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DOE-NE Fast Reactor Methods and Safety R&D Program

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Argonne National Laboratory

Advanced Reactor Modeling and Simulation Workshop

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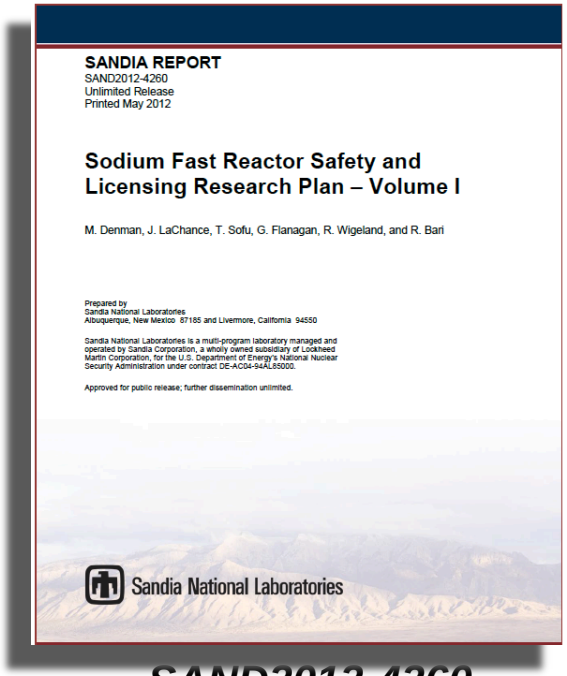
Fast Reactor Safety R&D Gap Analysis

- In 2012, five topical fast reactor safety gap analyses were completed through expert elicitation with 40+ specialists from the DOE lab complex, academia, industry and international bodies :
  - Fuels and Materials
  - Accident Initiators and Sequences
  - Computer Codes and Models
  - Sodium Technology
  - Source Term Characterization
- In 2014, smaller group of national lab experts developed a consolidated plan to prioritize R&D:
  - Coordinated knowledge preservation and management effort
  - Maintenance of, and improvements to, legacy fast reactor computer codes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Sodium Fast Reactor Safety and Licensing Research Plan – Volume I

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- Fast Reactor Methods & Safety R&D Program has since aimed to address these recommendations toward closure of the identified gaps

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SFR Fuels Irradiation Testing and Physics Analysis Databases

- **EBR-II Metallic Fuel Irradiation Testing Database**:
  - Detailed pin-by-pin fuel irradiation information: Digitized micrographs, profilometry measurements, gamma scans, porosity and cladding strain measurements, and scans for other microstructural characteristics to support fuel qualification and code validation.
- **IFR Materials Information System** and **EBR-II Physics Analysis Database**:
  - Pin-by-pin fuel fabrication and core load information for each EBR-II cycle.
  - Operating parameters, temperature, fluence, and burnup predictions as input to fuels performance codes and for validation of depletion capabilities.
- **FFTF Metallic Fuel Irradiation Testing Database:**
  - Test Design Descriptions (fabrication data and QA documentation for IFR-1 and MFF series of metal fuel tests)
  - Reports and available operational data for irradiation cycles
  - Results for impact of metal fuel tests on reactor operating parameters such as reactivity feedbacks and direct measurement data (in-core assembly growth, assembly pull forces, IEM cell exams).

SFR Safety Testing Databases

- **EBR-II Safety Test Database**: ~80 experiments from the comprehensive shutdown heat removal, balance of plant, and inherent control testing program conducted at EBR-II during 1984-97 period.
  - Including the landmark inherent safety demonstration test (unprotected station blackout)
- **FFTF Passive Safety Testing Database**: Natural circulation tests as a reliable means of decay heat removal during unprotected loss-of-flow transient, extending passive safety experience to a large-size SFR
  - Including impact of unique core restraint system design and GEM device
- **TREAT Test Database**: Archive of documents, meta- and numerical data from ~800 one-of-a-kind tests as the basis of present knowledge of transient fuel behavior on key phenomena related to transient fuel performance including fuel failures.
- **SFR Component Reliability Database**: Based on combination of original CREDO data, as well as revisited EBR-II, FFTF, and FERMI run logs, to support Fast Reactor PRA.

