Benchmark Analyses of the Shutdown Heat Removal Tests Performed in the EBR-II Reactor

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International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13)

March 6, 2013

Background and Objectives

- Evaluations of sodium-cooled fast reactor (SFR) passive safety capabilities acquired increased importance in the aftermath of the Fukushima accident.
- These evaluations require simulation tools and models validated against data from severe accident experimental simulations.
- In the U.S., the potential of SFRs to survive accident initiators more severe than Fukushima, with no core damage, was demonstrated during the Shutdown Heat Removal testing (SHRT) program with Experimental Breeder Reactor-II (EBR-II), during which extensive amounts of core and plant data were recorded.
- This IAEA coordinated research project (CRP) provides participants with the opportunity to improve their design and safety analysis capabilities through comparisons against test data from protected and unprotected loss of flow tests conducted during that extensive testing program.
- Participants will analyze benchmark specifications developed by Argonne for two of the experiments in the SHRT program and compare their predictions against recorded data.

EBR-II Benchmarks CRP Goals

- Validate state-of the-art fast reactor safety analysis computer software
- Use comparisons against EBR-II test data to improve analytical capabilities of the CRP participants in fast reactor design and analysis
- Extend the validation and verification base of SFR safety analysis codes to accurately simulate:
  - primary coolant system behaviour during loss-of-flow transients
  - fission power history during unprotected transients
  - natural convective cooling and reactivity feedback mechanisms
- Train the next generation of analysts and designers through international benchmark exercises

Benchmark Project Participants

China	Republic of Korea
France	Netherlands
Germany	Russian Federation
Italy	Switzerland
India	USA
Japan	

EBR-II CRP: Participating Organizations	
China: *CIAE*	Japan: *Kyushu University*
China: *North China Electric Power University*	Japan: *University of Fukui*
China: *Xi'an Jiatong University*	Korea, Republic of: *KAERI*
France: *IRSN*	Korea, Republic of: *KINS*
Germany: *HZDR*	Netherlands: *NRG*
Germany: *KIT*	Russian Federation: *IPPE*
Italy: *ENEA*	Switzerland: *PSI*
Italy: *UNIPI (GRNSPG)*	USA: *ANL*
India: *IGCAR*	USA: *TerraPower*
Japan: *JAEA*	USA: *MIT (pending proposal approval)*

