Plant
  - First-of-a-Kind Reactor
  - Demonstration Reactor
  - Non-Power Reactor
  - Test Reactor
  - Research Reactor
  - Production Facility
  - Utilization Facility

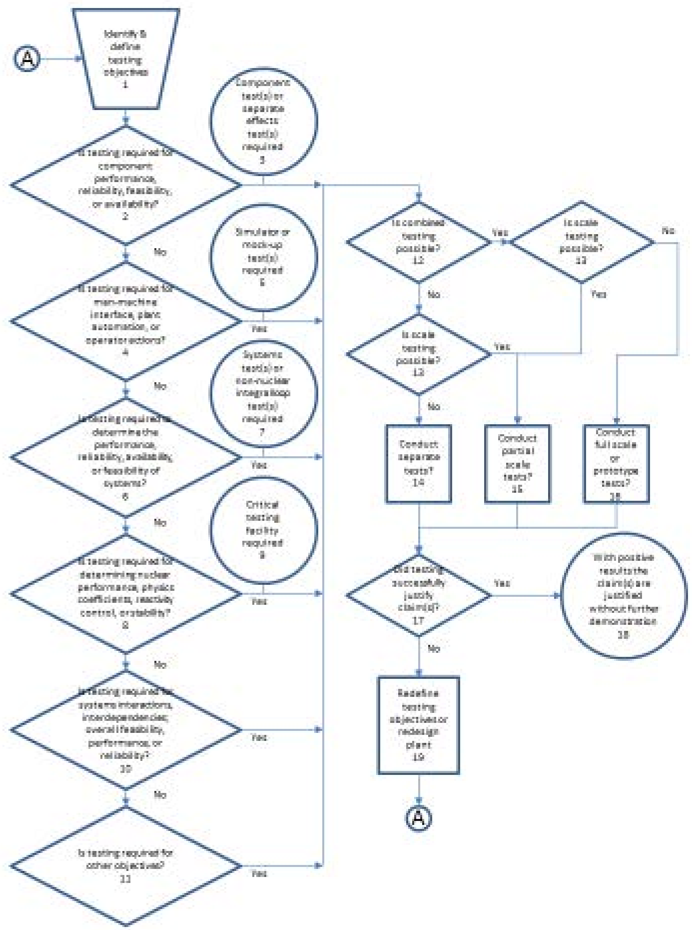
Terminology (cont'd)

- Categories of Tests Performed by Licensees
  - Preoperational Test
  - Initial Startup Test
  - ITAAC
  - Integral Effects Test
  - Separate Effects Test
  - Prototype Test

How to Determine Testing Needs

- SECY-91-074, Enclosure 2, described the process
  - Included in Appendix A to the white paper with clarifications and annotations
- Applicants and licensees should be thinking about testing plans as early as possible in their regulatory engagement plans and during pre-application discussions with the NRC

Testing Needs (cont.)



Process Description

1. Identify and define testing objectives.
2. Is testing required for component performance, reliability, feasibility, or availability?
3. Component test(s) or separate effects test(s) are required.
4. Is testing required for man-machine interface, instrumentation information transfer, plant automation, or operator actions?
5. Simulator or mock-up test(s) are required.

Process Description (cont'd)

6. Is testing required to determine the performance, reliability, availability, or feasibility of systems?
7. Systems test(s) or non-nuclear integral loop test(s) are required.
8. Is testing required for determining nuclear performance, physics coefficients, reactivity control, or stability?
9. Critical testing facility is required.

Process Description (cont'd)

10. Is testing required for systems interactions, interdependencies, overall feasibility, integrated system performance, or reliability?
11. Is testing required for other objectives?
12. Is combined testing possible?
13. Can test(s) objective(s) be demonstrated with scale test(s)?
14. Conduct separate test(s).

Process Description (cont'd)

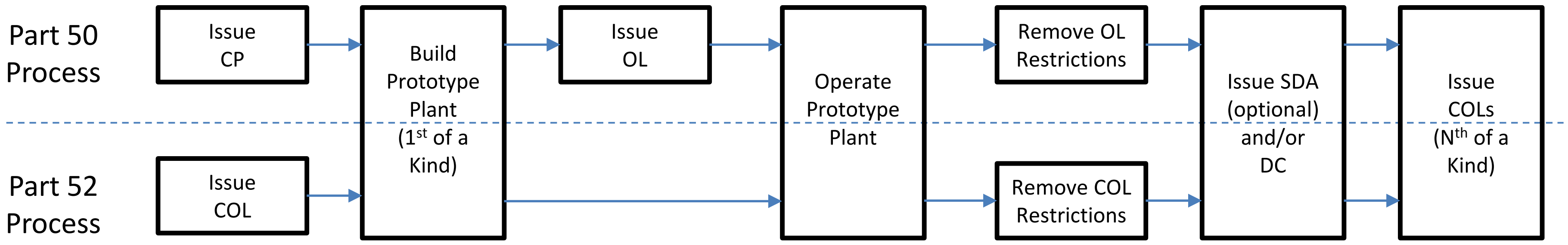
15. Conduct partial scale test(s).
16. Conduct full-scale integrated test(s) or prototype test.
17. Did the testing successfully justify the safety claims?
18. The safety claims are justified.
19. Redefine the testing objective(s) or redesign the plant.

Other Topics

- How Do I Determine Whether a Prototype Plant Is Needed?
- Is a Prototype Plant Needed To Perform Fuel and Materials Qualification Testing?
- Can the NRC Determine That an Application Must Be Submitted For a Prototype Plant?
- When Would the NRC Impose Additional Requirements on a Prototype Plant?

Other Topics (cont'd)

- How Would a Prototype Plant Fit into a Licensing Project Plan (Regulatory Engagement Plan)?
  - More detail described in Appendix B to the white paper



Other Topics (cont'd)

- How Would an Application Differ for a Reactor Design with a Prototype?
- How Would the License Issued, or the NRC's Safety Conclusions in Its Safety Evaluation, Differ for a Prototype Plant?
- How Is the Prototype Testing Period Determined?
- Is It Possible to License a Smaller Scale Reactor In Lieu of a Prototype Plant?
- How Would Prototype Testing Be Done for a Multi-module Facility?

Path Forward

- Is guidance on prototypes needed?
- What should the final product be? (Reg guide, NUREG, SRP, other)
- What parts of the white paper are more or less important?
  - What work should be done?
  - Who should lead the work?
  - Should we form a working group?
- Schedule target(s)?
- Follow-up public meetings needed?

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Initial Industry Feedback on NRC Draft: Nuclear Power Reactor Testing Needs and Prototype Plants for Advanced Reactor Designs

Edward M Burns
Advanced Reactor Regulatory Task Force

August 3, 2017

GENERAL COMMENTS

- The draft guidance presents a useful starting point for understanding how prototype controls are applied.
- The following comments are provided as initial insights.
- NEI is considering preparing a letter to the NRC that will provide industry comments.

NEI

MANAGEMENT OF PROTOTYPE REQUIREMENTS

- General
- Comment
  - What are the mechanics of imposing and removing prototype requirements?

QUALITY REQUIREMENTS FOR TEST DATA

- Page 8 states: “In particular, test data for a commercial nuclear power plant must be shown to meet quality assurance criteria commensurate with those in Appendix B to 10 CFR Part 50.”

- Comment
  - This provision is not necessary and could prove difficult, if not impossible, to satisfy in practice. Instead, NRC should accept the use of operating experience and test data from non-NRC licensed plants, provided that the applicant demonstrates that the information is reliable.

NEI

USE OF PRIOR OPERATING EXPERIENCE

- Page 11 states: “In accordance with 10 CFR 50.43(e)(1), testing is required to demonstrate that new safety systems function satisfactorily in accordance with the safety analysis.”
- Comment
  - This statement does not recognize the full range of described options, e.g., “analysis, appropriate test programs, experience, or a combination thereof”. For example, NUREG-1226 provides an extensive discussion on NRC expectations for the use of prior experience from NRC licensed and non-licensed (international) operating plants.

NEI

PROTOTYPE LICENSING PROCESS FIGURE

- Page 12 and 13 states: “The simplified prototype licensing process is depicted in Figure 1 below… Because of the variety of approval, licensing, and certification options presented in 10 CFR Part 52 and the combinations within that part and with those of 10 CFR Part 50, numerous possible approaches are available.”
- Comment
  - Figure 1 presents a notional depiction of ONE licensing approach. Some refinement is needed to ensure the figure and associated text are not taken as the ONLY licensing approach.

NEI

PROTOTYPE TESTING PERIOD

- Page 13 states: “[T]he prototype plant testing period may need to continue through equilibrium core conditions.”
- Comment
  - Waiting for equilibrium core conditions is not necessary. As the scope of testing is dependent on the sufficiency of data to demonstrate the performance of the intended safety feature(s), the availability of data at an earlier date would obviate the need for a long test period.

NEI

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Peter Hastings
Advanced Reactor Regulatory Task Force

August 3, 2017

REGULATORY ENGAGEMENT PLAN

NRC Advanced Reactors Public Meeting

REGULATORY ENGAGEMENT PLAN (REP)
INTRODUCTION

- Project scope: develop draft of REP guidance for industry and NRC feedback
- Approach
  - Draft annotated outline, review with ARRTF & NRC
  - Coordinate with industry to refine
  - Revise annotated outline (AO) and draft guidance

NEI

REP CONCEPT

- Informed by:
  - Prior licensing project plans
  - Annual new reactor planning Reg Issue Summaries
  - Desire for early identification of technical issues and risks
- Content
  - Menu: potential topics to populate REP
  - Guidance: FAQ-like descriptions of underlying elements of menu items, including options
- Overarching assumptions
  - Optional product
  - Flexible content & format

ANNOTATED OUTLINE
EXAMPLES

A. Introduction to Menu/Guidance Document

- Description of format and content
- Notes on usage
- Relationship to annual planning RIS responses
- Optionality of process and format/content

B. Template Menu Example

Purpose of Applicant REP

- Name of prospective applicant
- Company/project structure
- Strategic project approach/goals
  - Commercial
  - Research
  - Electric production (or not)
  - Development and deployment approach, incl. look-ahead to application type

Only to the extent useful for engagement planning

Guidance in Sec. C

NEI

B. Template Menu (continued)

Other potential menu items could include:

- Background
- Application Type
- Pre-Application Engagement
- Application Process
- Post-Application Engagement
- Partnerships and Industry Participation
- Schedule
- Budget
- References

C. Guidance (example)

- Application Type
  - DC
  - SDA
  - ESP
  - COLA
  - CP
  - OL
  - ML
  - LWA
  - RTR

From "Menu" section

Expand topics as needed, with pointers to other documents

NEI

REP
NEXT STEPS

REP PATH FORWARD

- Initial NRC discussions (today)
- Complete AO development/industry review
- Prepare draft
- Industry review/comment resolution and incorporation
- NRC feedback

QUESTIONS/COMMENTS

Staff Feedback
Licensing Basis Events (LBEs)

Framework

<table>
<tr><th></th><th colspan="4">Analysis Attributes/Documentation</th><th colspan="5">How Analysis Supports Selected Regulatory Areas</th></tr>
<tr><th></th><th>Approach</th><th>Acceptance Criteria</th><th>Analysis QA Standard</th><th>Lic Basis Document</th><th>SSC Safety Class</th><th>Tech Spec</th><th>Procedures</th><th>Used for Siting (<25 rem)</th><th>Environ Reviews (SAMDA)</th></tr>
<tr><td>AOOs</td><td rowspan="5">Risk Assess (PRA)</td><td>SAFDL, SARRDL, Part 20</td><td rowspan="5">PRA Reg Guide (1.200*), 50.69, Realistic mean values</td><td rowspan="5">Chapters 15/19</td><td rowspan="5">Non-Safety systems with enhanced regulatory treatment (e.g. NSRST, RTNSS, 50.69, D-RAP, etc.)</td><td rowspan="4">50.36 criteria 1,2,3, or 4</td><td>Normal, Alarm, AOPs</td><td>No</td><td rowspan="2">No</td></tr>
<tr><td>DBEs</td><td>FPB (eg, SAFDL, SARRDL), F-C (PAG option)</td><td>EOP</td><td rowspan="4">Yes</td></tr>
<tr><td>BDBEs</td><td rowspan="2">Safety Goal (PAG option)</td><td rowspan="2">Guidelines (eg.,SAMGs, FSGs)</td><td rowspan="3">Yes</td></tr>
<tr><td>Functional Containment Basis</td></tr>
<tr><td>EP-basis</td><td>PAGs</td><td>No</td><td>EP</td></tr>
<tr><td>DBAs</td><td>Deterministic</td><td>FPB, 50.34</td><td>Appendix B</td><td>Chapter 15</td><td>Safety Related</td><td>Yes</td><td>EOP</td><td>No</td><td>No</td></tr>
<tr><td>External Events</td><td>Probabilistic Hazards Analysis