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NQA Equivalence Plan to Assure Acceptability of Historical Test Data

- Characterization of Historical Information:
  - Database development efforts have so far emphasized preserving historical information and its organization in modern electronic format for user friendly access.
    - Data and information are entered and managed in accordance with applicable QA and regulatory requirements in the new electronic format, but its pedigree is not addressed.
  - An evaluation of the historical metallic fuel irradiation information is ongoing
    - Determine if it provides a sufficient technical basis and QA pedigree to support a future SFR fuel qualification activity.
- Following methods are considered for qualification of legacy data:
  - **QA Program Equivalency**: Determine if the acquisition, development, or processing of data have been performed in accordance with sound technical, administrative practices or procedures in compliance with requirements and guidance of NQA 1.
  - **Peer Review**: Independently evaluation of data to determine if the employed QA methodology is acceptable and confidence is warranted in the data acquisition.
  - **Data Corroboration**: Determine if subject matter data comparisons can be shown to substantiate or confirm parameter values.
  - **Confirmatory Testing**: When tests can be designed and performed to establish the quality of existing data.

Fast Reactor Benchmarks

- **MONJU Benchmark:** Mixing and heat transfer in upper plenum of MONJU during the turbine trip test for validation of thermal stratifications using CFD methods
- **PHENIX End-of-life Tests**
  - Natural circulation heat removal test
  - Asymmetric control rod withdrawal test
- **EBR-II Benchmarks:** Analysis of a protected loss of flow and the unprotected station blackout tests to demonstrate potential of a pool-type metal-fueled SFR to survive accidents far more severe than Fukushima
- **FFTF Benchmark:** Analysis of unique FFTF passive safety test (unprotected loss of flow) based on benchmark specification from PNNL
  - Including the response of passive gas expansion module (GEM) device and core radial expansion feedback from the unique core restraint system design based on “limited free-bow” concept

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EBR-II Benchmarks: Structure

- Four-year program (June 2012-June 2016)
  - Phase 1: Blind simulation of two tests (effectively five separate benchmarks)
  - Phase 2: Model refinements, extended comparisons against plant data, code-to-code comparisons, sensitivity studies, and results qualification
- 19 participants, representing 11 countries – largest IAEA FR CRP
  - Developed benchmark specifications and provided technical support
  - Organized and conducted four research coordination meetings and assembled final IAEA report
  - Involved millennials in analysis of legacy tests (early career+postdocs)
- Primarily supported with nodal neutronics, systems and subchannel analysis codes (DIFF3D/REBUS/PERCENT, SAS4A/SASSYS-1, SAM)
  - Neutronic benchmark for k_{eff}, β, and reactivity feedback coefficients
  - Two separate benchmarks for SHRT-17 and SHRT-45R tests
  - Additional benchmark for two instrumented subassemblies

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EBR-II Benchmarks: Tests studied

- Two EBR-II Shutdown Heat Removal Tests studied:
  - SHRT-17: the most severe protected loss of flow, started from 100% power and flow, natural circulation flow established
  - SHRT-45R: the most severe unprotected station blackout, started from 100% power and flow
  - Instrumented subassemblies XX09 (fueled) and XX10 (steel pins)
- EBR-II core, reactivity feedback coefficients, and primary coolant system modeling (with given IHX-IS inlet flow and temp. as BCs)
- Sensitivity studies performed on:
  - Heat transfer coefficients
  - Fuel porosity
  - Reactivity coefficients
  - Axial heat conduction
  - Inclusion of gamma heating
  - Decay heat modeling
  - Pump characteristics
  - Plena and cold pool modeling
  - Neutronics modeling level (spherical harmonics, discrete ordinates)

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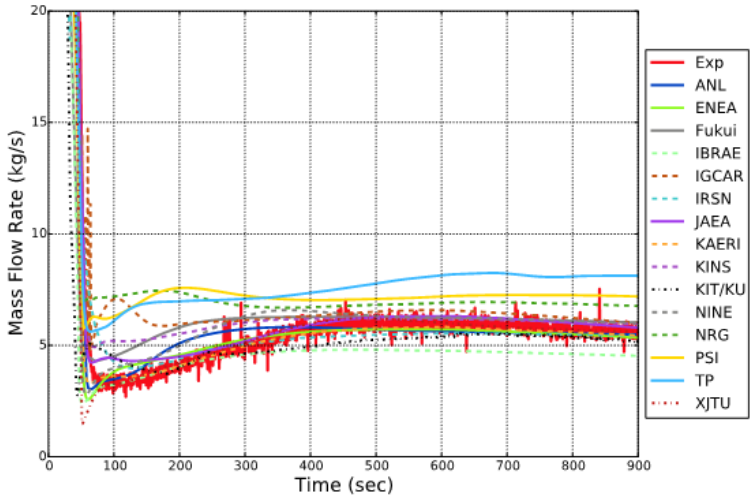
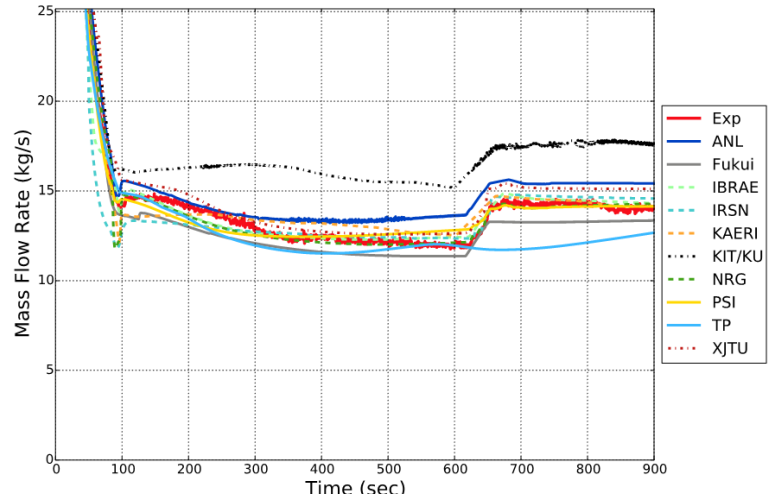
EBR-II Benchmarks: Results

