N.IN.E. Nuclear and INdustrial Engineering

Technical Meeting on State-of-the-art Thermal Hydraulics of Fast Reactors

26-30 September 2022, C.R. ENEA, Camugnano, Italy

NINE Thermal Hydraulic Contribution in IAEA CRPs Focused on Sodium Fast Reactors

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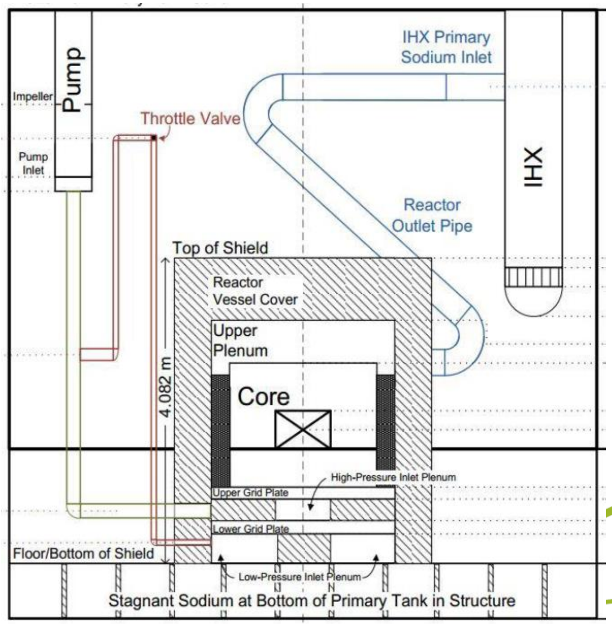
- Global interest in fast reactors has been growing since their inception in 1960 because they can provide efficient, safe, and sustainable energy
  - Their closed fuel cycle can support long-term nuclear power development as part of the world's future energy mix and decrease the burden of nuclear waste
- In addition to current fast reactors construction projects, several countries are engaged in intense R&D and innovation programs for the development of innovative fast reactor concepts
- In this framework, NINE is very actively participating in various International benchmark
  - Aiming at demonstrating the applicability of its modeling methodology to fast reactor design (in particular SFRs)
  - To evaluate the level of assessment of computer codes available at NINE in respect to SFR specific features
  - To check the applicability of the NINE Validation Process, which is part of the more general framework of NEMM (NINE Evaluation Model Methodology)
- Two IAEA CRPs are analysed
  - Experimental Breeder Reactor II (EBR-II) Shutdown Heat Removal Test (SHRT-17) benchmark
  - Benchmark analysis of Fast Flux Test Facility (FFTF) Loss of Flow Without Scram (LOFWOS) test

EBR-II Benchmark

- **IAEA CRP** "*Benchmark Analyses of an EBR-II Shutdown Heat Removal Test*" (2012)
  - 4 Phases: Blind Simulation, Open Calculation, Sensitivities and Final Calculation, Qualification Process
- **Experimental Breeder Reactor II** (**EBR-II**) was a **pool type Sodium-cooled Fast Reactor** (**SFR**) located in Idaho, U.S.A., and it was designed and operated by ANL for U.S. Department of Energy
  - Rated thermal power was **62.5MW** (electric output of **20 MW**)
- Main components:
  - Two primary pumps, High and low pressure pipes, RV with lower and upper plena, Z-pipe (from UP to IHX), Intermediate Heat Exchanger
- Core configuration:
  - 637 hexagonal subassemblies (SA) divided in three regions:
    - ✓ **Central core** with 61 SA (driver-fuel, instrumented, experimental, safety and control rod)
    - ✓ **Inner blanket**, originally with blanket, now with driver or reflector SA
    - ✓ **Outer blanket**, 510 SA either blanket or reflector



Description of the SHRT-17

- In order **to demonstrate the inherent safety** of LMR type reactor, several loss of flow tests were conducted between 1984 and 1986, **Shutdown Heat Removal Test** (**SHRT**) series
- SHRT-17
  - June 20, 1984
  - Protected Loss of Flow Test
  - Full power and full flow at the beginning of test
  - Simultaneous trip of sodium pumps and control rod scram
  - Demonstrated effectiveness of natural circulation cooling to remove residual heat and keep core cooled during accident
    - ✓ Temperature rose to high, but still acceptable levels
    - ✓ Thermal expansion and thermal inertia of sodium effective in protecting reactor from potentially adverse consequences from PLOF or PLOHS
  - Coupling of thermal hydraulic phenomena in core and primary loop created challenging benchmark problem
    - ✓ E.g. th. stratification in Z-pipe and UP; axial conduction; γ-heating;…

Parameter	Condition
Initial Power	57.3 MW
Initial Flow Through Core	456.7 kg/s
Initial Intermediate Flow	301 kg/s
Initial Core Inlet Temperature	624.7 K
Primary and Intermediate Pump Coastdown Conditions	Power to motor-generator sets removed
Control Rods	Full insertion
Auxiliary EM Pump Conditions	Power removed

RELAP5-3D Nodalization Development

- Sliced nodalization approach
- Improvements after benchmark results submission:
  - 3D pool, modelled with a cylindrical 3D component
  - **Leakages paths** distributed along the primary circuit
  - Accounting for the γ-heating (after the sensitivity)
  - Detailed core model
- Main issues, due to lack of geometrical data on:
  - Inlet plena
  - Subassemblies adapters
  - Upper plenum
  - IHX primary side