Role of “design basis external events (DBEE)”?</td><td>Facility protection requirements</td><td>PRA Reg Guide (1.200*), Realistic mean values</td><td>Chapters 2/3/19</td><td>Safety related for DBEEs?</td><td>Via equipment qualification Or 50.36 criteria 1,2,3, or 4</td><td>OPs, AdmPs, EDMGs</td><td>Yes</td><td>Yes</td></tr>
</table>

Containment - Areas for Consideration and Prioritization (draft)

	Design Criteria associated with containment	Release mitigation limit	Source term	Analysis method	Test Requirement	EP Requirement	Licensing Basis Document	Tech Specs	Safety Classification	Env. Reviews (SAMDA)	Procedures
LWR Containment	GDC - 16	50.34 Part 100	Specified accidents in SRP and prescriptive siting event 50.34(a)(1)(ii)(D) footnote 6	Conservative analysis with only safety-related SSCs credited, including single failure assumption	Appendix J	Prescriptive (current) EP rulemaking (future)	Chapters 6 & 15	50.36 Criteria 1,2,3,4	Safety related	No	EOP
Adv. Non-LWR Functional Containment	ARDC - 16 *	50.34 Part 100	Mechanistic Source Term approach accepted per NRC SECY-16-0012 Establish non-LWR alternative to footnote 6 *	Conservative upper bound analysis Both single and common cause failures are considered during event sequence evaluation *	TBD - since there is no regulation for non-LWRs **	Flexible EP rulemaking (future)	Chapters 6 & 15	50.36 Criteria 1,2,3,4 Assess Criterion 1 re:RCPB **	Safety related	No	EOP

*	Resolve now
**	Resolve later - technology and/or design specific

TRISO Coated Fuel Particle Qualification

Limited Scope Topical Report

NEI – Advanced Reactor Working Group
HTGR Technology Working Group

Table of Contents

HTGR TWG

HTGR
Simple Safe Secure

- Developer Companies

– AREVA	SC-HTGR
– X-Energy	X-100
– Star Core Nuclear	StarCore
– Kairos Power	KP-FHR
– BWX-Technology	TRISO Particle Fuel Supply

- Other Supporters and Observers

– Duke Energy	- DOE
– EPRI	- NEI

HTGR-TWG

HTGR
Simple Safe Secure

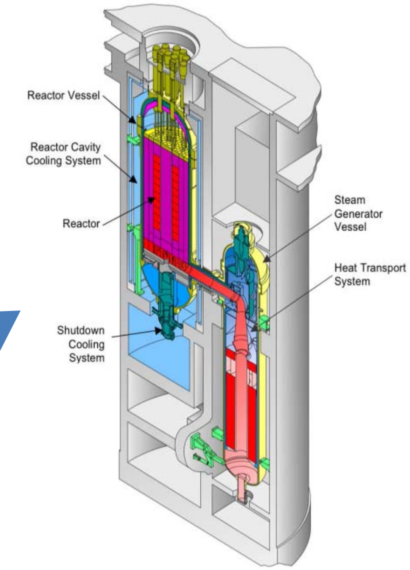
- Mission
  - Ensure that RD&D infrastructure is created, maintained, and available to support the timely development, demonstration and deployment of high temperature gas-cooled reactor technology*.
- Objectives
  - Coordinate with DOE and National Laboratories our R&D needs to ensure that relevant work is aligned with the technology goals of the reactor designers
  - Support the advancement, development and deployment of high temperature gas-cooled reactor technology
  - Establishment of a domestic U.S. fuel supply chain including identification of and resolution to front-end of the HTGR fuel supply needs, industrial supply manufacturing of TRISO coated particle fuel, and storage or recycling of used fuel.
  - Support and coordinate efforts with other organizations and technology specific working groups to achieve shared objectives

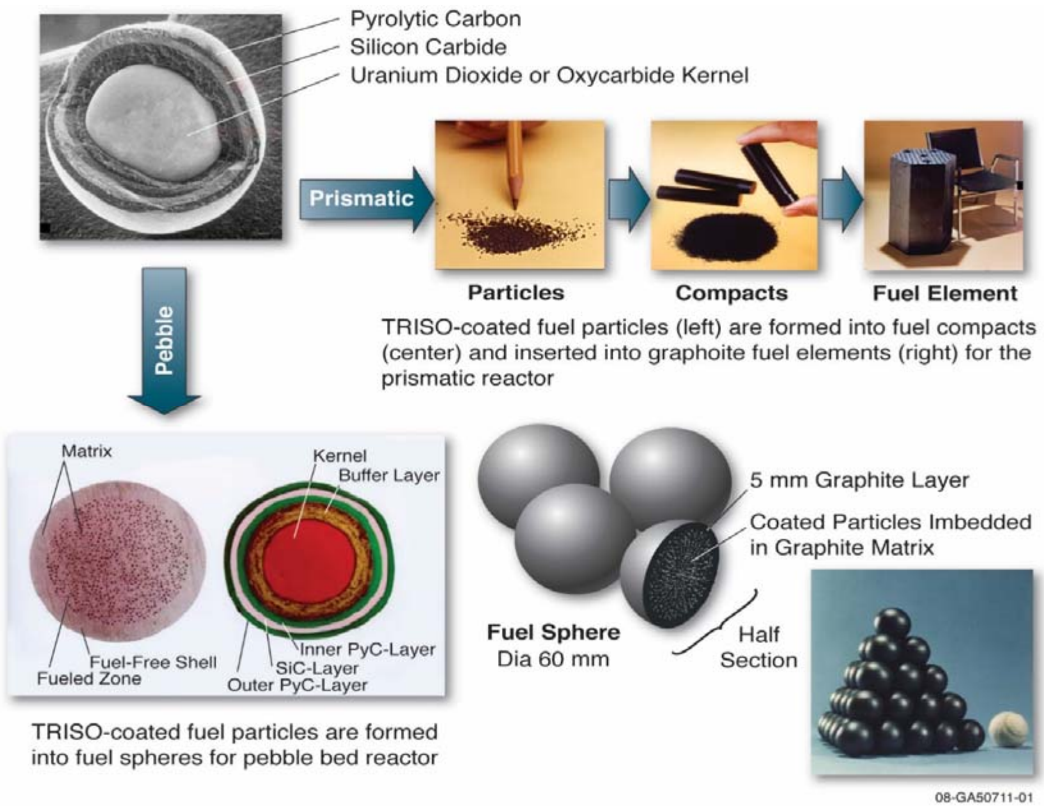
*** Kairos Power – a salt cooled HTR developer recently joined the HTGR-TWG**

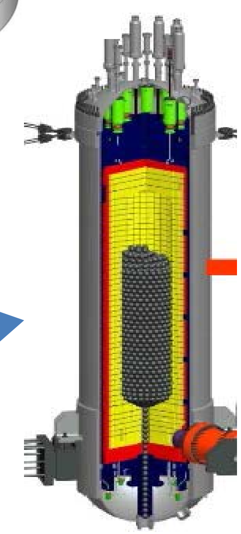
HTGR
Simple Safe Secure

TRISO Coated Particle Fuel

A key Common Element HTGR Technology

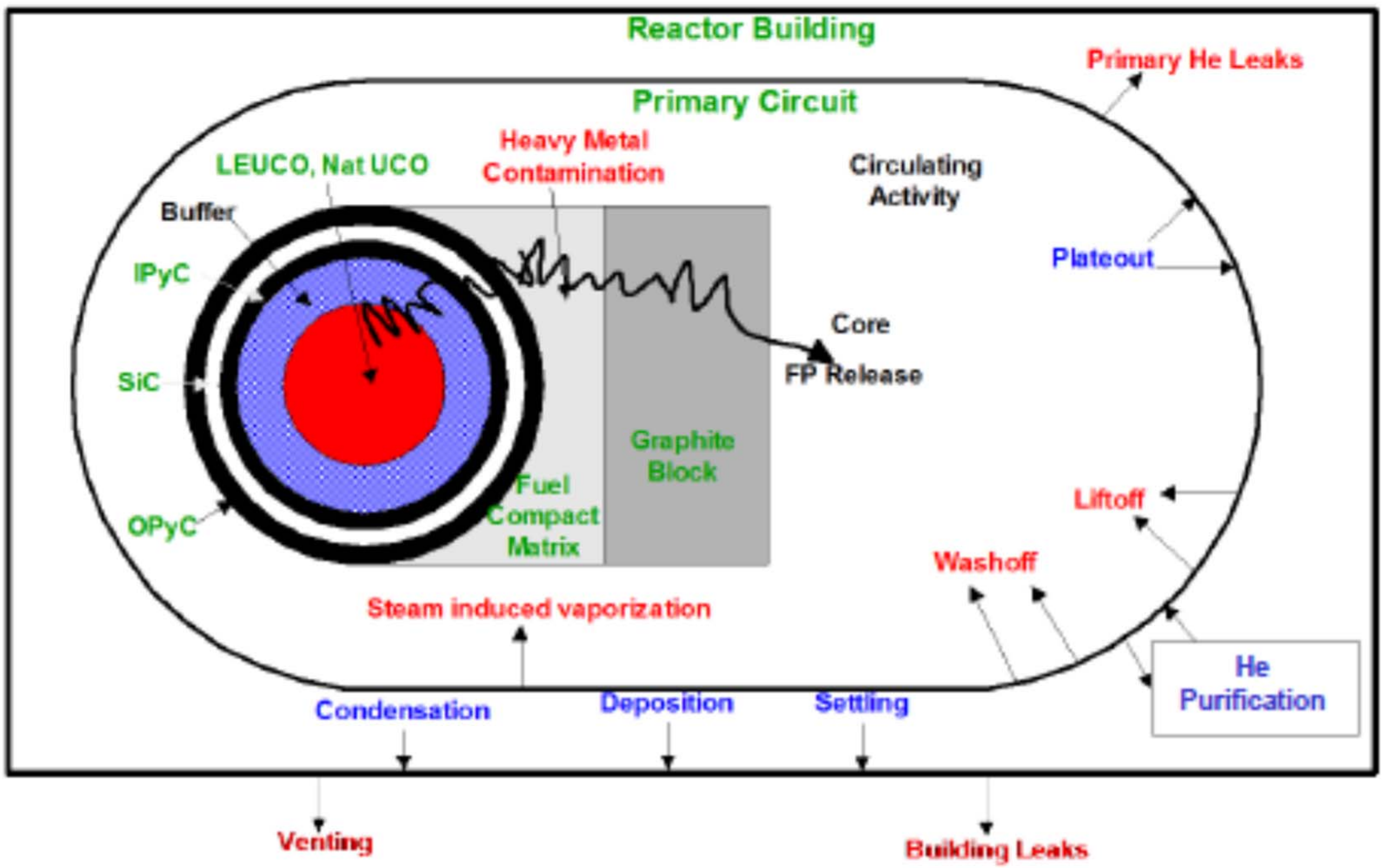






Fission Product Generation, Transport -- Source Term Model --

HTGR
Simple Safe Secure

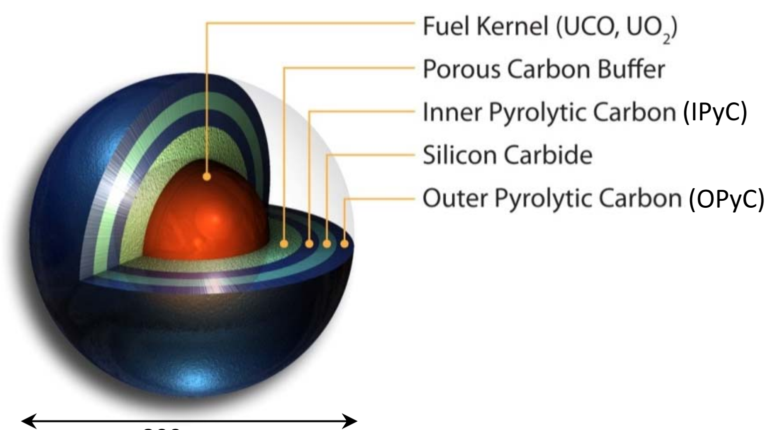


The Advanced Gas Reactor Fuel Development and Qualification Program – Overview and Status

Paul Demkowicz
AGR Program Technical Director
Idaho National Laboratory

Tristructural isotropic (TRISO) Fuel

HTGR
Simple Safe Secure



TRISO particle

- TRISO fuel is at the heart of the safety case for modular high temperature gas-cooled reactors
- Key component of the "functional containment" licensing strategy
  - Radionuclides are retained within multiple barriers, with emphasis on retention at their source in the fuel

High-quality, low-defect fuel fabrication

Robust performance during irradiation and during high-temperature reactor transients