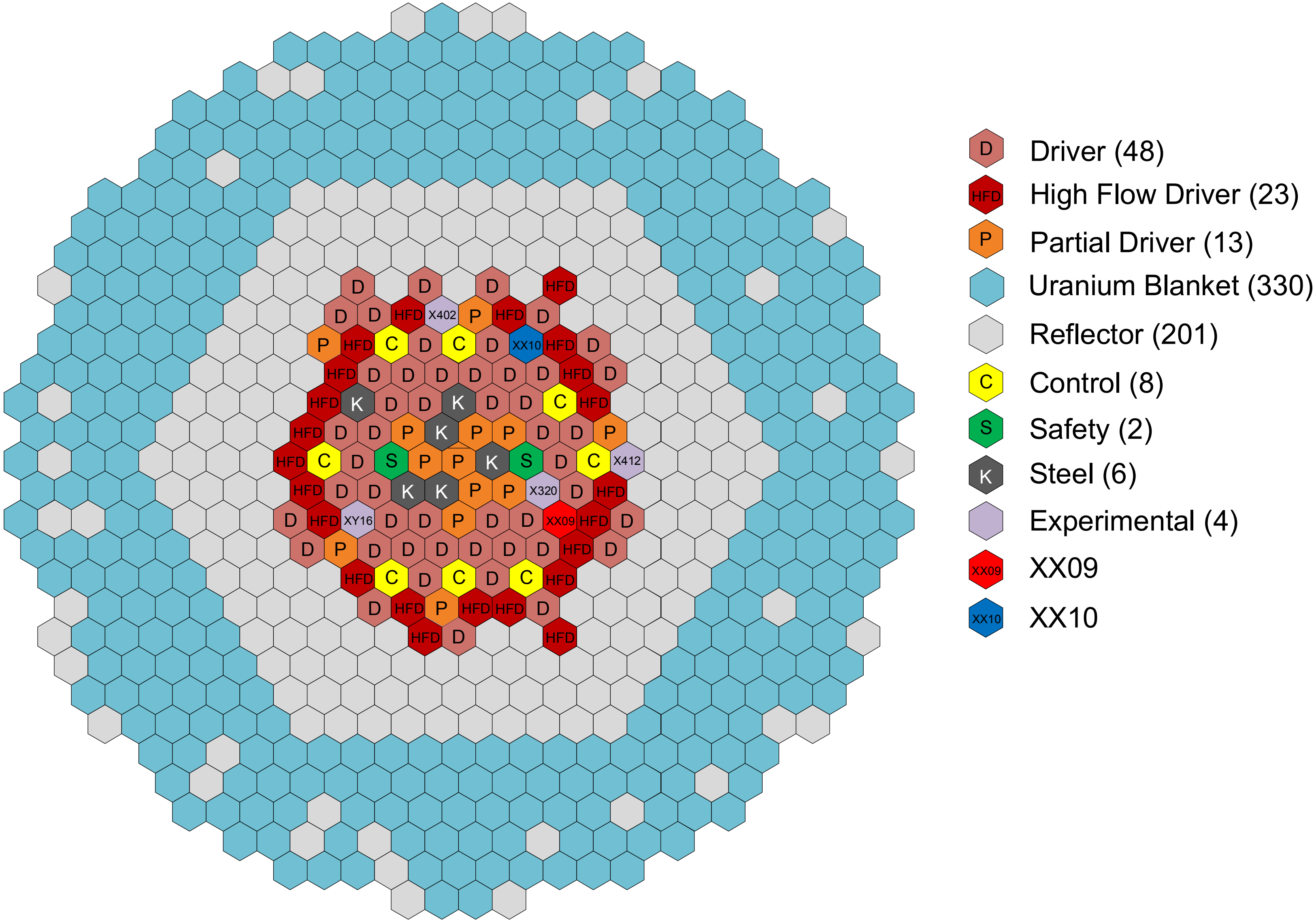
CRP Schedule

Activity	2012	2013	2014	2015	2016
Complete benchmark specifications	X				
First research coordination meeting	X				
Initial model development and blind simulations	X	X			
Preliminary assessment of simulation results		X			
Second research coordination meeting – exchange of results and evaluation of modeling issues		X			
Refinement of simulations			X		
Uncertainty analysis			X		
Third research coordination meeting			X		
Draft of final report				X	
Identification of future R&D and experimental needs to resolve open issues				X	
Final research coordination meeting				X	
Final evaluation, publication of final report					X

The EBR-II Reactor and Plant

- A uranium metal-alloy-fuelled sodium-cooled fast reactor coupled to a full power plant and fuel recycle facility
- Designed and operated by Argonne for the USDOE at the former Argonne-West site in Idaho (now the Materials and Fuels Complex at the current INL)
- Thermal power of 62.5 MW, electric output of about 20 MW
- Operated for thirty years, 1964-1994
- Original mission (first five years) was demonstrating the feasibility of a closed fuel cycle that would allow for sustained operation
- Operated for the next ten years primarily as an irradiation test facility for advanced metal and oxide fuels
- Focus of the final fifteen years was experiments to demonstrate the importance of passive safety in LMR's

EBR-II Core Layout



EBR-II Shutdown Heat Removal Tests

- Between 1984 and 1987, Argonne conducted the SHRT landmark whole-plant test program to study anticipated off-normal conditions and severe accident initiators.
- SHRT program objectives:
  - Support US LMR plant design
  - Provide test data for computer code validation
  - Demonstrate passive reactor shutdown and decay heat removal in response to a range of protected and unprotected transients
- Of particular interest are the shutdown heat removal tests that demonstrated the potential for SFR's to survive severe accident initiators.
- Two of these tests – SHRT-17 and SHRT-45R – have been chosen as the benchmark problems for this IAEA CRP
- Both tests emulate a total loss of pumping power (normal and emergency) to the plant operating at full power and flow conditions.

EBR-II Shutdown Heat Removal Tests (cont.)

- *SHRT-17*
  - June 20, 1984
  - Protected Loss of Flow (PLOF) test
  - Full power and full flow at beginning of test
  - Simultaneous trip of sodium pumps and control rod scram
    - Auxiliary sodium coolant pump was off throughout
  - Demonstrated effectiveness of natural circulation cooling characteristics to remove residual heat and keep core cooled during accidents
    - Temperatures rose to high, but acceptable, levels
    - Thermal inertia of sodium effective to protect reactor from potentially adverse consequences from PLOF or PLOHS
  - Transient involved transition from forced convection (established during the initial steady-state and during the pump coast-down) to natural convection (after the pumps stopped)
    - One of the lowest minimum transient coolant flow rates of any EBR-II tests

EBR-II Shutdown Heat Removal Tests (cont.)