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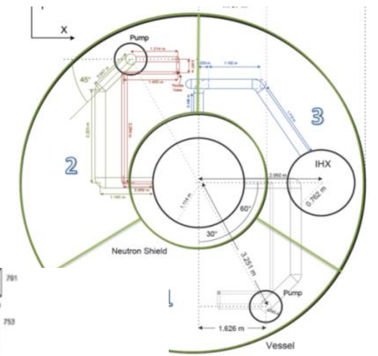
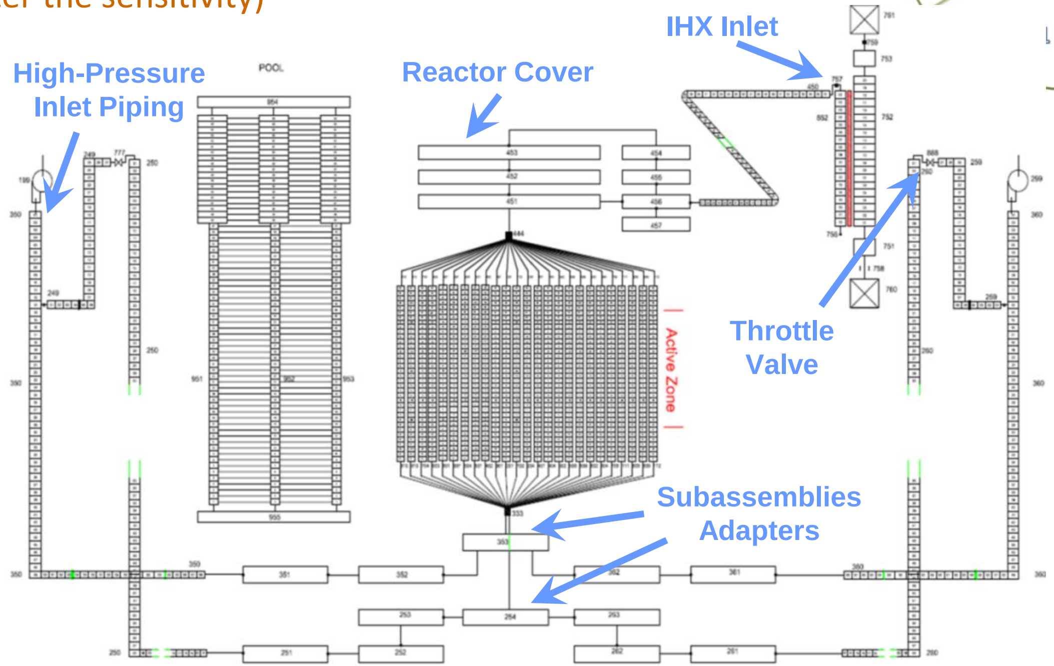
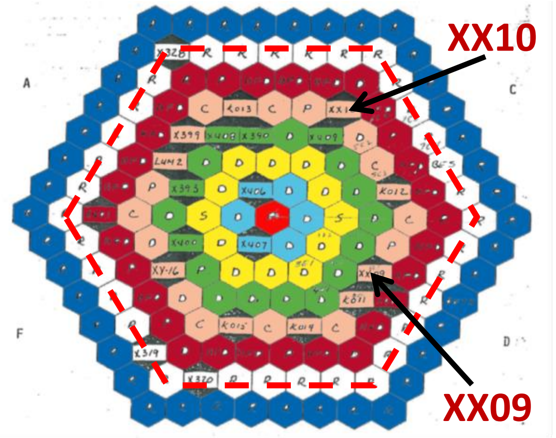
Thermal inertia can affect the transient results

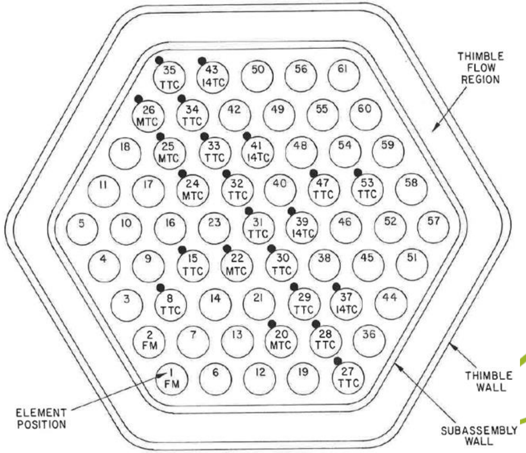




RELAP5-3D Nodalization Development

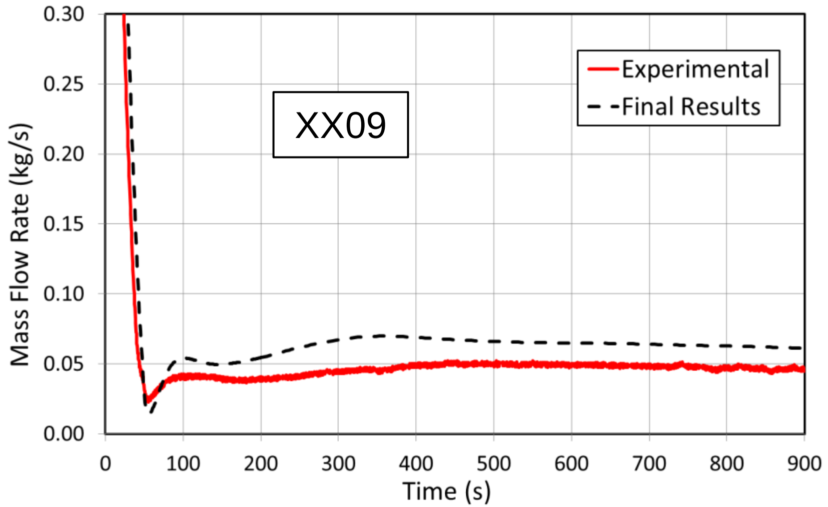
- Core nodalization (totally 97 channels)
  - First 6 rows:
    - ✓ Subassemblies modeled 1 by 1 with 81 PIPE (except for safety/control rods)
    - ✓ One HS used to simulate the active part of the rod
    - ✓ One passive HS to model the steel rods (if present)
    - ✓ 6 HS to represent the subassembly walls
  - Rows form 7 to 16:
    - ✓ 1 PIPE per type of subassembly at each row
    - ✓ 2 HS for each row
      - 1 for internal rods
      - 1 for subassembly wall
- Instrumented subassemblies (XX09 and XX10)
  - XX09 (59 fuel elements), XX10 (16 steel rods)
    - ✓ Thimble flow region modeled with a PIPE
    - ✓ 1 cylindrical HS used to simulate the subassembly wall