Low fission product release

AGR fuel compact

AGR program

Objectives and motivation

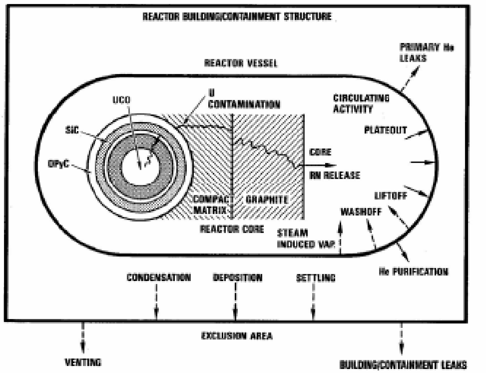
- Provide data for fuel qualification in support of reactor licensing
- Establish a domestic commercial vendor for TRISO fuel

Reduce market entry risk

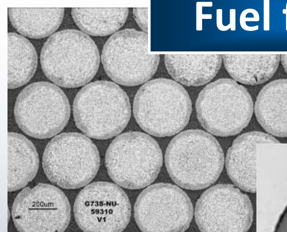
Approach

- Focus is on developing and testing **UCO** TRISO fuel
  - **Develop fuel fabrication and QC measurement methods**, first at lab scale and then at industrial scale
  - **Perform irradiation testing** over a range of conditions (burnup, temperature, fast neutron fluence)
  - **Perform post-irradiation examination and safety testing** to demonstrate and understand performance during irradiation and during accident conditions
  - **Develop fuel performance models** to better predict fuel behavior
  - **Perform fission product transport experiments** to improve understanding and refine models of fission product transport

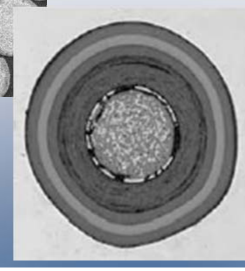
Advanced Gas Reactor Fuel Development and Qualification Program Elements



Fission product transport & source term

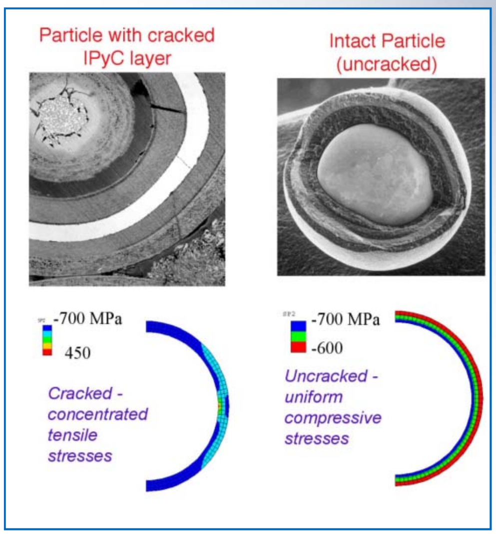


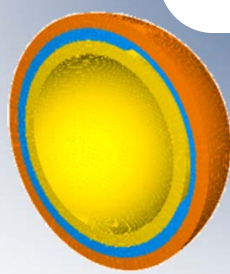
Fuel fabrication



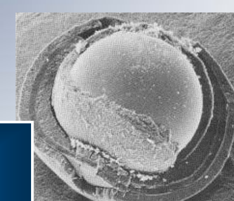
Fuel irradiation

Program participants:
INL, ORNL, BWXT

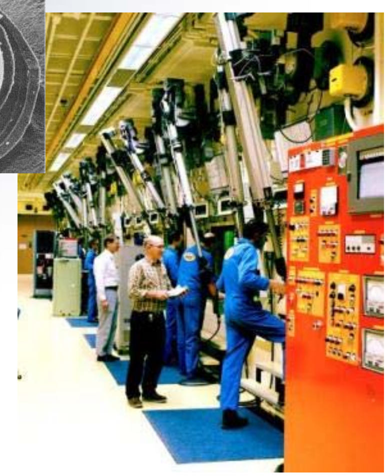




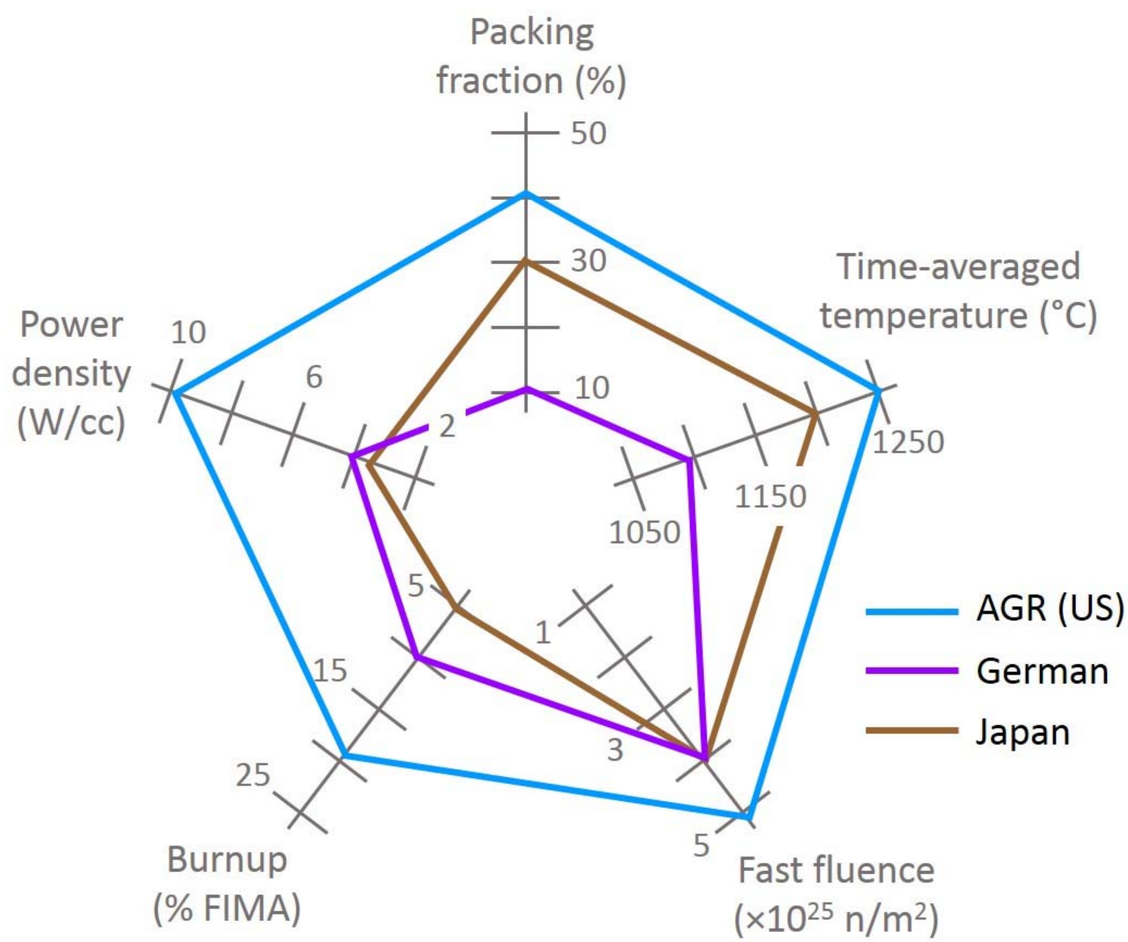
Fuel performance modeling



PIE and safety testing

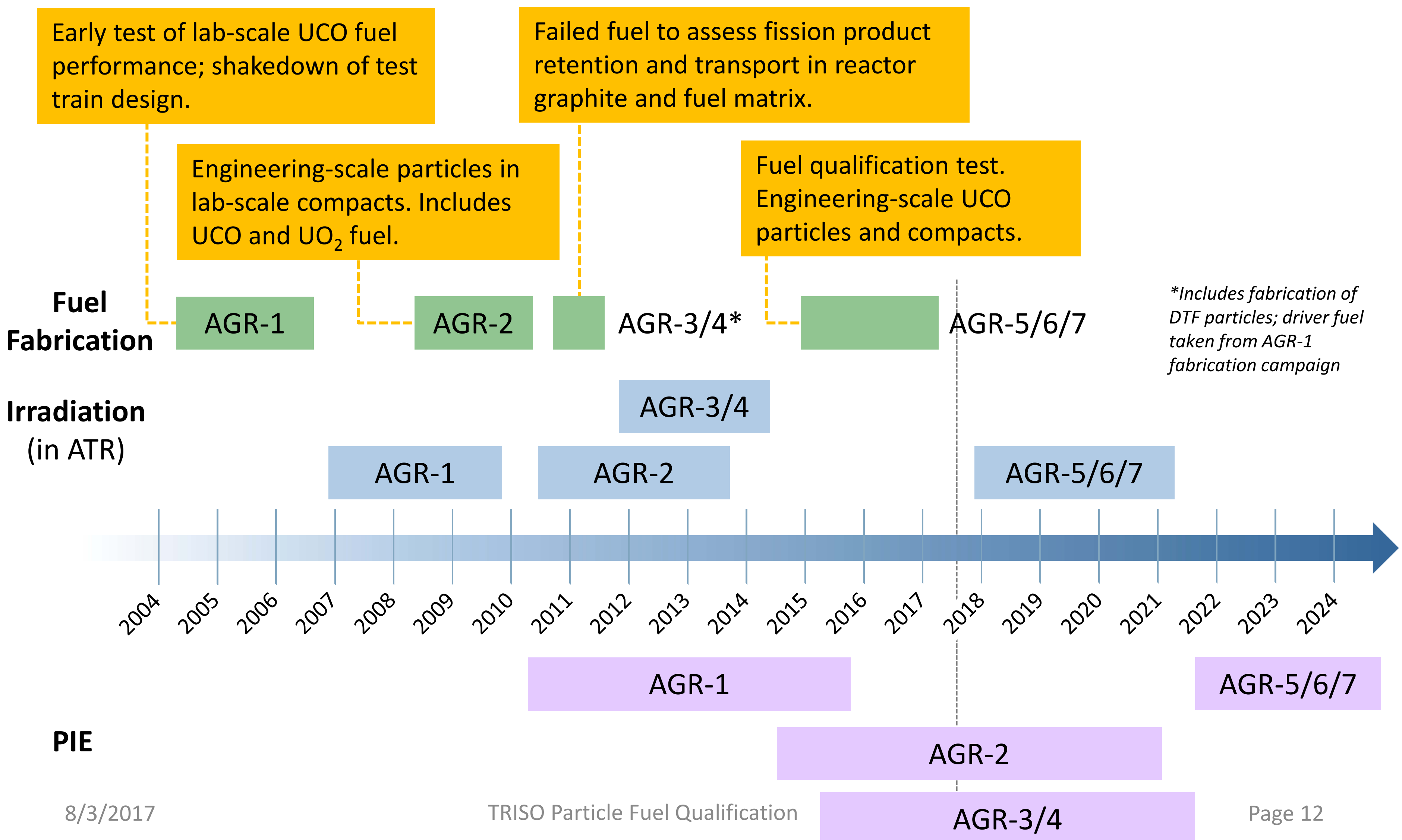


Targeted fuel performance envelope



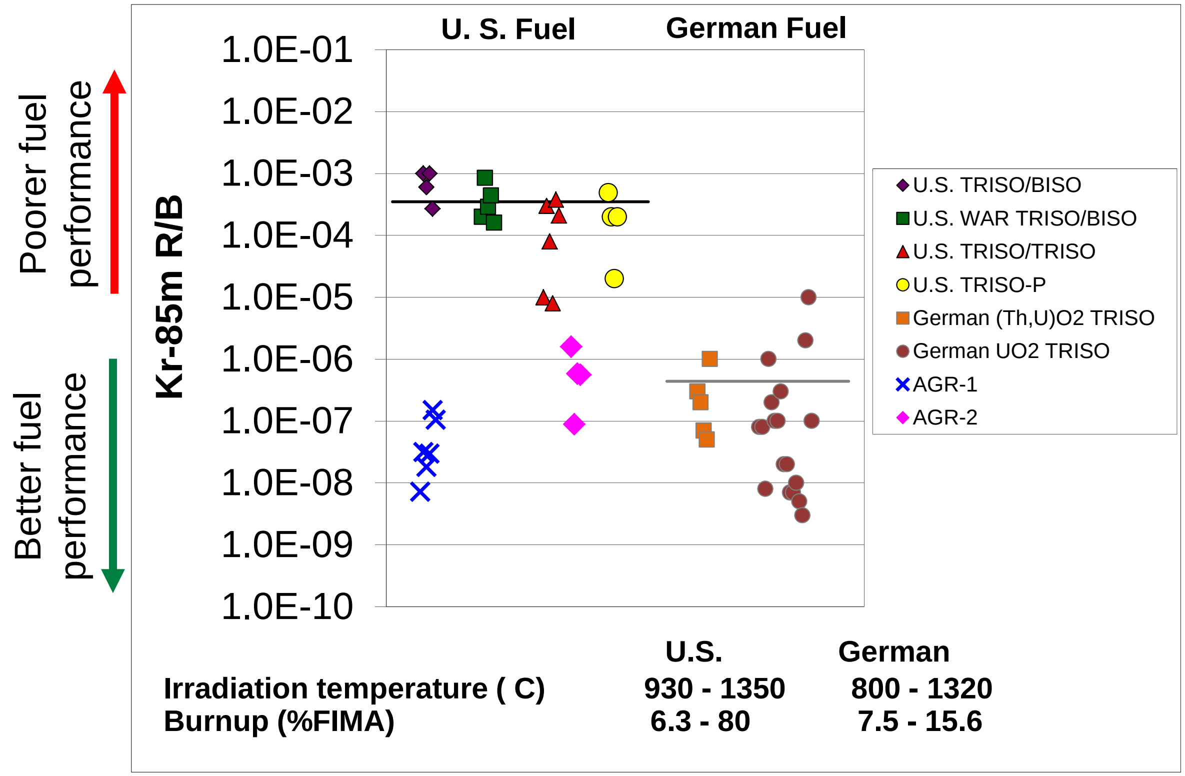
- Program goal is to qualify fuel to a performance envelope that is more aggressive than previous German and Japanese qualification efforts

AGR Program Timeline



AGR Fuel Irradiation Performance

German fuel has historically demonstrated ~1,000 times better performance than U.S. fuel.