SHRT-17

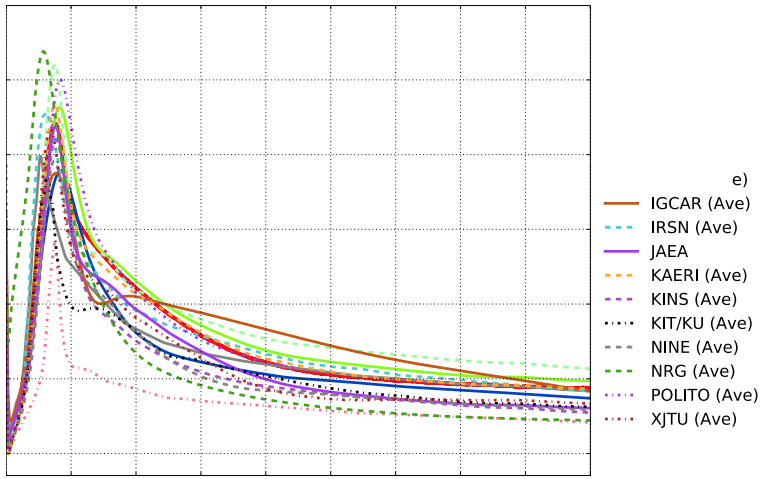
SHRT-45R

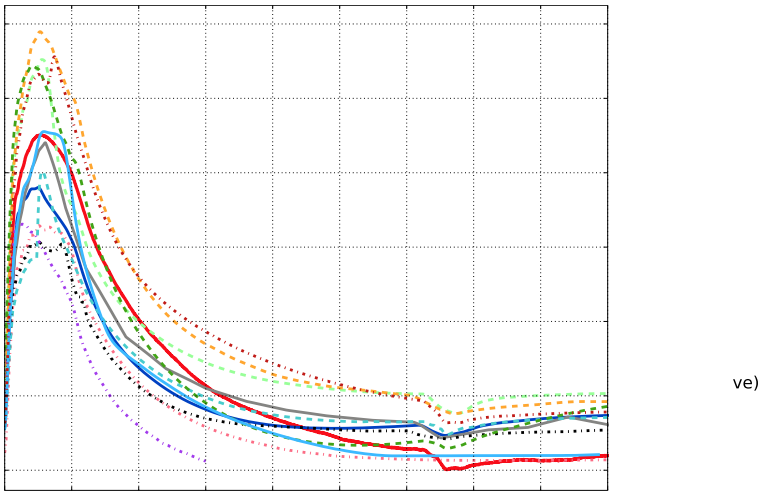
Primary Flow





XX09 Outlet Temp.





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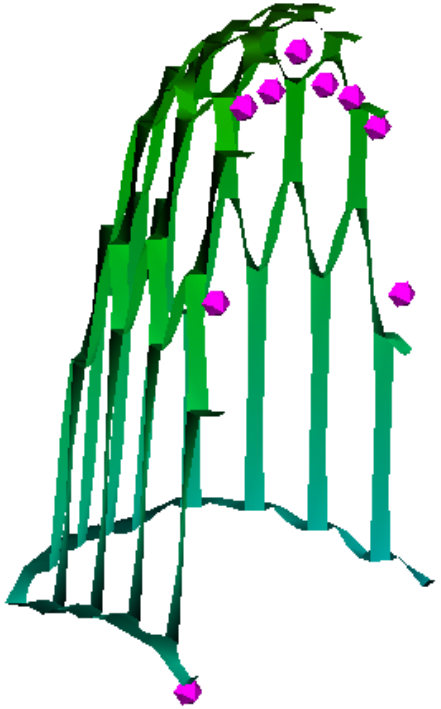
EBR-II Benchmarks: Lessons Learned