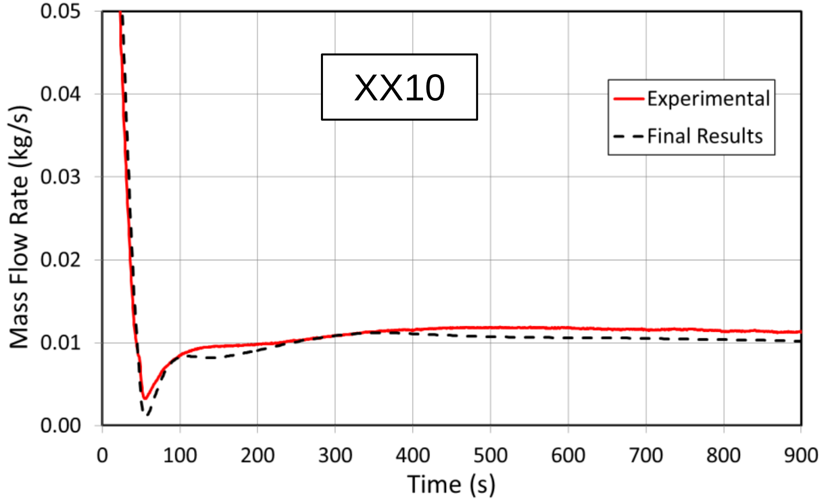




Analysis of the Transient Results

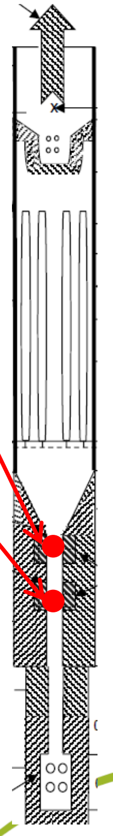
- Instrumented subassemblies mass flow rate
  - Due to the lack of geometrical details of the SA inlet nozzles, it is difficult to match the experimental MFR of the instrumented SAs.
  - In XX09 it remains at higher value throughout all the transient
  - In XX10 it reaches a lower value at the end of the pumps coastdown, and then it stabilizes at about the correct value, when natural circulation is established

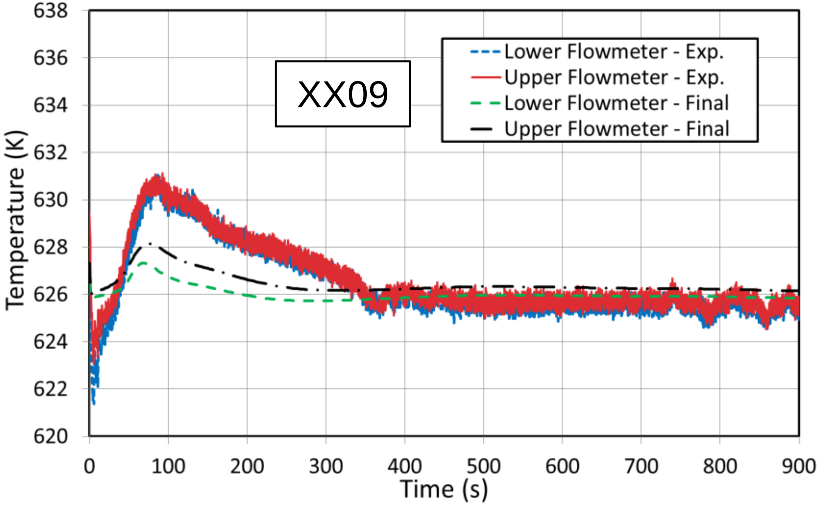


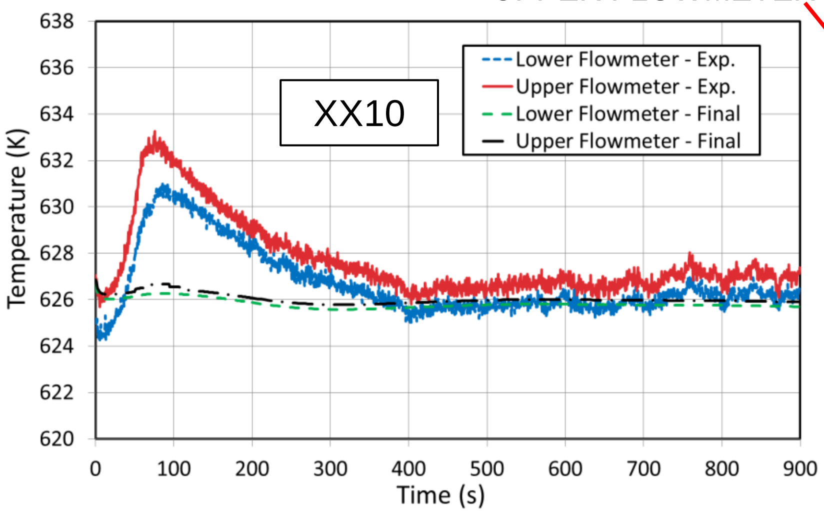


Analysis of the Transient Results

- Lower and upper flowmeters temperatures
  - Accounting for the g-heating below the BAF improves the accuracy of predictions of temperature time trends
  - In XX10 subassembly, where the MFR is small, the thermal inertia and g-heating affect more the temperatures during the first part of the transient