Plot of Kr-85m release-to-birth ratio for various fuel types

AGR-1:

- Zero TRISO failures out of ~300,000 particles in the experiment
- Peak burnup ~20% FIMA

AGR-2

- 0 or 1 exposed kernel at beginning of irradiation in each capsule
- Possibility of small number of failures during irradiation

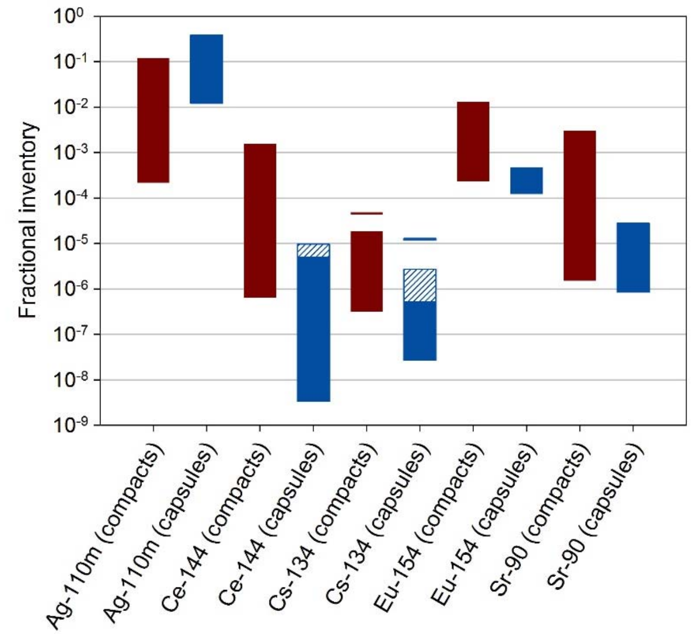
Today, in-reactor AGR TRISO fuel performance is as good as German fuel at twice the burnup

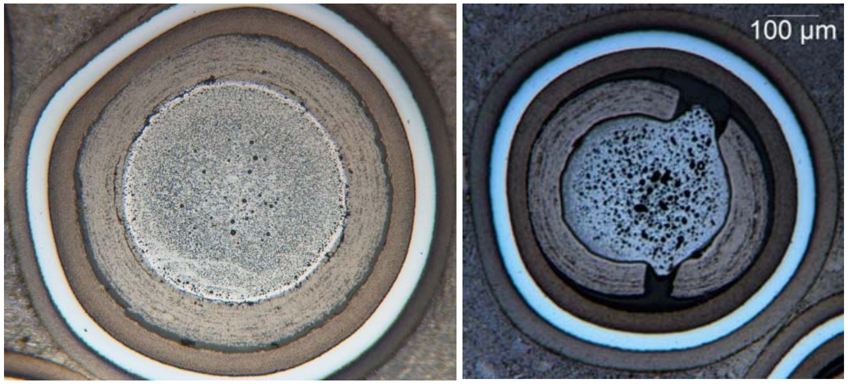
AGR-1 and AGR-2 irradiation performance

HTGR
Simple Safe Secure

- **Low coating failure fractions** (AGR-1 TRISO failure fractions are below existing reactor design specs)
- **Low release of key fission products** (Kr, Cs, Sr)
- **Modest release of Eu; high release of Ag** (influenced by irradiation temperature)
- **Buffer fractures are common** but do not appear to be detrimental to outer coating integrity
- **UCO effective at controlling CO production** which limits gas pressure and kernel migration
- **Significant leaps in understanding causes of coating failures and fission product transport in coatings**

AGR-1 capsule-average fission product release





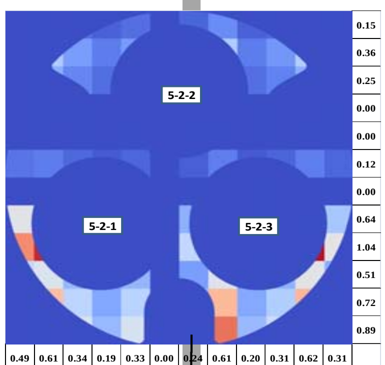
AGR-1 4-1-3 (19.3% avg burnup)

Studying failed particles greatly improves ability to characterize and understand fuel performance

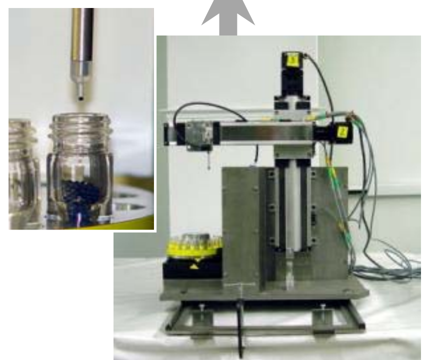
72 fuel compacts containing 300,000 particles in AGR-1 irradiation

AGR-1 Test Train Vertical Section

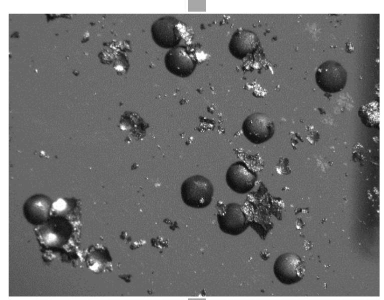
Capsule disassembly



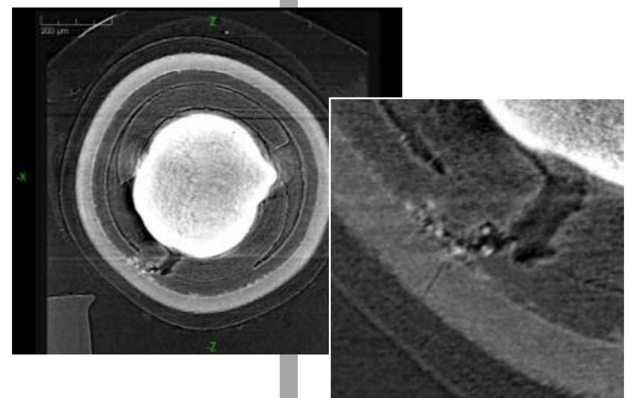
Gamma scan to identify cesium hot spots and compact location



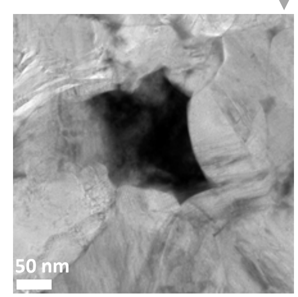
IMGA to find particles with low cesium retention



Deconsolidation to obtain ~4,000 particles from compact



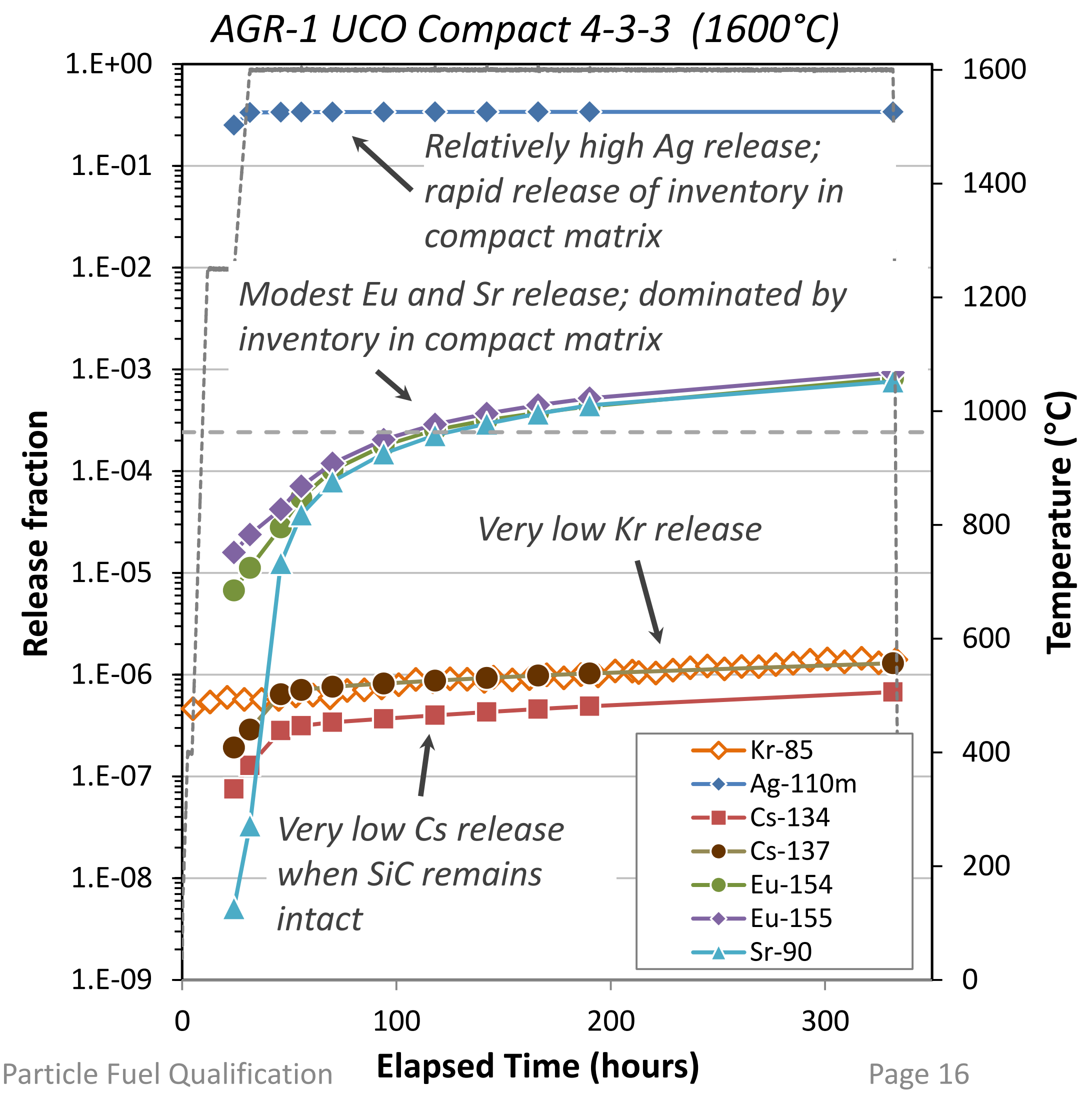
X-ray tomography to nondestructively locate defects/fractures



Advanced microscopy to study microstructure in detail

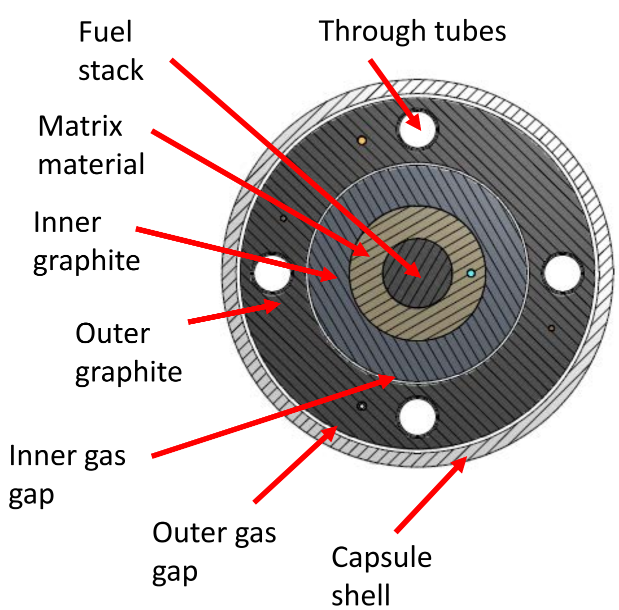
AGR-1 and AGR-2 safety test performance

- **Excellent UCO performance up to 1800°C**
- **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)
- **Low coating failure fractions** (UCO)
- **Accelerated SiC attack by Pd at higher temperatures**
- **UO_2 demonstrates much higher incidence of SiC failure due to CO attack**