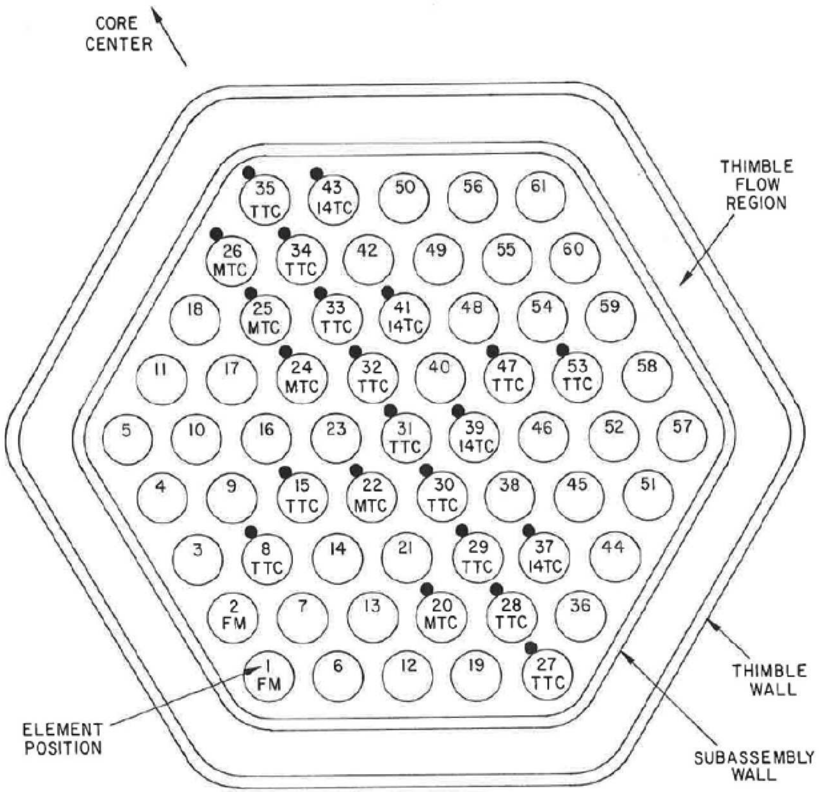
- *SHRT-45R*
  - April 3, 1986
  - Unprotected Loss of Flow (ULOF) test
  - Full power and full flow at beginning of test
  - Simultaneous trip of all sodium pumps
    - Auxiliary sodium coolant pump on battery power
  - <u>Reactor control system was manipulated to avoid scram</u>
  - Decay heat removal system continued to operate at its rated capacity as a passive device
  - Demonstrated effectiveness of passive reactivity feedbacks to terminate the fission process
    - Temperatures rose to high, but acceptable, levels
    - Natural phenomena (e.g. thermal expansion) effective to protect reactor from potentially adverse consequences from ULOF or ULOHS
  - Coupling of neutronic, thermal-hydraulic and structural phenomena creates challenging benchmark problem

SHRT-45R Available Core Reactivity Feedback Data

- Whole core (integral) coefficients and distributed worth curves for a fixed Argonne nodal mesh structure
  - Fuel temperature (Doppler)
  - Axial clad and hex tube steel expansion (fuel section only)
  - Sodium expansion (density)
- Control rod and safety rod bank extension
  - Insertion worth curve for control rods
  - Insertion worth curve for safety rods
- Core grid plate expansion
  - Coefficient for uniform dilation of core
- Thermal bowing (subassembly)
  - Distributed worth curves for a fixed ANL nodal mesh structure for fuel, steel and sodium
  - Historical EBR-II whole-core bowing coefficient