- The two primary pumps and associated piping must be modeled individually
- 0-D representation of upper plenum and IHX inlet plenum is inadequate
- Thermal stratification in the cold pool is also important improve pump inlet temperature predictions
- Heat transfer between the Z-pipe and the cold pool and leakage paths between the upper plenum and cold pool are important and evaluated only parametrically
- Heat transfer between the instrumented subassemblies and the surrounding subassemblies requires detailed modeling, particularly for XX10 (non-fueled subassembly surrounded by fueled subassemblies)
- Modeling axial heat conduction near the instrumented subassembly flowmeters produce improved temperature predictions



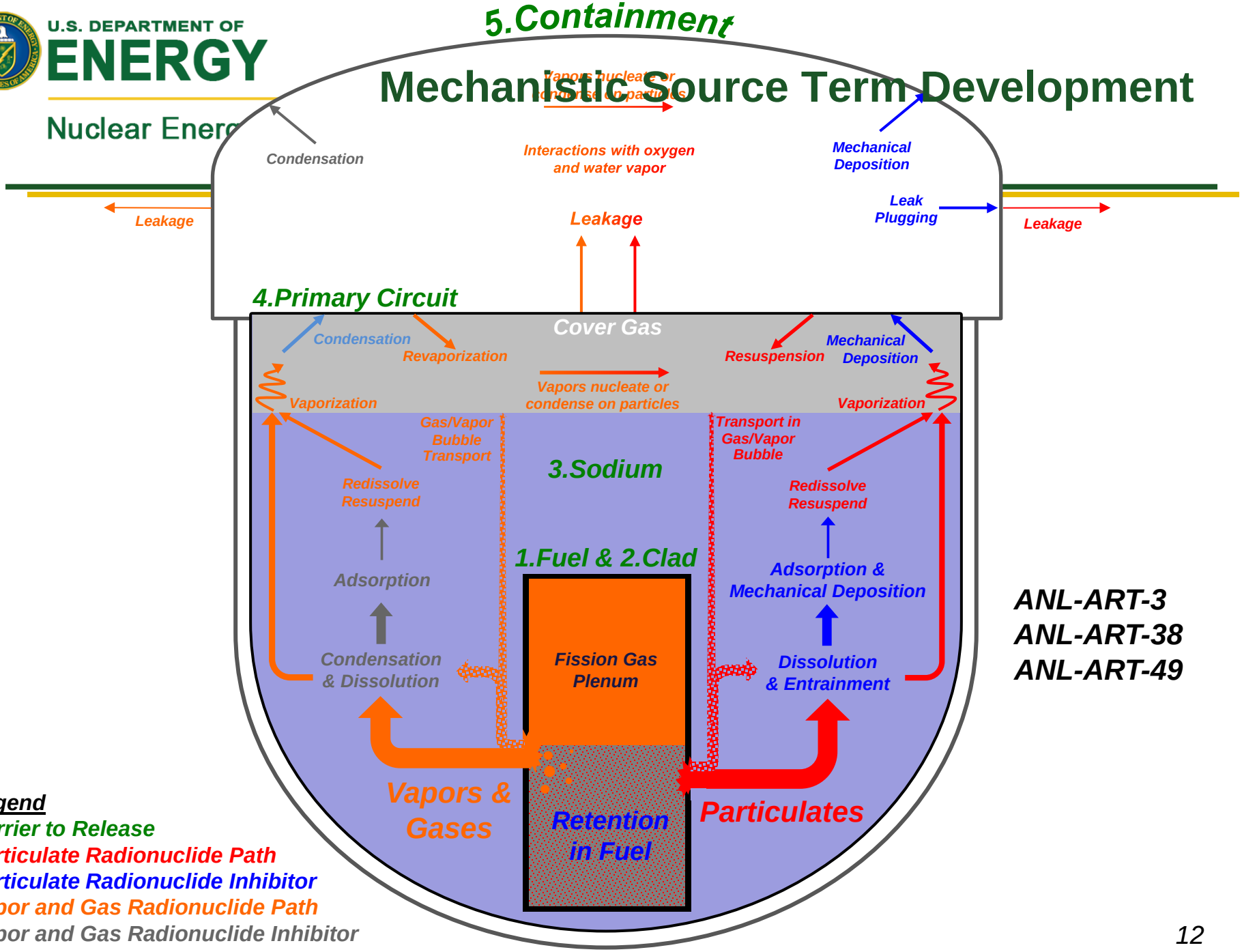
SAS4A/SASSYS-1 sub-channel results compared to thermocouple data at the beginning of SHRT-17