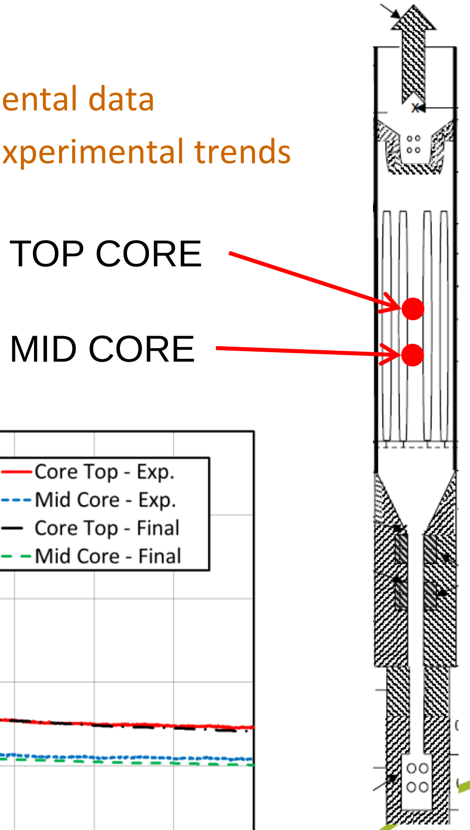


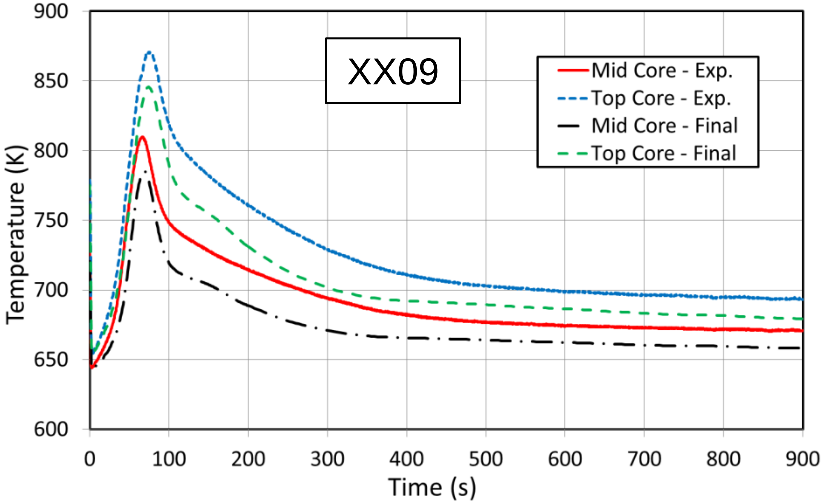


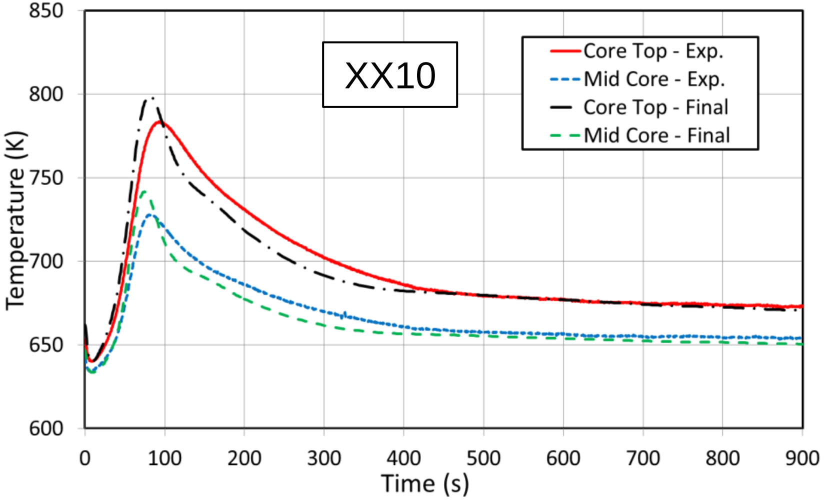


Analysis of the Transient Results

- Cladding temperatures at middle and top of the core
  - The temperature time trends are qualitatively in good agreement with the experimental data
  - The small discrepancies are due to the mass flow rate, not perfectly matching the experimental trends