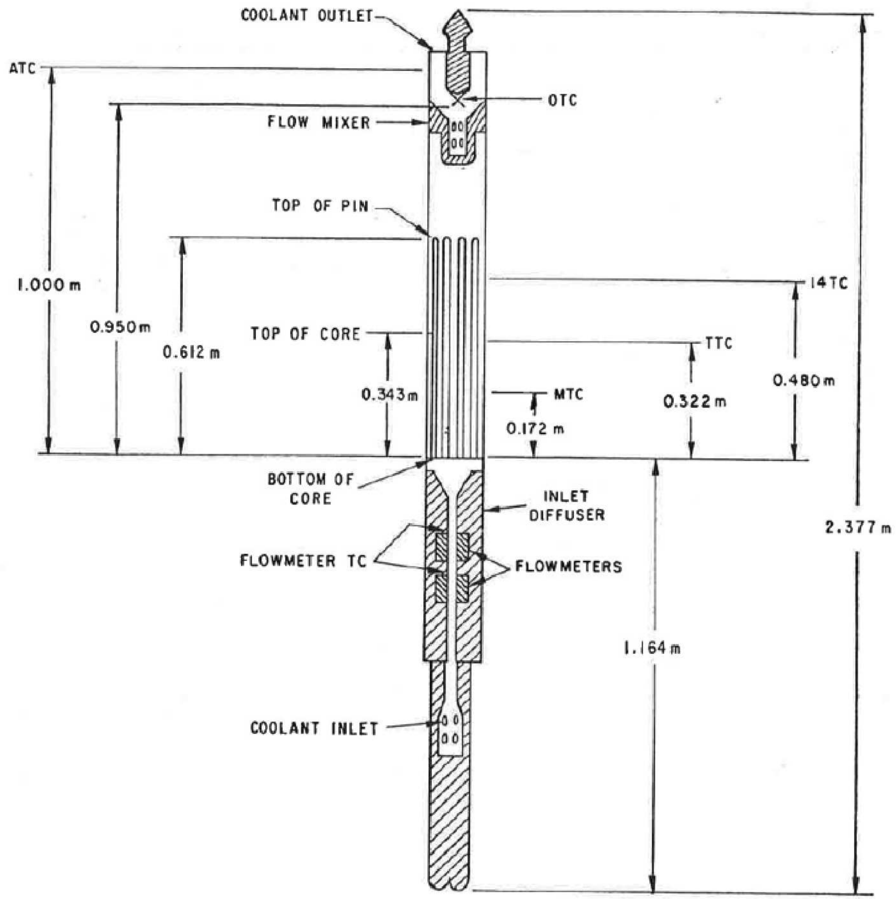
Instrumented Subassembly Measurements

- Availability of intra-assembly thermal and flow data between the two instrumented subassemblies (XX09 and XX10) is a unique feature of the tests



LEGEND

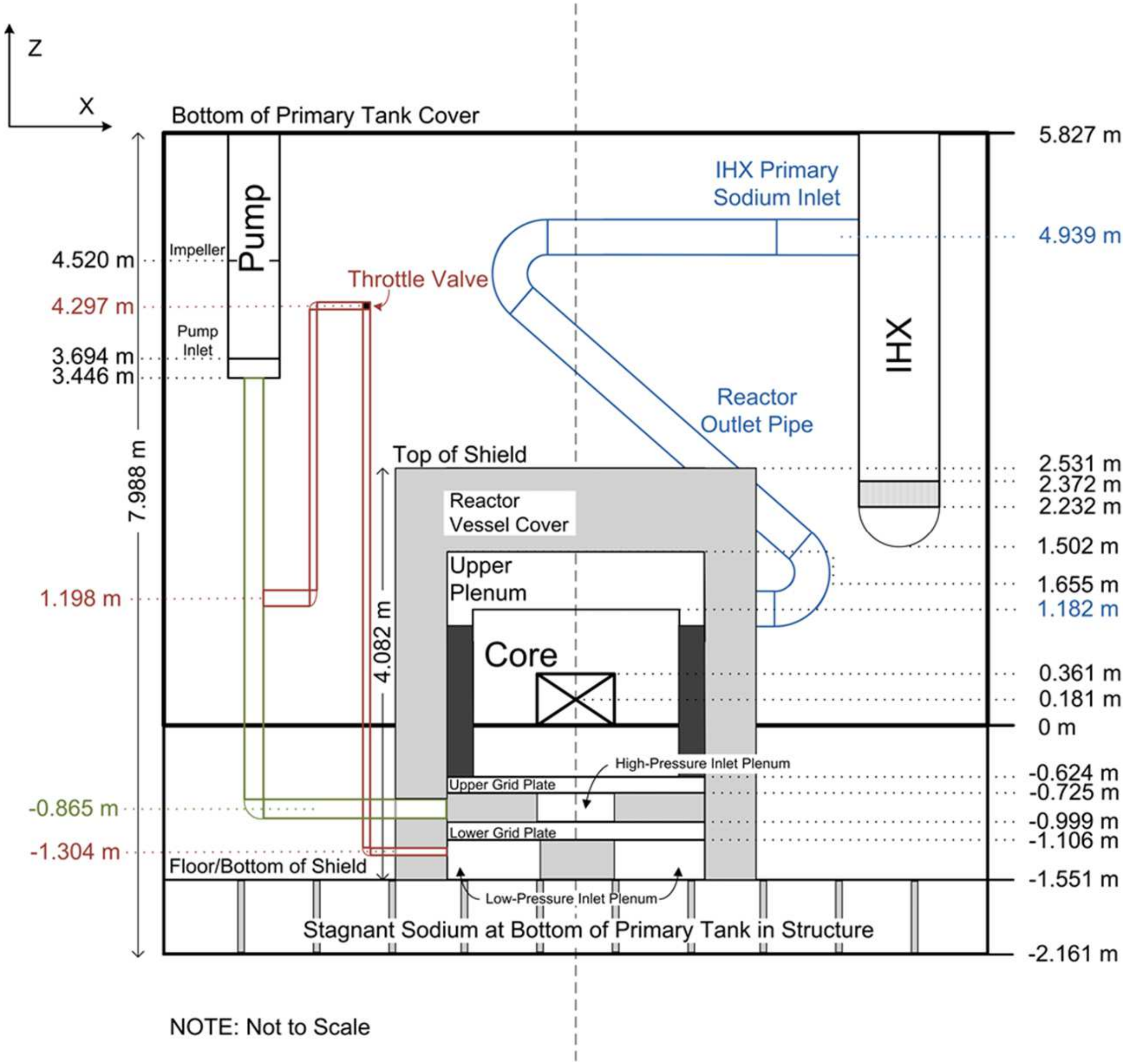
FLOWMETER-CONDUIT LEADS	FMC	2
MIDPLANE CLAD TC	MTC	5
TOP-OF-CORE	TTC	13
ABOVE-CORE TC	14TC	4
OUTLET-COOLANT TC	OTC	2. (not shown)
ANNULUS THIMBLE REGION TC	ATC	2 (not shown)
FLOWMETER TC	T	2 (not shown)
FLOWMETER	FM	2 (not shown)