Additional EBR-II Tests Analyzed

Additional benchmark specifications developed for other EBR-II tests

- **BOP-301 and BOP-302R**
  - Loss-of-heat-sink tests – intermediate pump trip at full primary flow
  - BOP-301: 50% power, 2/3 intermediate loop flow
  - BOP-302R: full power and intermediate loop flow
  - Simple flow mixing model in cold pool was added
  - Good prediction of inlet plena and Z-Pipe inlet temperatures, improved IHX inlet temperature predictions
- **SHRT-43R and SHRT-45**
  - Unprotected loss-of-flow
  - SHRT-43 initiated at full flow and 2/3 power but at the same window as SHRT-45R
  - Same as SHRT-45R, but a different test window (different core load)
  - Similar prediction accuracy as for SHRT-45R, giving added confidence to the SHRT-45R results

Mechanistic Source Term Development



ANL-ART-3
ANL-ART-38
ANL-ART-49

Legend
Barrier to Release
Particulate Radionuclide Path
Particulate Radionuclide Inhibitor
Vapor and Gas Radionuclide Path
Vapor and Gas Radionuclide Inhibitor

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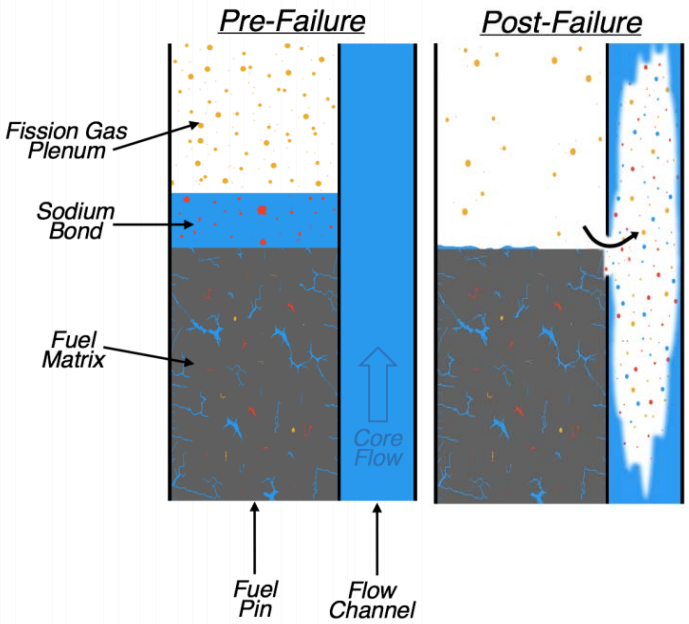
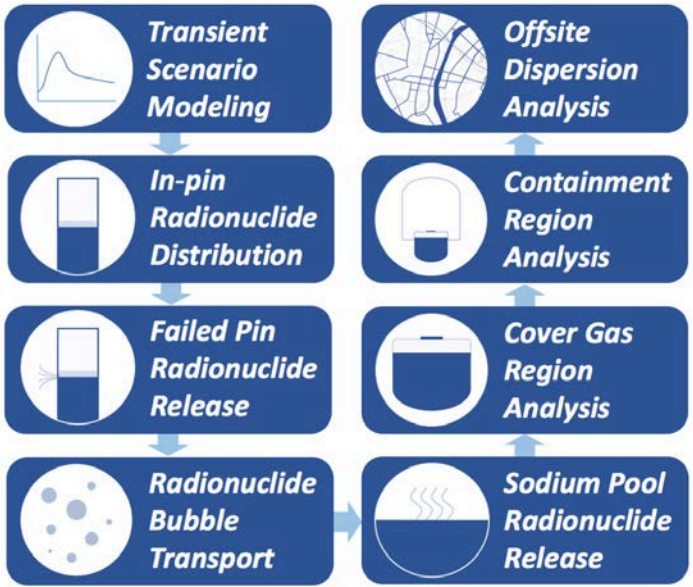
Mechanistic Source Term Development

- Identified as Possible Licensing Gap
  - No previous mechanistic source term assessments for metal fuel, pool-type SFRs
  - Need to identify and communicate unique phenomena to the regulator
- Radionuclide Release from Failed Metal Fuel Pins
  - Release fraction estimates developed based on fuel pin burnup level and failure conditions
    - Extensive review of past accidents and experimentation as well as chemistry modeling
- Trial Mechanistic Source Term Calculation
  - A best-estimate calculation of radionuclide release
    - From initiating event to offsite consequence
    - Additional goal to identify influential radionuclides & phenomena and possible code/data gaps
  - Analysis includes many computer codes
    - SAS4A/SASSYS-1, HSC Chemistry, Bubble Transport Code, CONTAIN-LMR/MELCOR, etc.