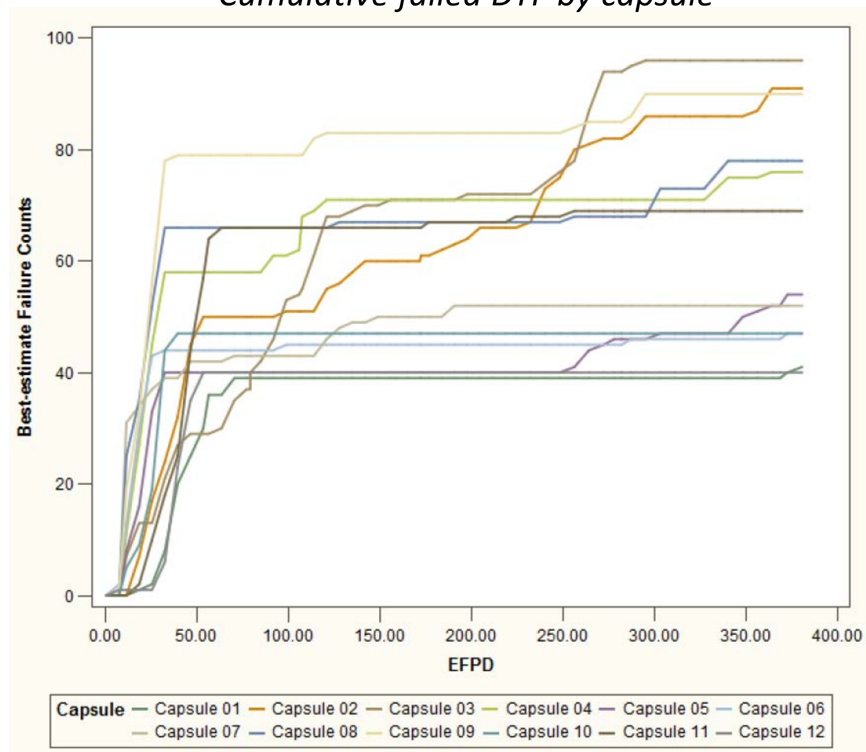
AGR-3/4 Irradiation

- AGR-3/4 irradiation completed April 2014
- Good performance of DTF particles, however:
  - Some difficulty identifying individual DTF failures during irradiation
  - Apparently not all DTF failed



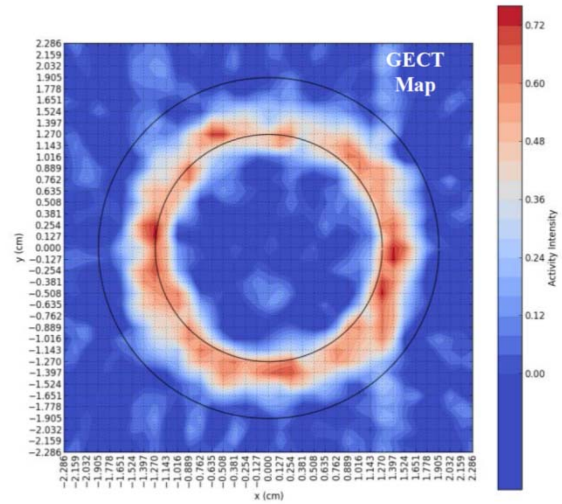
AGR-3/4 Capsule Cross Section

Cumulative failed DTF by capsule

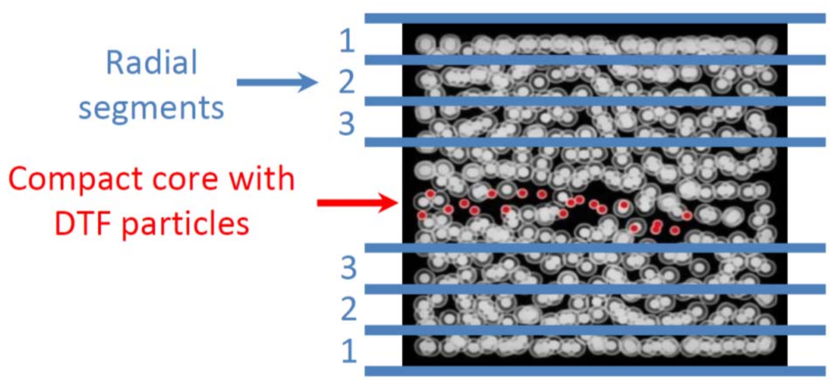


AGR-3/4 post-irradiation examination

- Extensive PIE is in progress
  - Focus is understanding fission product transport in fuel kernels, fuel matrix, and reactor core graphite
- Analyze fission product distribution in rings
- Analyze fission products in compact matrix
- Determine fission product release from fuel at high temperatures in inert and oxidizing atmospheres



Cs-134 gamma emission computed tomogram of an AGR-3/4 inner ring



Schematic of approach for performing radial deconsolidation on AGR-3/4 compact

AGR-5/6/7 irradiation

- Final fuel qualification irradiation; critical link in verifying fuel made at the commercial vendor meets performance requirements
  - Kernels, coated particles, and fuel compacts all made on pilot-scale fuel fabrication line at the commercial vendor
- AGR-5/6: Fuel qualification test
  - Irradiate sufficient number of particles to obtain fuel failure statistics
  - ~530,000 particles in four capsules
  - Temperature and burnup ranges attempt to represent HTGR core-wide distributions (~600 to 1400°C; ~7 to 18% FIMA)
- AGR-7: Fuel performance margin test
  - Explore the threshold for fuel performance
  - ~55,000 particles in a single capsule
  - Upper range of burnup values (~18% FIMA)
  - Time-average peak temperatures up to 1500°C

Conclusions

- AGR program is approximately 2/3 complete
- Key successes to date
  - Excellent overall UCO performance
  - Significant leaps in understanding fuel performance
- Major tasks to completion
  - Complete AGR-2 PIE and safety testing
  - Complete AGR-3/4 PIE
  - Complete AGR-5/6/7 irradiation, PIE, and safety testing
  - Perform key safety tests in oxidizing atmospheres
  - Support NRC interactions on licensing
  - Code comparisons to data
  - Program closeout and reporting
- Several domestic vendors are depending on AGR program completion to establish domestic vendor and qualify fuel and decrease market entry risk

Limited Scope Topical Report

- Purpose
- Approach to TRISO Fuel Qualification
- LSTR Preparation Plan
- Provisional Schedule

Limited Scope Topical Report Purpose

HTGR
Simple Safe Secure

- A "generic" Topical Report covering key safety aspects of UCO based TRISO coated fuel particle
- To be referenced by multiple license applicants as a part of their overall fuel qualification program and reactor licensing approach
- Will be developed by a team of reactor developers, industry representatives and DOE/INL
- To reduce regulatory risks for applicants intending to use UCO based TRISO particles

Regulatory Risk Reduction Opportunity

- TRISO-coated particle fuel performance is fundamental to reactor safety and a major source of licensing uncertainty
- TRISO fuel safety qualification success is critical for:
  - Prismatic block and pebble bed modular HTGRs
    - AREVA, X-energy, StarCore
  - Certain types of molten salt reactors (FHRs)
    - Kairos Power
- FQ is a long-lead activity without well-defined regulatory criteria against which success can be benchmarked and assured
- TRISO fuel particle development, design, and manufacture is complete and is not expected to change
- While applicants are ultimately responsible for qualifying fuel used in their design, assistance in developing a foundational portion of the TRISO particle fuel qualification basis is available from:
    - INL/Advanced Gas Reactor Program
    - BWXT
    - EPRI

Approach to Fuel Qualification

HTGR
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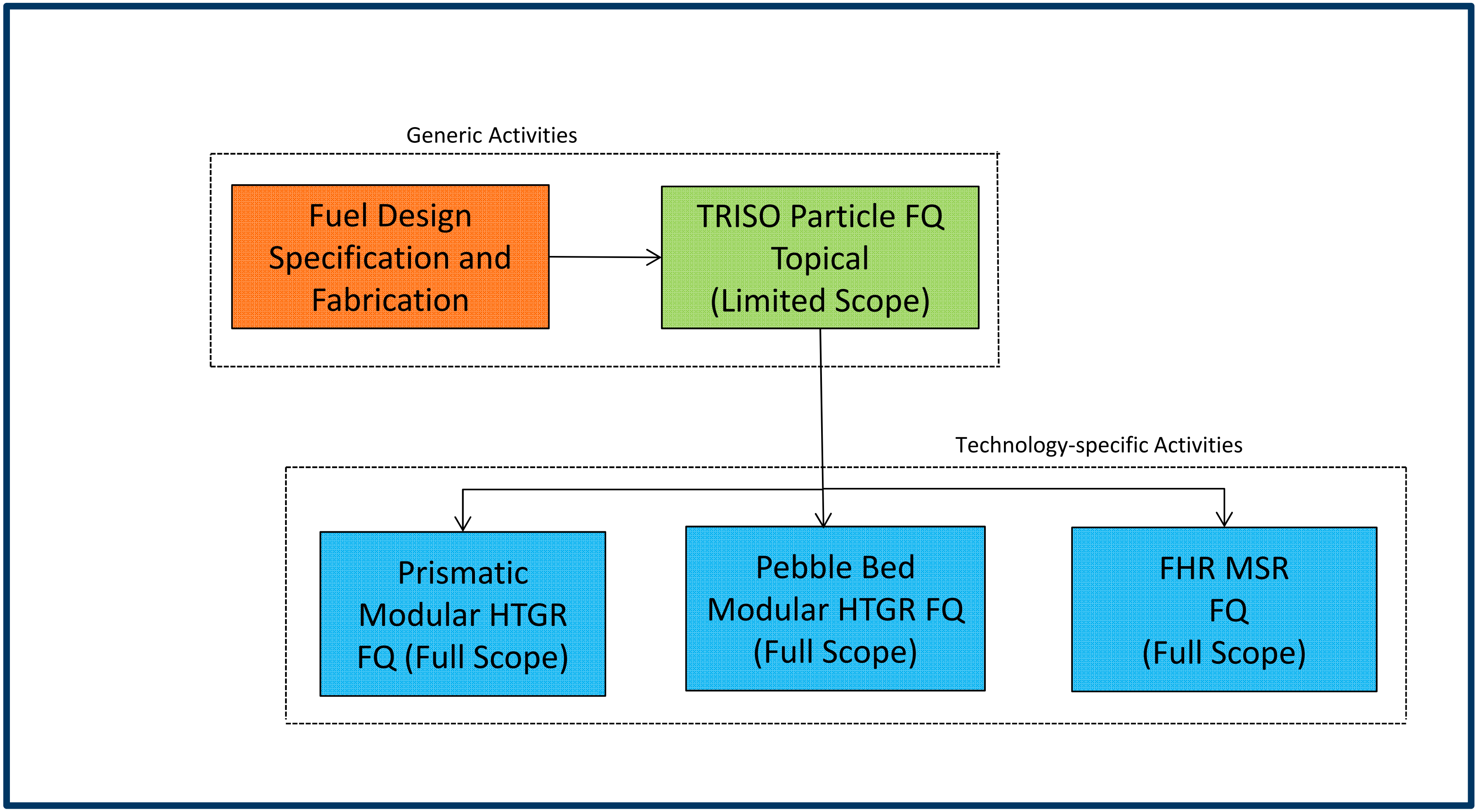
Time-phased NRC submissions enable a staged analysis of elements required to support uranium Oxycarbide (UCO) fuel qualification

- Generic - Limited Scope Topical would examine:
  1. TRISO UCO fuel design characteristics and rationales
  2. TRISO UCO fuel product specifications
  3. Description of fuel fabrication process
  4. Statistical QA methods that assure specifications are met
- Full Scope Topical supplements the Limited Scope by later addressing:
  5. Irradiation behavior of fuel (in-pile performance and PIE)
  6. Fuel safety test results
  7. Establish TRISO UCO fuel performance envelope with failure rates for normal and off-normal conditions

TRISO Fuel Qualification



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Preparation Plan

- **Establish a team to prepare the LSTR**
  - Reactor developers
  - BWXT – TRISO manufacturer
  - EPRI – Project Manager
  - INL – Experimental and irradiation data from AGR program
- **LSTR preparation**
  - Annotated outline
  - Periodic meetings for Staff interactions, familiarization, update on progress of the LSTR, and emergent issues discussion and resolutions prior to LSTR submittal
  - Preparation and review of the LSTR
- **Submit LSTR for NRC Safety Review**
  - NRC acceptance review
  - NRC Review (RAIs)
  - Closure of all RAIs
  - ACRS review (if necessary)
- **Final Safety Evaluation Report for the LSTR**