Benchmark Specifications

- Two volumes –
  - Benchmark specification and data requirements for both tests
  - Neutronic benchmark specification for SHRT-45R
- Request complete specifications from IAEA for formal CRP participation
- These documents include:
  - Detailed descriptions of both tests
  - Detailed reactor core descriptions for both tests
  - Detailed descriptions of the two instrumented subassemblies
  - Detailed description of the primary cooling system
  - Benchmark model of the cooling system
  - Material properties
  - Data needed for calculation reactivity feedback coefficients
  - Detailed decay heat parameters
  - Isotopic compositions of core and blanket subassemblies
  - Essential measurements for benchmark forcing functions and unique plant-specific component characteristics
  - Detailed descriptions of the instrumentation used to measure the data

Benchmark Model of Primary Vessel Components



SHRT-45R Neutronics Benchmark Results to Calculate

- Core multiplication factor
- Effective delayed neutron fraction
- Power distribution of each subassembly
  - Including fission and gamma heat
- Fission and decay heat power for 15 minutes, assuming reactor scram at the beginning of SHRT-45R
- Reactivity feedback coefficients:
  - Axial expansion
  - Radial expansion
  - Sodium density
  - Doppler constant
  - Control rod expansion

Basic Benchmark Values to Calculate

- Parameters to be calculated as a function of time, for both transients, for
  - comparison with measurements:
    - High-pressure and low-pressure inlet plena temperature
    - Z-pipe inlet temperature
    - IHX primary side inlet temperature
    - IHX intermediate side outlet temperature
    - Primary sodium pumps sodium mass flow rates
    - XX09 and XX10 temperatures and sodium mass flow rates
    - SHRT-45R only: fission power
  - code-to-code comparison:
    - Peak fuel temperature
    - Peak cladding temperature
    - Peak in-core coolant temperature
    - Minimum margin to coolant boiling
    - SHRT-45R only: net reactivity and decay heat power

Summary

- IAEA-coordinated research project on benchmark analyses of two landmark EBR-II passive safety tests has been initiated.
  - Objective is to improve SFR design and analysis capabilities for predicting reactivity feedback and/or natural convective cooling characteristics.
  - Extensive data recorded during the EBR-II tests provide an excellent basis for code and model verification, validation, and qualification.
- The benchmark specifications describe the EBR-II geometry and plant components, provide recommended material properties and heat transfer correlations, and identify the initial conditions, power production conditions, and transient boundary conditions needed for the simulation of the tests.
- The ultimate assessment of success will be in the accuracy of the prediction of the fission power history during an Anticipated Transient Without Scram by the coupled neutronic/thermal-hydraulic/structural models.
- Evaluation of uncertainties for prediction of flow rates, fuel, cladding, coolant, and structure temperatures, and transition to natural circulation will also be within the scope of the project.