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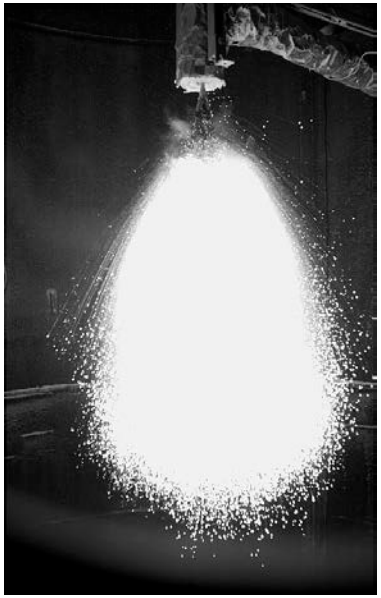
Safety Analysis Code Improvements: MELCOR and CONTAIN-LMR Integration

- Motivation:
  - Provide CONTAIN-LMR sodium accident analysis capability under MELCOR integrated severe accident code for SFR source term assessments, level 2/3 PRA, and containment DBA analyses.
- Status:
  - Sodium chemistry models from CONTAIN-LMR are implemented into MELCOR 2.1
  - Additional interface data variables are being added for the atmosphere chemistry model.
  - A combination of experimental and code-to-code and benchmarking studies are being conducted
- Initial applications:
  - Trial mechanistic source term calculations
  - JAEA sodium fire modeling collaboration with data from Sandia and JAEA experiments
  - Planned MELCOR and ASTEC-Na crosswalk



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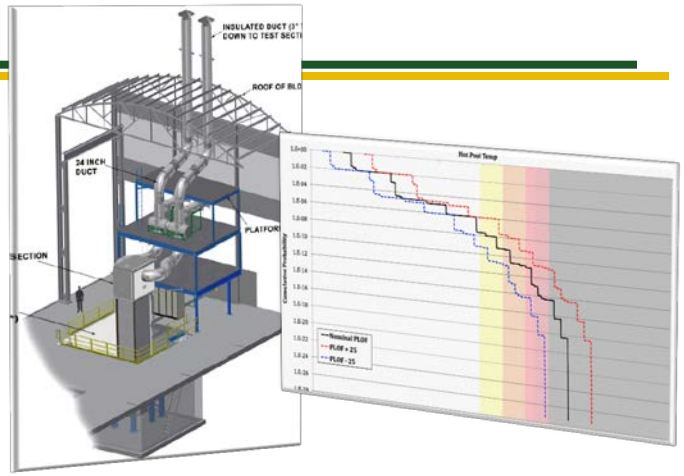
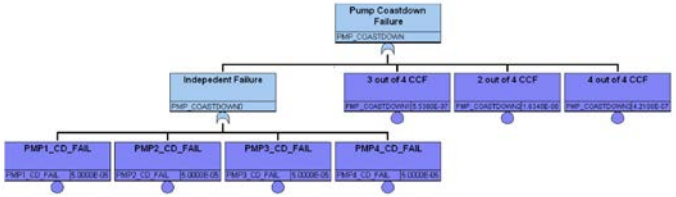
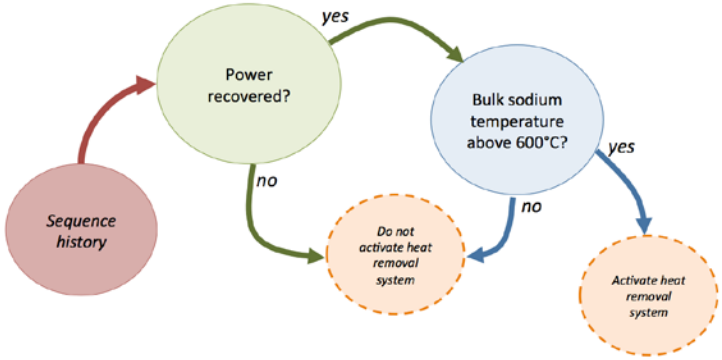
Probabilistic Risk Analysis for Advanced Reactors