HTGR
Simple Safe Secure

Provisional Schedule

TRISO Fuel Topical Preparation and Review	Year-1				Year-2				Year-3			
	Q1	Q2	Q3	Q4	Q5	Q6	Q7	Q8	Q9	Q10	Q11	Q12
Topical Report Preparation (TWG – EPRI - INL)												
Topical Report Submital (EPRI)			Δ									
NRC Reviews (RAIs)												
RAI Review and Resolutions				↓	↓↑	↑↓	↓↑	↑↓	↓↑			
Final Topical Report												
Quarterly Status Meetings	◊	◊	◊	◊	◊	◊	◊	◊	◊	◊	◊	◊
Safety Evaluation Report												Δ

- **TRISO Coated Fuel Particle <u>Limited Scope Topical Report</u>**
  - **Prepare content (TWG- EPRI - INL)**
  - **LSTR submittal (EPRI)**
  - **NRC Review (off-fee)**
  - **Quarterly status meeting (HTGR-TWG, INL, EPRI and NRC)**
  - **ACRS review (if necessary)**
  - **Safety evaluation report (NRC)**

Q&A

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Metallic Fuels Irradiation Database and Data Qualification

NRC Public Meeting with Nuclear Industry Groups and other Stakeholders on Possible Regulatory Process Improvements for Advanced Reactor Designs

August 3, 2017

A. M. Yacout

Argonne National Laboratory

Presentation Overview

- Presentation Purpose
- Metallic Fuel Experience
- Metallic Fuel Data Qualification
- Summary

Presentation Purpose

- Provide overview of existing metallic fuel database relevant to licensing of fast reactor designs
- Provide a summary of ongoing effort to evaluate the data and making it available to industry and other stakeholders

Metallic Fuel Experience

Metallic Fuel Historical Data

- **Over 30 years of irradiation experience in: EBR-I, Fermi-1, EBR-II, FFTF**

- **Different types of metallic fuel:**
  - U-Fs*, U-Mo, U-Pu-Fs*, U-Zr, U-Pu-Zr, others

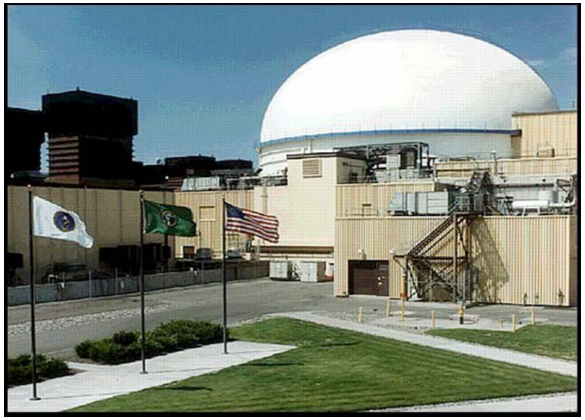
- **EBR-II**
  - > 40,000 U-Fs* pins, **> 16,000 U-Zr pins** & > 600 U-Pu-Zr pins irradiated
  - Clad in 316 stainless steel, D9 & HT9

- **FFTF**
  - **> 1000 U-Zr pins**, mostly in HT9
  - Broad experience with HT9 cladding

*Fs – Simulated Fission Products

EBR-II

FFTF



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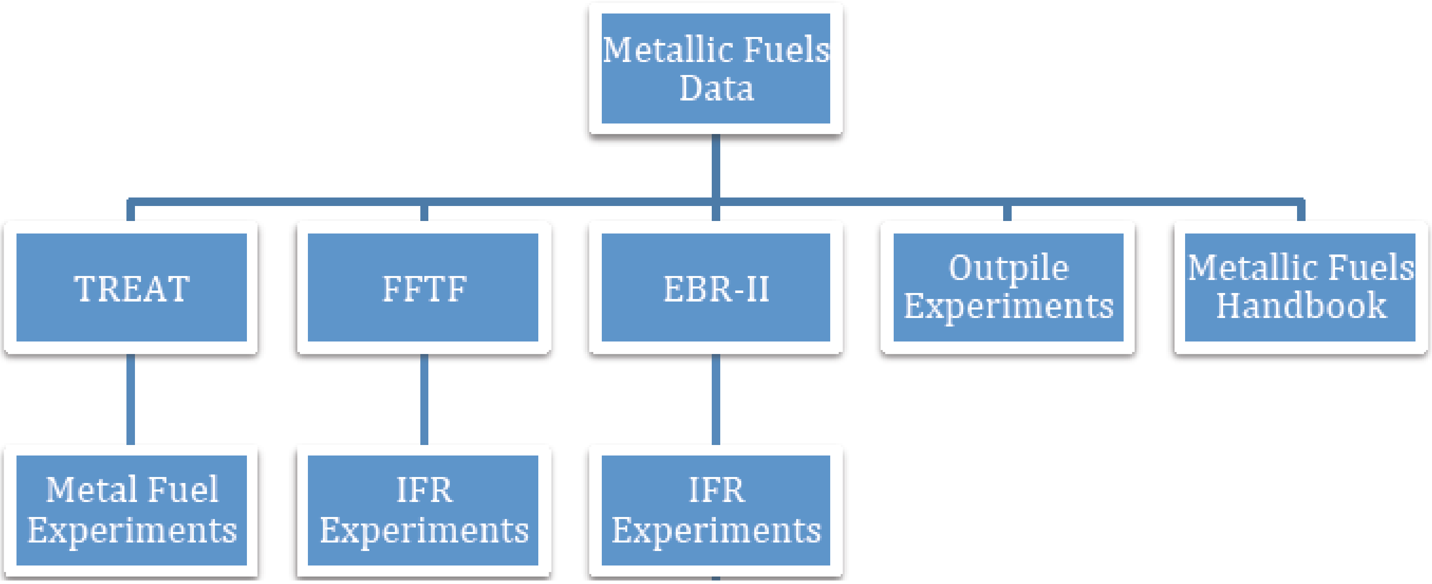
Metallic Fuel Data Sources Relevant to Advanced Designs



Metallic Fuel Experiments

- **EBR-II experiments to look at parameters and phenomena of interest to fuel performance**
  - Prototype fuel behavior
  - RBCB* and failure mode
  - Fuel swelling and restructuring
  - Lead IFR** fuel test
  - Fabrication
  - Design parameters
  - High clad temperature
  - Large fuel diameter
  - Blanket safety
  - Fuel qualification
  - Fuel impurities
- **FFTF experiments to look at**
  - Fuel column length effects
  - Lead metal fuel tests
  - Metal fuel prototype
  - Metal fuel qualification

- **In-Pile (transient)**
  - Run Beyond Cladding Breach (RBCB) experiments:
    6 RBCB tests U-Fs &
    U-Pu-Zr/U-Zr
  - 6 **TREAT** tests:
    U-Fs in 316SS&
    U-Zr/U-Pu-Zr in D9/HT9
- **Out-Pile (transient)**
  - Whole Pin Furnace Tests (WPF)
  - Fuel Behavior Test Apparatus (FBTA)
  - Diffusion compatibility tests

*RBCB – Run Beyond Cladding Breach
**IFR – Integral Fast Reactor

Metallic Fuel Databases

- DOE-NE Advanced Reactor Technology Program (ART) support the development of metallic fuel databases for both steady state and transient fuel behavior
  - **EBR-II**
    - FIPD (Fuels Irradiation and Physics Database)
  - **FFTF**
    - PST (Passive Safety Test) Database
    - Metallic Fuel Experiments Database & Plant data
  - **TREAT**
    - TREXR (TREAT experiments relational) Database
  - **Out-Pile Experiments**
    - FBTA (Fuel Behavior Test Apparatus)
    - WPF (Whole Pin Furnace)

EBR-II Fuels Irradiation and Physics Database (FIPD)

Motivation

- Integral Fast Reactor (IFR): knowledge base for U-Zr metallic alloy fuel
- PIE reports and data, drawings, experiments QA reports, and other documents
- Wealth of data needed for design and code validation efforts

Objective

- Create an online database that archive all information from EBR-II fuels irradiation experiments as well as calculations-based information

End Use

- Industry and institutions with interest in developing metallic fuel based fast reactors
- Validation and calibration of metallic fuel performance codes

EBR-II Fuels Irradiation and Physics Database (FIPD)

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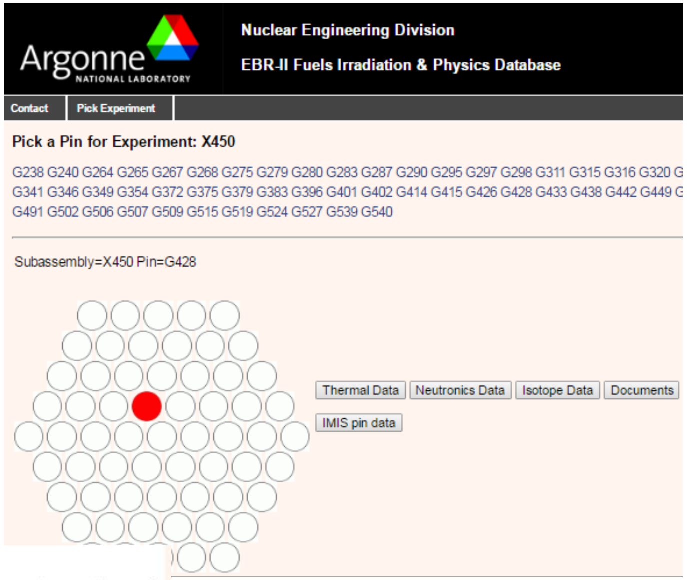
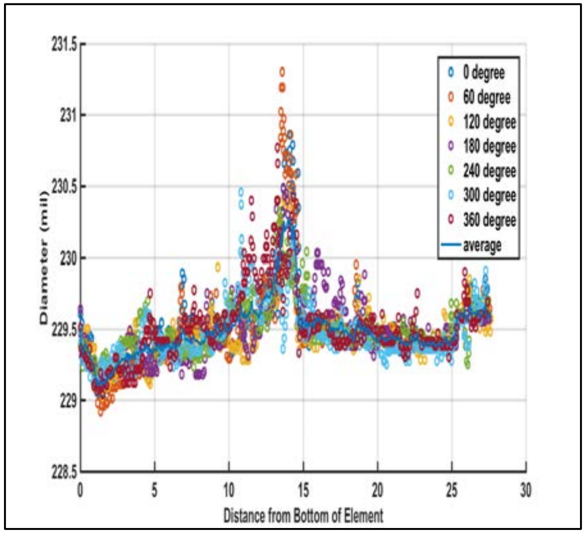
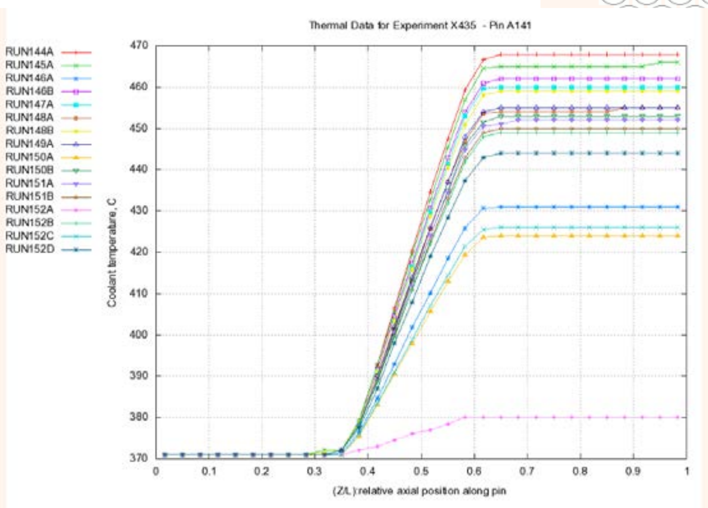
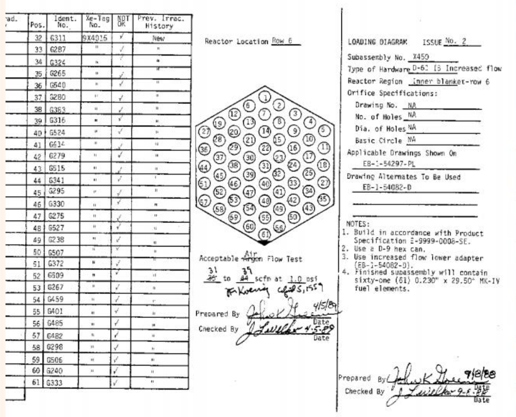
- **Data and information for 24 EBR-II IFR experiments were compiled and archived in database**
  - Raw experimental data and processed data
  - Original documents and memos
  - Simulated detailed operating conditions
- **Access to database through a user friendly Web-interface**