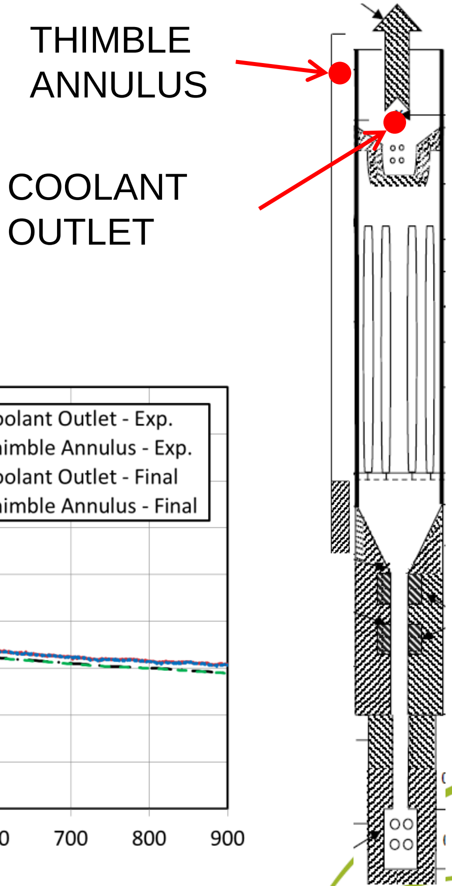


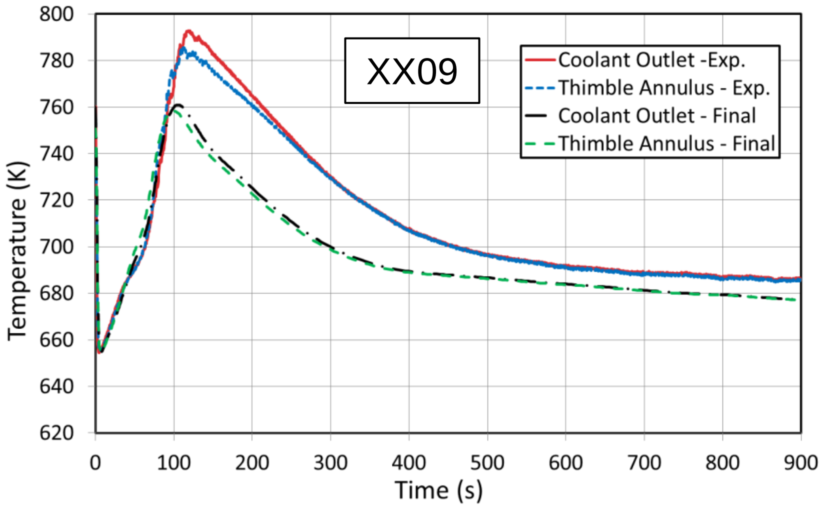


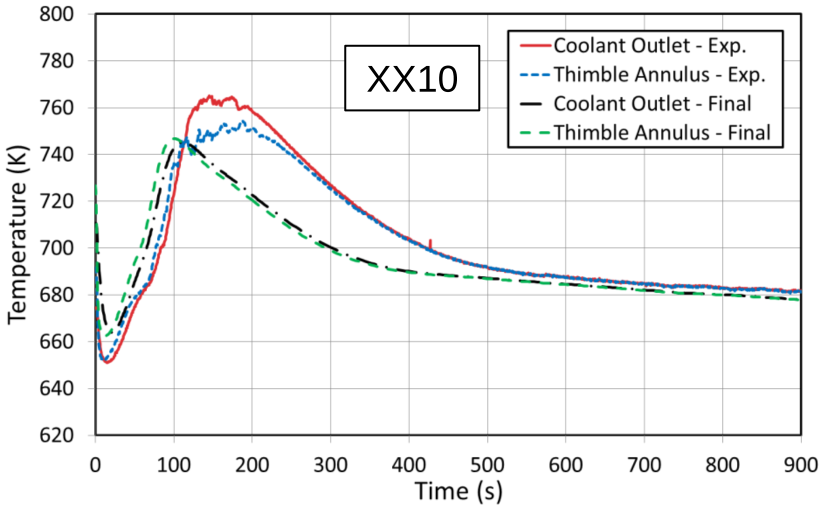


Analysis of the Transient Results

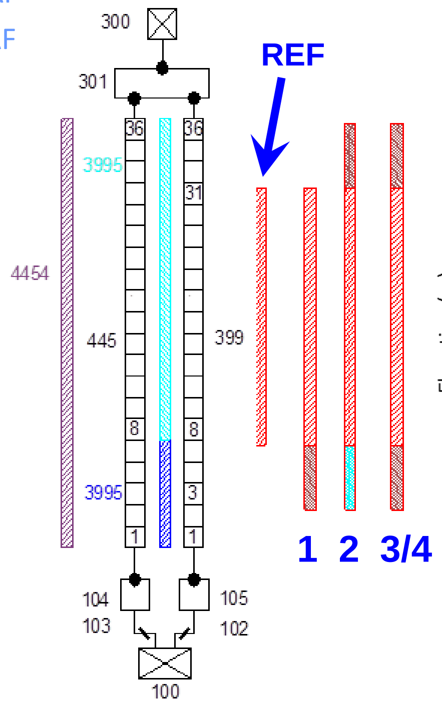
- Coolant outlet and thimble annulus temperatures
  - The calculated temperatures above the TAF in XX10 start to increase slightly before the experimental data
  - In both subassemblies, the thermal inertia and the γ-heating seem to affect more the temperature time trends above the TAF

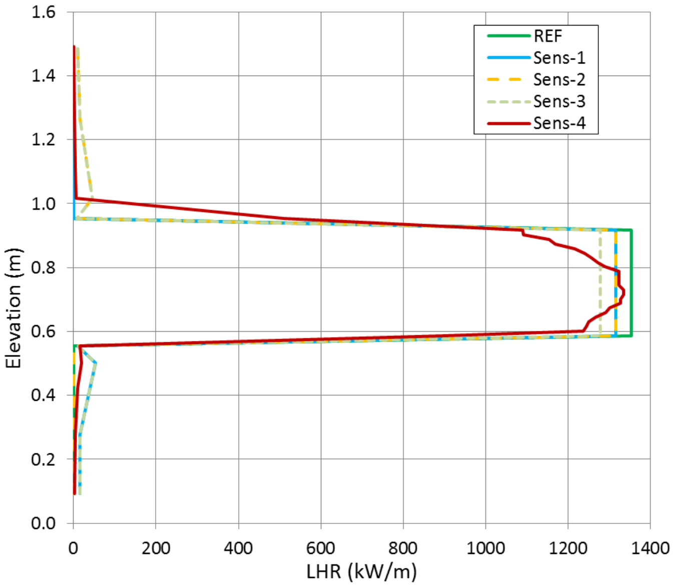




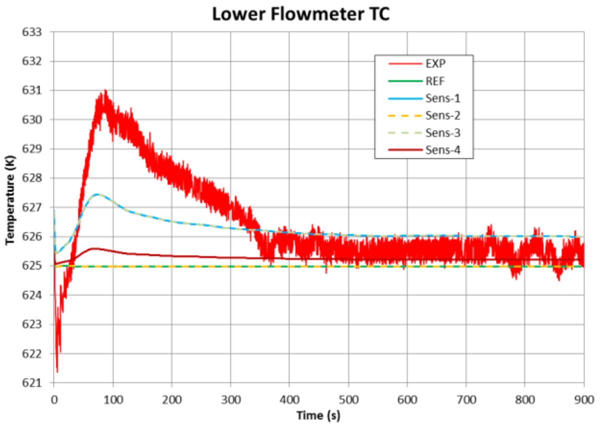


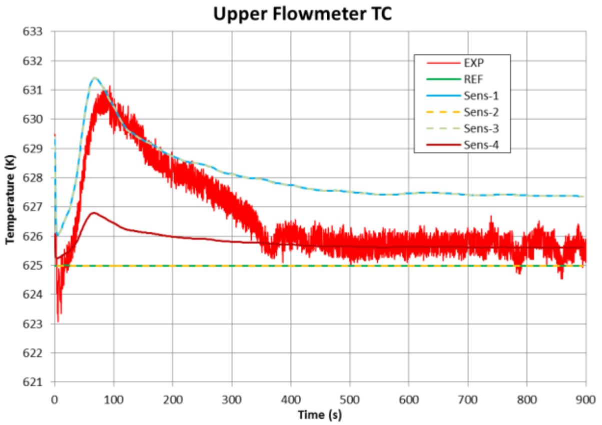
- Sensitivity on gamma heating and axial power distribution
  - Subassembly XX09 (BIC imposed and isolated thimble walls)
    - ✓ Ref.: open calculation phase, before last improvements
    - ✓ 1: γ-heating considered only below the BAF
    - ✓ 2: γ-heating considered only above the TAF
    - ✓ 3: γ-heating considered both below and above the active part of the fuel
    - ✓ 4: axial power distribution taken from the SHRT-45R
  - γ-heating calculated to match the SS temperature values at the lower and upper flowmeters (below the BAF)
  - The same distribution applied above the TAF (symmetrically)

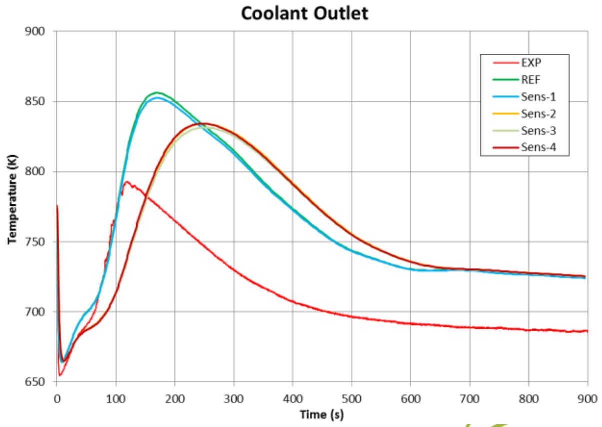




- Results
  - The power supplied below the BAF (cases #1, 3, 4) positively affects the temperature trends in the lower and upper flowmeter thermocouples
  - The power supplied above the TAF (cases #2, 3, 4) produce a small delay in the coolant outlet temperature increase after the pump coastdown
  - The higher time trends of the coolant outlet temperature are due to the fact that the outer boundary condition of the heat structure simulating the guide thimble walls is isolated
  - The cladding temperatures are not really affected by the power supplied outside the active part