- Passive Safety System Reliability
  - Development of a mechanistic assessment of passive safety system reliability and success criteria
    - Integration of system code modeling results directly into PRA event sequences
    - Allows best-estimate plus uncertainty analysis rather than the use of conservative assumptions
- Component Reliability
  - Developing reliability estimates based on failure data from a variety of past sources and analyses
    - Leveraging U.S. database development efforts
- Simulation-based (Dynamic) PRA
  - Seamless integration of time-dependent processes and mechanistic assessments into PRA
    - Important for passive system performance and external event analysis
    - Coupling system codes with PRA tools such as ADAPT and RAVEN





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ARC FOA: PRISM PRA Update

Program Team

HITACHI

Argonne NATIONAL LABORATORY

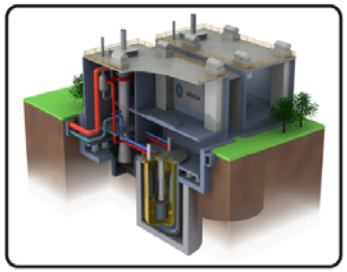
Relevant Prior Work

- **ESBWR DCD PRA** in which passive design and digital I&C modeling techniques were advanced
- **ALMR program** - Initial PRA development state-of-the-art in its era
- **PRA standard development** including Non-LWR standard
- **ANL expertise in SAS4A/SASSYS-1** mechanistic tool, passive reliability, and sodium component data

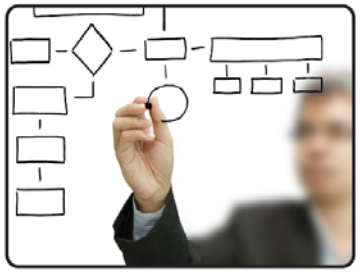
Development/Modernization of an Advanced Non-LWR Probabilistic Risk Assessment (PRA)

Program Objective: *Next generation PRA methodologies for Non-LWRs and a modern full scope PRISM PRA to serve as the foundation for risk informed R&D.*

Advanced Reactor Design



Next Generation Analysis Tools



Technical Approach

- Develop detailed internal events model with support from various disciplines including thermal hydraulics, I&C, and human factors.
- Leverage the internal events PRA logic and analysis to estimate the design sensitivity to seismic, fire, high winds, flooding, tsunami and other hazards.
- External event analysis in areas such as seismic, fire, high winds, tsunami. Priority on breadth for external event analyses so that a cliff edge event is not overlooked

Technical Challenges

- Current PRA industry methodologies are focused on Gen II active plants, with less focus on passive failure modes
- Although it does demonstrate a significant improvement in safety, the early 1990s PRISM PRA cannot support licensing and other risk informed applications in its present state
- The Non-LWR standard has not been tested against an actual Non-LWR technology
- Physics-based Success Criteria has not been tested

Program Deliverables

- PRA Methodologies Report including Initiating events selection, Mechanistic success criteria analysis, & Passive system reliability modeling.
- PRISM PRA summary report for all hazards, all modes

Anticipated Benefits of the Proposed Technology

- New PRA methodologies will be developed that will help other non-LWRs more efficiently analyze risk
- Models in a modern code will greatly simplify future risk studies--in response to regulator inquiries for example
- Risk informed R&D will allow future work to prioritized by its impact on safety and reliability