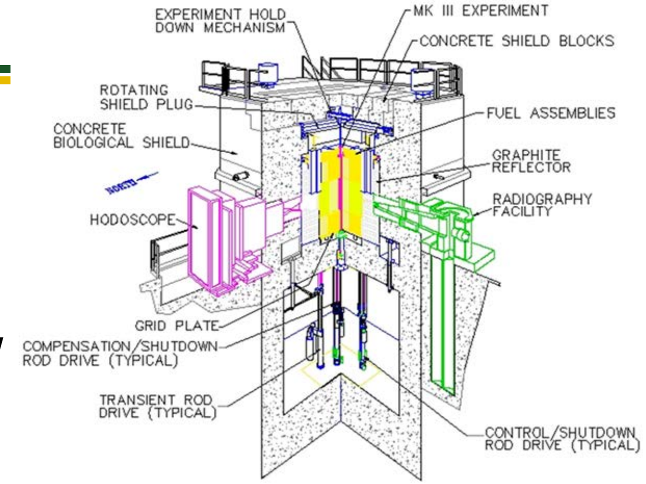
FFTF Database

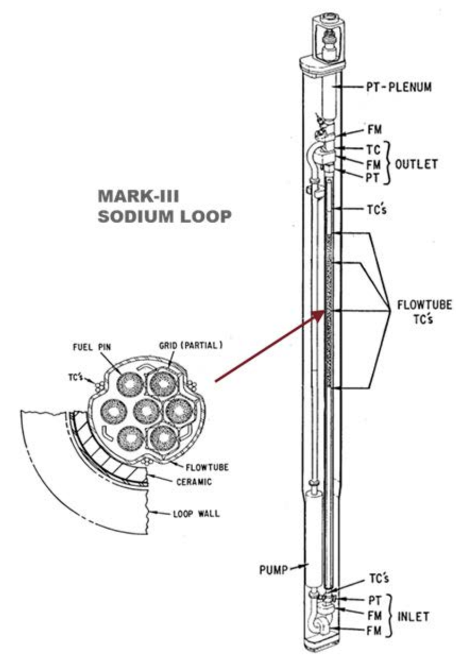
- **Aggressive irradiation testing of 8 metallic fuel assemblies containing long fuel pins (similar to full-scale LMRs) was successfully conducted in the FFTF, with no cladding breaches observed up to burnups approaching 150 MWd/kgM**

- **PNNL compiled existing information on FFTF metallic fuel irradiation tests**
  - Plant data (powers, flows, temperatures)
  - Test reports
    - Test design descriptions
    - QA fabrication records
    - Irradiation reports
    - PIE reports

TREAT Experiments Relational Database (TREXR)

- **TREXR content:**
  - Searchable collection of info on reactor transient tests conducted in TREAT (1959-1994)
  - ~**6000 text-searchable PDFs** in digital archive w/ content descriptions (e.g. tests referenced, type of document)
  - ~**900 tests** & categories w/ parametric information (e.g. fuel, transient info, results)
- **Recent milestones:**
  - **Demonstration of the TREXR web application** at the Fuel Safety Workshop – May 2017
- **Experimental records & reports included in TREXR can be used to support qualification of fast reactor fuels.**





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TREXR supports qualification of metallic fuels
Example: M-series tests

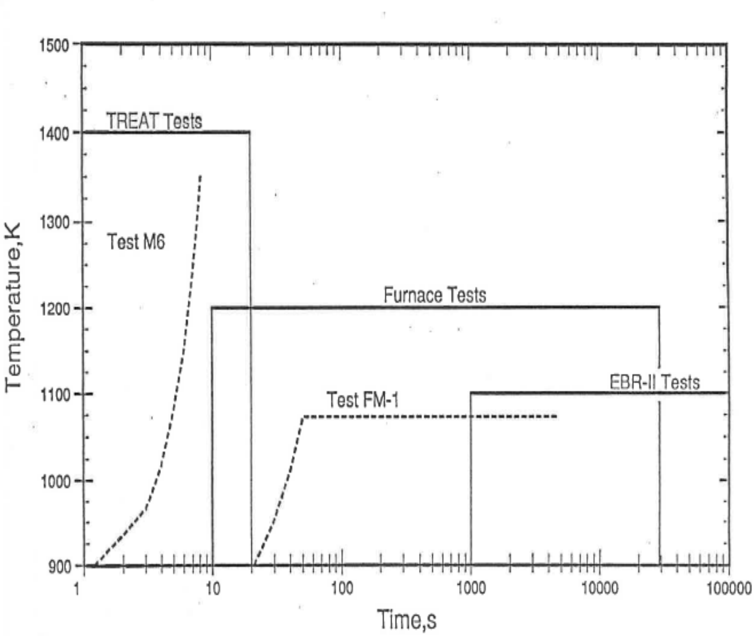
- Metallic Fuel Transient Overpower Tests
- Loop w/ recirculating liquid sodium coolant
- U-5Fs, U-10Zr, U-19Pu-10Zr w/ D9, HT9 clad
- Experiment specifications, test plans, digital data, and technical reports available to users.

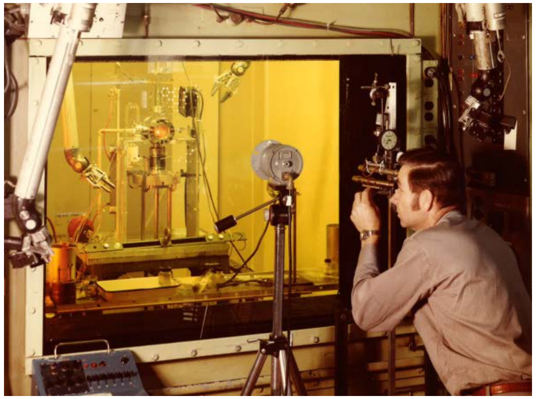
test	date	transient #	digital data
M1	05/10/1985	2608	Y
M2CAL	10/02/1984	2576-2583	
M2	03/21/1985	2596	Y
M3	04/11/1985	2599	Y
M4	01/13/1986	2683	Y
M5	08/15/1986	2714	Y
M6	02/06/1987	2734	Y
M7CAL	05/1987		
M7	10/15/1987	2775	Y
M8CAL	10/1990	2874	Y

	objectives or outcomes							fuel		B/U			clad		transient type		transient meas.		post-test analyses						posttest examinations & measurements			
	Limited fuel damage, no clad breach	Pre-failure in-pin fuel motion	Cladding failure threshold	Mild cladding failure	Fission product release	Fuel-coolant interaction	Post-failure fuel and cladding disruption & dispersal	U-Zr	U-Pu-Zr	Zero (fresh fuel)	Low (up to 5 at.%)	Medium (5 to 10 at.%)	D9	HT9	Overpower	Heat balance	Fast Neutron Hodoscope	Thermal-hüdraulic (Press, Temp, Flow)	Thermal-hydraulics	Cladding failure	Fuel motion	Severe-accident behavior, interaction of multiple Phenomena	Microstructural and mm-scale changes	Fuel-cladding interface or gap behavior	Examinations of irradiated sibling samples	Non-destructive examination of test remains	Destructive examination of test remains	Micro-examination of fuel & cladding remains
M5	✓	✓							✓		✓		✓		✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓
M6	✓	✓	✓	✓		✓	✓		✓		✓	✓	✓		✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓
M7	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓

Out of Pile Experiments

- **Transient Fuel Failure Test**
  - Hot cell furnace testing of pin segments & full elements
    - Fuel Pin Test Apparatus (FBTA)
    - Full length pins, Whole Pin Furnace (WPF)
    - Showed significant safety margin for particular transient conditions.
- **U-xPu-yZr (x=0-26, y=10) / D9, HT9, 316SS**
  - Fuel compatibility tests on clad fuel segments
  - Fission gas retention examinations
  - Penetration depth data were measured and provided the basis for penetration depth correlations
- **Furnace tests simulated reactor accidents, varying ramp rates, conditions, peak temps.**
  - ~1hr-1day, T=600 to 900°C & to melt
  - B/U: 2-3 a/o (WPF), 6 and 12 a/o (FBTA)
- **Results support fuel qualification:**
  - a) metallurgical examination of the tested materials
  - b) fission product release measurements





Metallic Fuel Data Qualification

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Experimental Fuel Design Parameters

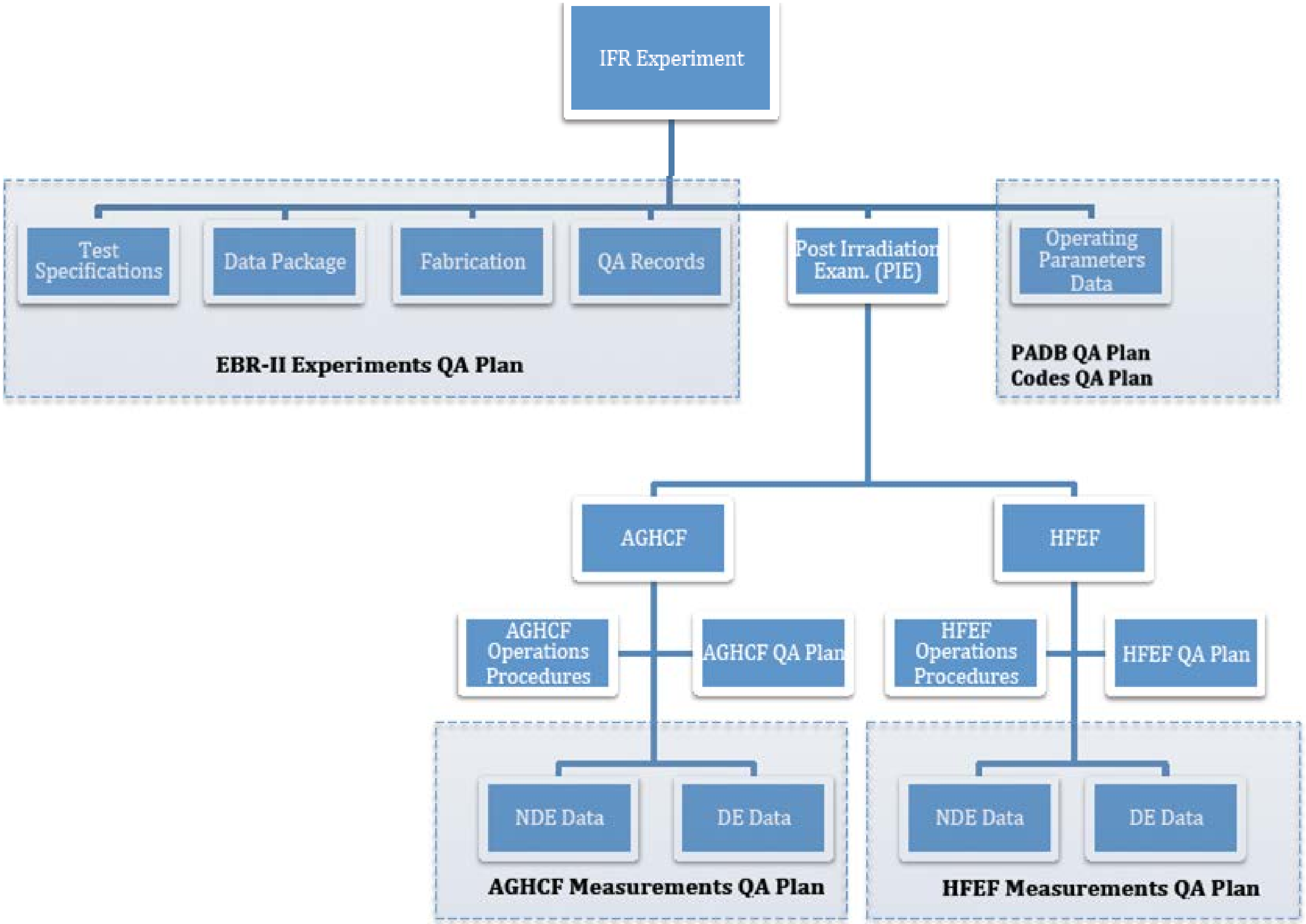
Key Parameter	**EBR-II/FFTF**
Peak Burnup, 10^4MWd/t	5.0 – 20
Max. linear power, kW/m	33 – 50
Cladding hotspot temp., ^oC	650
Peak center line temp., ^oC	<700
Peak radial fuel temp. difference, ^oC	100 - 250
Cladding fast fluence, n/cm^2	up to 4 x 10^{23}
Cladding outer diameter, mm	4.4 - 6.9
Cladding thickness, mm	0.38 – 0.56
Fuel slug diameter, mm	3.33 – 4.98
Fuel length, m	0.3 (0.9 in FFTF)
Plenum/fuel volume ratio	0.84 to 1.45
Fuel residence time, years	1 - 3
Smeared density, %	75*

*Limited experience with higher and lower smeared density fuel pins

SFR Metallic Fuel Data Qualification

- **DOE-NE ART Program supported (Licensing Technical Area): Pre-licensing evaluation of legacy SFR metallic fuel data**
- **The metallic fuel database provides fuel performance data required for demonstrating safe operations within bounding operating conditions**
- **The database provide information necessary for establishing fuel design criteria and failure thresholds**
- **Ongoing work to establish the qualification of the legacy data from metallic fuel irradiation experiments for use in SFR regulatory related activities above**
    - Also, support identifying data gap areas for establishing future programs to fill the gaps