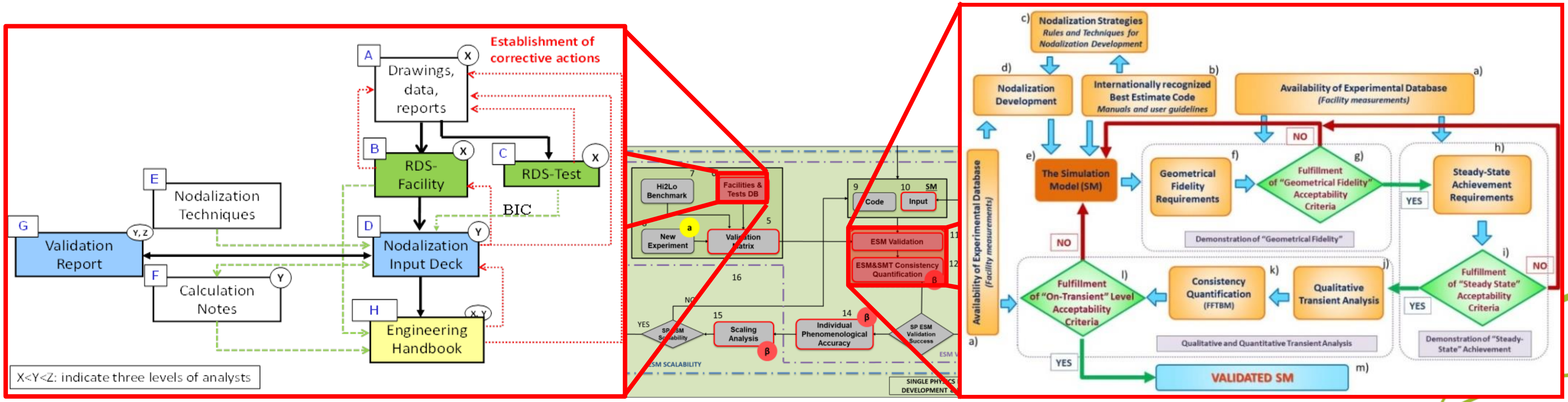






NEMM Validation Procedures

- The SM development and validation process is based on a systematic and comprehensive comparison between experimental data and simulation results, following strict procedures embedded in NEMM (NINE Evaluation Model Methodology)
  - The creation of a Database of Facilities and Tests
    - ✓ SCCRED (Standardized and Consolidated Calculated and Reference Experimental Database) Methodology
  - The development of the Simulation Model
    - ✓ NINE Simulation Model techniques
  - The Validation of the Simulation Model
    - ✓ **NEMM Validation Procedures**
      - **Demonstration of the Geometrical Fidelity**
      - **Demonstration of the Steady State Achievement**
      - **Transient Analysis**



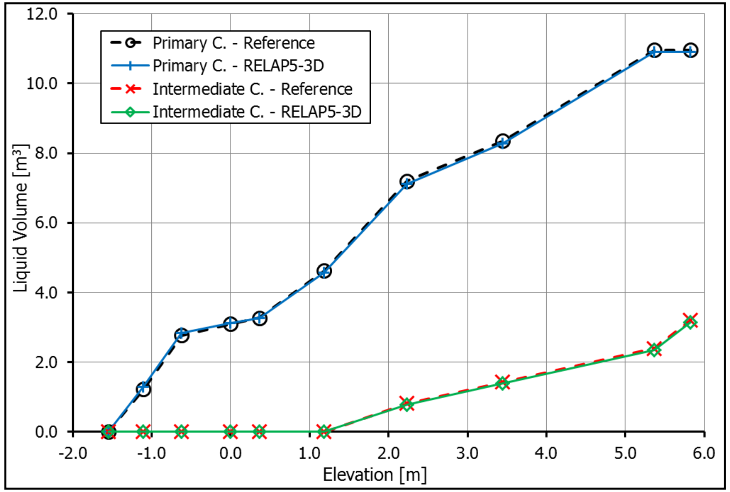
Validation Process Applied to EBR-II

- Procedure already applied in several OECD-NEA benchmarks/projects
  - OECD BEMUSE, OECD PKL, OECD ATLAS, IAEA EBR
  - Several ITF Tests
- First application to non-LWR!
- "Reduced version" of the procedure adopted for the benchmark
  - The "on-transient" validation is a very complex step requiring several different phases
  - It includes that the qualitative accuracy evaluation must be completed before any meaningful attempt to perform the quantitative evaluation
  - In this case, the focus (for practical reasons) is only on the **Quantitative Accuracy Evaluation**, which is performed by the **FFTBM**
- Main goal: to support the interpretation of the results calculated by the participants
  - to provide quantitative measures of the discrepancies between participants assumptions and reference specification data to support the understanding of the reasons of the differences between the participants' results and the experimental data
- **The applied process has no the objective to provide a ranking between participants**

Geometrical Fidelity

- Overall 54 parameters requested
  - 29 related to Hydraulic Volumes
  - 25 for Heat Structures
    - ✓ The lack of accurate geometrical data in some part of the reactor causes higher error in some parameters