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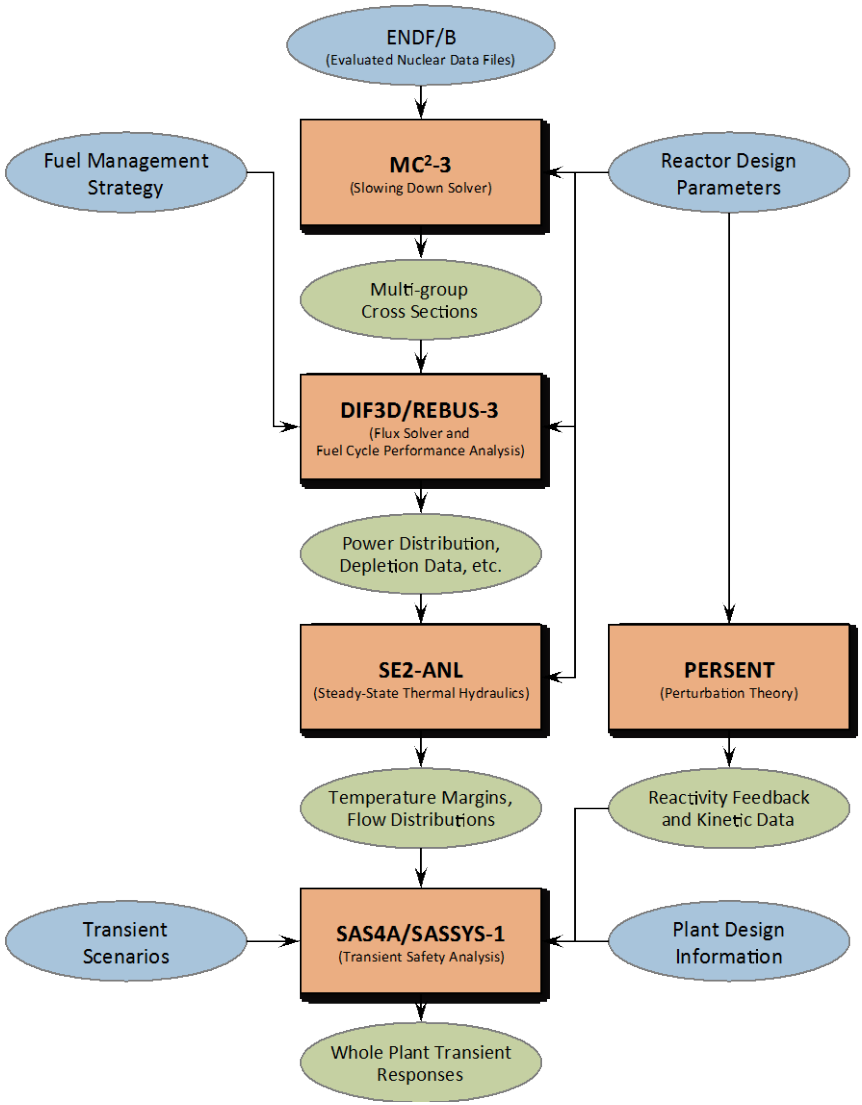
Fast Reactor Codes: Steady-state characterization

Phenomenon	Code
Neutron and Gamma Diffusion/Transport	MC^2-3 DIF3D/VARIANT
Fuel Cycle Performance	REBUS ORIGEN
Fuel Performance	LIFE-METAL
Core-Wide Thermal Hydraulics	SAS4A/SASSYS-1 SE2-ANL
Single-Channel Thermal Hydraulics	Nek5000
Fuel-Assembly Bowing & Core Radial Expansion	NUBOW-3D



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Fast Reactor Codes: Transient characterization

Phenomenon	Code	Role
Fission Gas Behavior	LIFE-METAL SAS4A/SASSYS-1	*Primary* *Secondary*
Fuel and Clad Motion	SAS4A/SASSYS-1 LIFE-METAL	*Primary* *Secondary*
Primary/Intermediate System Heat Transport	SAS4A/SASSYS-1	*Primary*
Structural Response	NUBOW-3D SAS4A/SASSYS-1	*Primary* *Primary*
Inherent Reactivity Feedback	PERSENT SAS4A/SASSYS-1	*Primary* *Primary*
Passive Heat Removal	SAS4A/SASSYS-1	*Primary*
Sodium-Water Interactions	SWAAM-II	*Primary*
Sodium Fires	MELCOR CONTAIN-LMR SOFIRE	*Primary* *Secondary*
Source Term	ORIGEN MELCOR CONTAIN-LMR	*Primary* *Primary* *Secondary*

Qualification of Fast Reactor Codes

- Analyses supporting a license application will require NRC-acceptance of codes and methods
  - Development and maintenance of codes in accordance with an acceptable QA framework, demonstration of sufficient model maturity and fidelity
- Current RTDP effort dedicated to development of software quality assurance (SQA) framework for SFR safety codes
  - Goal: Establish and implement plans to ensure that code development activities are performed in accordance with applicable SQA requirements
- Initial focus on development and implementation of a provisional SQA program for SAS4A/SASSYS-1
  - Compliance with regulatory guidance and commercial dedication reqs:
    - NUREG/BR-0167, NQA-1-2008/2009, EPRI TR 3002002289, IEEE Standards, etc.
  - Sustainable SQA: Necessary to account for increased quality rigor in budgeting and scheduling

Qualification of SFR Codes/Methods: SAS4A/SASSYS-1 SQA Plan

- **Staged development process:**
  - Identify and develop appropriate supporting documentation
    - Software Quality Assurance Plan, Configuration Management Plan, Coding Standards, various procedures, Software Requirements, Software Design, Verification and Validation...
  - Implement provisional program: Training, procedures etc.
  - Regular program audits to identify and prioritize gaps
  - Resolution of gaps as per audit results
  - Continued maintenance and improvement throughout software lifecycle
- **Technical activities:**
  - Expansion of V&V test suite, implementation of expanded automated testing, improved compiler support, etc.