EBR-II Qualification Data Sets



Data Qualification Plan

- **Overall plan follows ASME NQA-1 for Data and Software QA**
  - Data and Software QA Plan commits to NQA-1 2008/2009a:
    - NQA-1 Part III, Subpart 3.3, Nonmandatory Appendix 3.1 Guidance on Qualification of Existing Data
    - NQA-1 Part II, Subpart 2.7, Quality Assurance Requirements for Computer Software for Nuclear Facility Applications
- **Qualification methods include:**
  - QA program equivalency, data corroboration, confirmatory testing, & peer review
    - Appendix 3.1 guidance was applied which require one or more of those criteria to be met
- **Reports:**
  - ANL-ART-76 “Pre-Licensing Evaluation of Legacy SFR Metallic Fuel Data”
  - ANL/NE-16/17, Rev. 0: “Quality Assurance Program Plan for SFR Metallic Fuel Data Qualification” (QAPP)

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Example Data Evaluations

- **Process and Procedure for Historical SFR Metallic Fuel Data Qualification Procedure NE-NSA-PROC-1 is followed:**
  - All information relevant to the specific measured data are identified including reports, data books, drawings, instrument calibration data, memos, ..
  - Relevant hot cell operations procedures and QA program plans existing at the time of measurements are identified and used in the evaluation process (e.g., Operations Manual and Measurements QA Plans for Alpha Gamm Hot Cell Facility - AGHCF)
  - Both QA equivalency and peer review methods are used to evaluate the data based on all collected information with participation of subject matter experts
- **Procedure used to evaluate the following data from specific experiments:**
  - Metallographic Examination Data for HT9 Cladding FCCI (fuel-cladding chemical interaction)
  - Fuel Diameter Measurements
  - Cladding Density Measurements
  - Low-Burnup Fuel Density Measurements
- **Peer review method was satisfied by technical evaluation of the employed methodology, data acquisition and development, test plans, interpretations, and potential uncertainties in the results**

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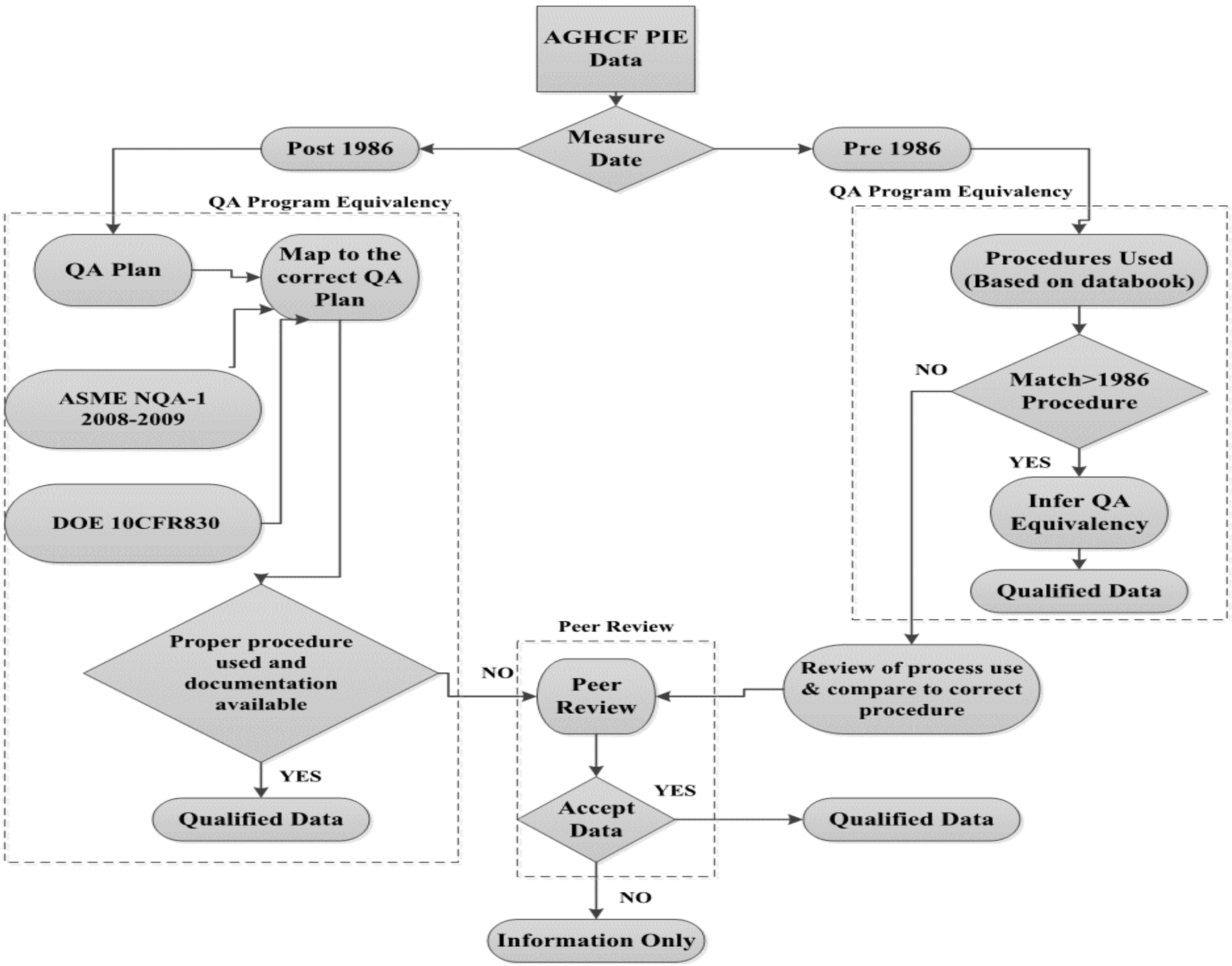
EXAMPLE: Evaluation Process for PIE Data Generated at AGHCF

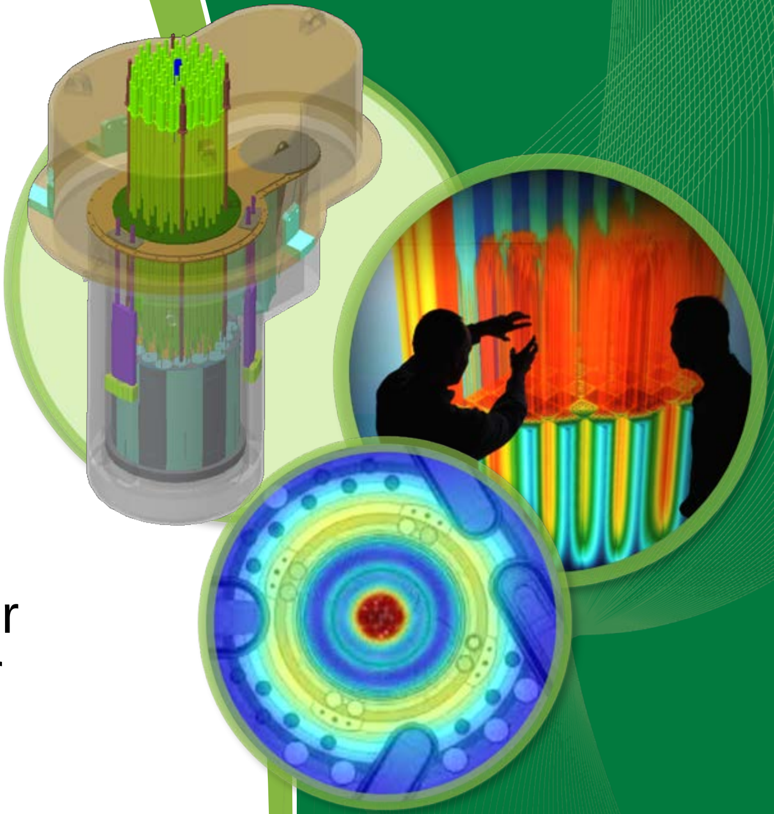


Summary

- **Over 30 years in-reactor experience with metallic fuel irradiation.**
- **Extended databases of metallic fuel behavior is available that cover wide range of operating and design parameters.**
- **Databases include steady state and transient fuel behavior data necessary for regulatory licensing evaluation of reactor design that utilize metallic fuels.**
- **DOE-NE ART program supports the databases development and supports effort to evaluate and qualify the data for use by industry stakeholders as they interact with licensing authorities**
- **A data and software quality assurance program has been established to evaluate the data and qualify the analytical and database software to make it available to stakeholders.**

Fuel Qualification for MSRs

George Flanagan, Ph.D.
Advanced Reactor Systems and Safety
Reactor and Nuclear Systems Division



Presentation for:

NRC/Nuclear Industry Group/Stakeholder Meeting on Regulatory Improvements for Advanced Reactor Designs

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OAK RIDGE National Laboratory

Fuel Qualification Needs to be Defined for MSRs

- Fuel Qualification – Traditional
  - For heterogeneous reactors the fuel/cladding system is the principal barrier to release of fission products
  - Extensive effort has been placed by the industry and NRC on assuring that the behavior of the fuel is well understood under all perceived operational conditions (including AOOs and postulated accidents) ⟺ fuel qualification
    - Heterogeneous fuel performance is substantially impacted by radiation and temperature history
    - Liquid fuel has no history effects (accumulated stress, creep, swelling, etc.) beyond changes to chemical composition
  - Includes extensive irradiation and hot cell examinations

Fuel Qualification Needs to be Defined for MSRs (cont.)

- Fuel Qualification – MSRs
  - Have no equivalent to the traditional fuel qualification process
  - MSRE indicates that fluoride salt compounds are insensitive to irradiation damage
  - Chloride salts still need irradiation data
  - Major concern will be changing chemical behavior during residence time in the reactor and in storage
  - In MSRs the fuel is also the coolant
    - Changes in chemical composition may change the thermal/hydraulic properties of the salt and thus impact the safety case
  - **Issue:** there are no regulatory precedents for identifying the controlling parameters that need to be addressed in the MSR fuel qualification

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Fuel Qualification Needs to be Defined for MSRs (cont.)

- Fuel Qualification – MSRs
  - Properties are generally known for pure fuel salts at beginning of life
  - Properties are not well known for salts containing corrosion products, fission products and minor actinides as a result of irradiation (outside of MSRE)
  - Need fuel performance modeling with data benchmarks (may not require irradiated materials)
  - NRC must be involved to assure that information generated is adequate and complete
  - Appropriate quality assurance must be applied

Fuel Qualification Needs to be Defined for MSRs (cont.)

- Fuel Qualification – MSRs
  - Need for basic information to be generated to assure that all parameters associated with fuel salts that can affect safety or operations are understood (impurity limitations/cliff edge effects) – Simultaneous Fuel Performance Specification
    - Radionuclide retention (source term)
    - Container attack (fission makes fuel salt more oxidizing)
    - Progressive degradation of heat removal capabilities and restoration via chemistry control system
      - Density
      - Boiling point
      - Melting point
      - Viscosity
      - Thermal conductivity
      - Heat transfer properties
    - Fissile material plate-out
    - Solubility (fuel, actinides, fission products)

Safety Implications of These Changes Need to be Determined

- Not all possible chemical compositions need to be explored
- Need to determine which parameters are important to the safety case and degree of variability allowed
- This might be considered to be a chemical version of the Specified Acceptable Fuel Design Limits (SAFDL) as described GDC 10 in Appendix A of 10CFR50
- Cladding as a principal fission product barrier is included in current heterogeneous fuel qualification program
  - For MSRs the reactor vessel and associated piping is covered by ASME standards
  - Thus, only the fuel itself needs to be considered in the MSR qualification process

A Possible Approach

- In submitting an application, the applicant has assumed certain properties of fuel/coolant which are then used in the safety analysis in order to meet the regulatory requirements
- This set of data form the basis for the determination of the parameters impacting the safety case
- Sensitivity studies can be used to determine the limiting values for these properties, that if exceeded, could result in a plant exceeding the safety envelope
  - Not only thermal hydraulic properties need be considered, but other properties such as
    - Solubility of fission products in the salt which might impact the source term
    - Plate out of radioactive material or solubility limits assumed for the fuel

A Possible Approach (continued)

- Once the important parameters are identified, experimental measurements and data can be developed to indicate the impact of salt composition on the important parameters
- Major drawback is the variability of salts being proposed by various designers
- Chloride salts need irradiation data to confirm their stability in reactor environments

Qualification of MSR Fuel May Be Less Expensive and Time Consuming than Current Heterogeneous Fuel but Still Will Require Significant Expenditure of Resources

- No irradiation
- No hot cell examination
- Can be accomplished without radioactive isotopes. May use natural U and surrogate for Pu (chemical behavior not dependent on isotopic composition)
  - Only chemically insignificant quantities of trans-plutonium elements anticipated
- Small samples (special effects tests) / no geometric requirements
- It is necessary to define a process that meets the regulatory requirements before DOE or Industry can establish the R&D needs