Liquid volume versus elevation curves

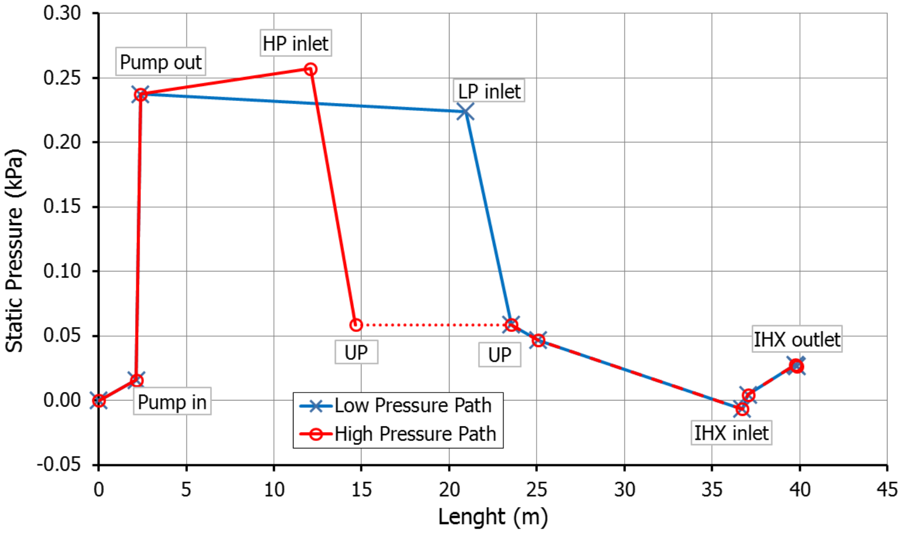


#	Parameters	Unit	Reference	RELAP5-3D	Error (%)
	Hydraulic Volumes				
1	Pool liquid volume	m^3	329.724	330.892	-0.35
2	Primary circuit liquid volume (without pool)	m^3	10.9630	10.8977	0.60
3	High pressure Inlet Plenum liquid volume	m^3	0.7212	0.6934	3.86
4	Low pressure Inlet Plenum liquid volume	m^3	1.1309	1.1581	-2.41
5	Liquid volume in core, Drivers	m^3	0.0793	0.0784	1.07
6	Liquid volume in core, Partial Drivers	m^3	0.0239	0.0238	0.44
7	Liquid volume in Core, High Flow Drivers	m^3	0.0410	0.0404	1.41
8	Liquid volume in Core, Experimental Subassemblies	m^3	0.0306	0.0309	-1.06
9	Liquid volume in Core, Instrumented (XX09 and XX10)	m^3	0.0064	0.0065	-0.90
10	Liquid volume in Core, Steel/Dummy Subassemblies	m^3	0.0118	0.0115	2.69
11	Liquid volume in Core, Reflector Subassemblies	m^3	0.2008	0.1977	1.55
12	Liquid volume in Core, Blanket Subassemblies	m^3	0.3449	0.3410	1.12
13	Upper plenum liquid volume	m^3	2.8882	2.7771	3.85
14	Z pipe liquid volume	m^3	1.0755	1.0762	-0.07
15	IHX primary side liquid volume	m^3	1.9957	1.9957	0.00
	Heat structures				
	Heat structure volume in Core (D, HFD, PD):				
16	- Stainless Steel 316	m^3	0.03926	0.03926	0.00
17	- Stainless Steel 304	m^3	0.12306	0.12306	0.00
18	- Fuel Alloy, U-5Fs	m^3	0.01528	0.01528	-0.02
19	- Plenum Gas (Helium)	m^3	0.01241	0.01241	0.02
20	- Stagnant Sodium	m^3	0.00612	0.00612	0.05
21	IHX volume (tubes and intermediate inlet pipe)	m^3	0.56125	0.56125	0.00
	Z-pipe heat structure volume:				
22	- Stainless Steel 304	m^3	0.18598	0.18610	-0.07
23	- Stagnant Sodium	m^3	0.65093	0.65136	-0.07

Steady State Achievement

- 37 parameters requested to demonstrate the SS achievement
  - Pressures, temperatures, flow rates, etc.

Static pressure VS. length curve (both high-p and low-p path)



#	Parameter	Unit	Reference	RELAP5	Error (%)
1	Primary circuit power balance	MW	57.29	57.29	0.00
2	Pump 1 velocity	rpm	799.03	792.05	0.87
3	Pump 2 velocity	rpm	764.92	792.05	-3.55
4	Pump 1 volumetric flow, @800^{0}F (4352.6 gpm)	m^3/s	0.275	0.2708	1.52
5	Pump 2 volumetric flow, @800^{0}F (4346.5 gpm)	m^3/s	0.274	0.2727	0.49
6	Core outlet volumetric flow, @800^{0}F (8500 gpm)	m^3/s	0.5363	0.5487	-2.32
7	Z pipe volumetric flow, @800^{0}F (8438.5 gpm)	m^3/s	0.532	0.5452	-2.48
8	IHX inter. side volumetric flow, @582^{0}F (5614.6 gpm)	m^3/s	0.354	0.3567	-0.77
9	Low pressure inlet plenum temperature	K	623.7	625.4	-0.28
10	High pressure inlet plenum temperature	K	624.7	625.7	-0.16
11	Average core inlet temperature	K	624.56	625.6	-0.17
12	IHX primary inlet temperature	K	710	724.0	-1.98
13	IHX primary outlet temperature	K	628.5	628.4	0.01
14	IHX intermediate inlet temperature	K	574.15	574.1	0.00
15	IHX intermediate outlet temperature	K	714.2	724.3	-1.41

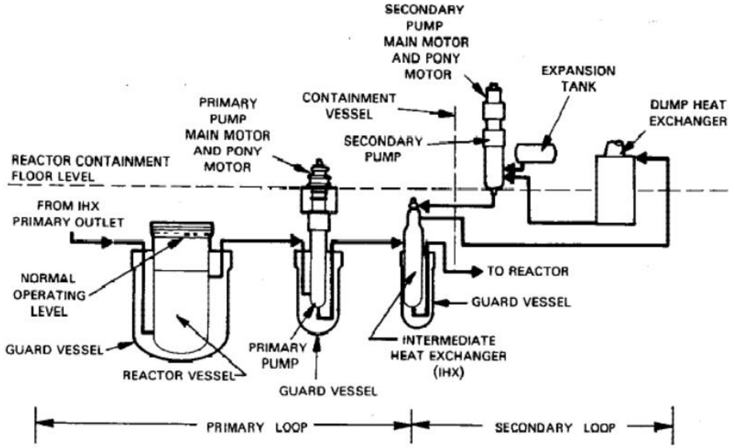
Quantitative Accuracy Evaluation

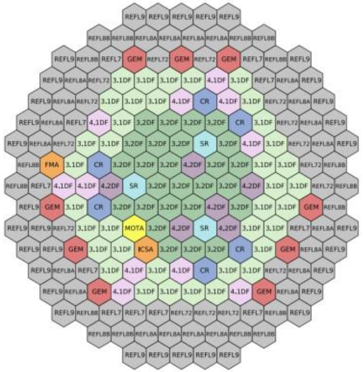
- 26 parameters (among 50 requested) used to perform the FFTBM
  - Only the parameters for which the experimental trends are available
  - Upper plenum temperature: compared with 8 experimental time trends
  - IHX primary side outlet temperature: compared with 4 experimental time trends
- Totally 36 time trends used for the FFTBM
- Higher AA values for:
  - Subassemblies mass flow rate, due to the limited information on the inlet flow holes
  - IHX Primary temperatures, due to particular geometry of the IHX itself

Parameters	AA value
Pump #2 Mass Flow Rate	0.0866
XX09 Subassembly Mass Flow Rate	0.178
XX10 Subassembly Mass Flow Rate	0.2298
Low Pressure Inlet Plenum Temperature	0.0511
High Pressure Inlet Plenum Temperature	0.0495
Upper Plenum Temperature (min/max)	0.0676/0.0925
IHX Primary Inlet Temperature	0.1291
IHX Primary Outlet Temperature (min/max)	0.2736/0.293
IHX Intermediate Outlet Temperature	0.0714
XX09 Flowmeter Temperature	0.0236
XX09 Mid-Core Temperature	0.0849
XX09 Top of Core Temperature	0.0766
XX09 Above Core Temperature	0.1404
XX09 Core Outlet Temperature	0.1214
XX10 Flowmeter Temperature	0.0202
XX10 Mid-Core Temperature	0.0575
XX10 Top of Core Temperature	0.0679
XX10 Above Core Temperature	0.1163
XX10 Core Outlet Temperature	0.148

FFTF Benchmark

- **IAEA CRP** on *“Benchmark Analysis of FFTF Loss Of Flow Without Scram Test”*
  - Blind and open phase exercises
- Fast Flux Test Facility (FFTF)
  - 400 MWth loop-type SFR with MOX fuel with 3 primary loops and 3 secondary loops
    - ✓ 3 intermediate heat exchangers (IHX)
  - 12 air Dump Heat Exchangers (DHX) in secondary loops as ultimate heat sink
- FFTF reactor core
  - Core height/diameter: 91.44/120 cm
  - 199 hexagonal assemblies
    - ✓ 91 core locations (inner region)
      - 82 Fuel and Test locations
        - 2 Proximity Instrumented Open Test Assembly (PIOTA)
      - 6 Control Rods and 3 Safety Rods
    - ✓ 108 Inconel Reflector and Test Locations (outer region)
  - 16 assembly flow zones based on hydraulic testing

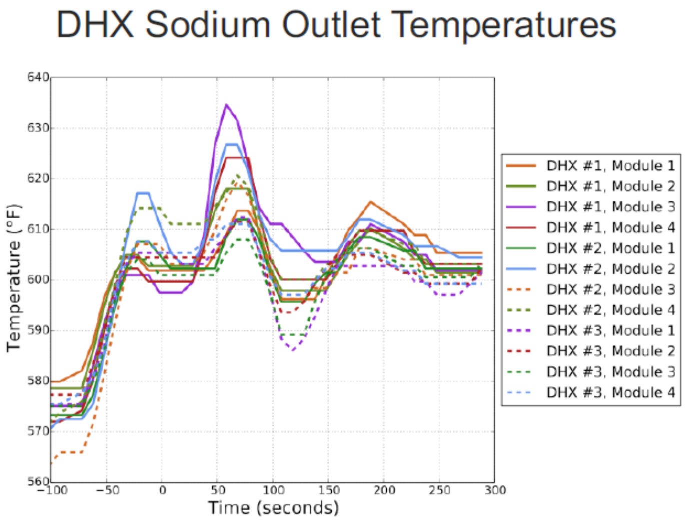




FFTF Cycle 8C Core Loading

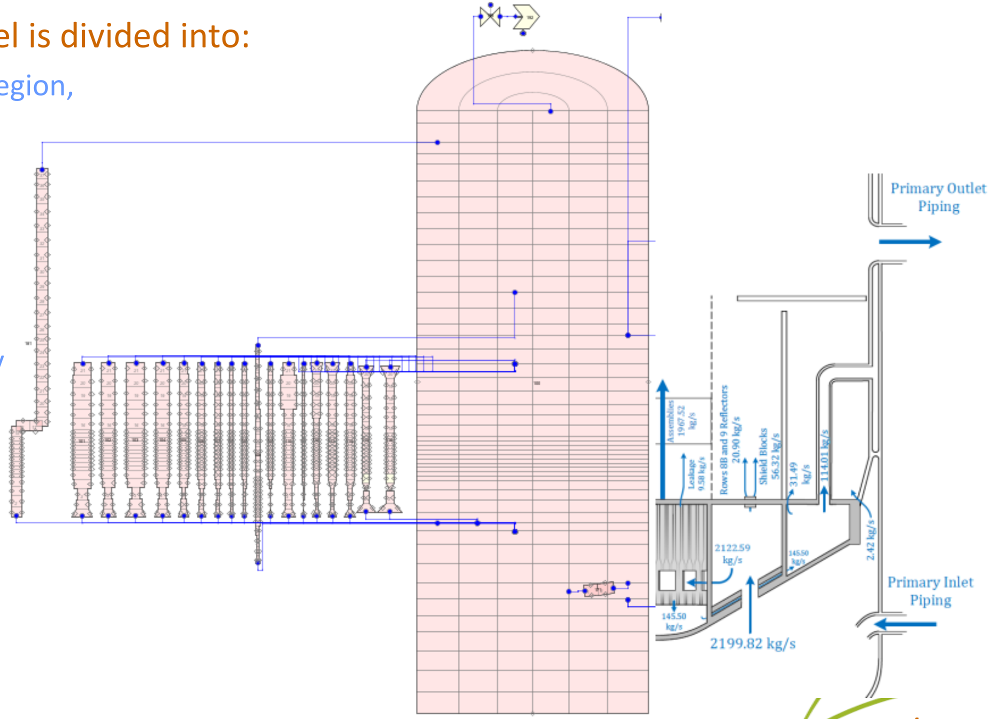
LOFWOS Test #13 Scenario

- Part of FFTF Passive Safety Test Program
  - For verification of decay heat removal through natural circulation
  - Demonstration of GEMs and inherent core reactivity feedback mechanisms ability to take the core subcritical with a modest peak coolant temperature
- LOFWOS: Loss Of Flow Without Scram
  - Unprotected Loss Of Flow (ULOF)
  - Starting from 50% power and 100% flow
  - Normal control rod scram response disabled
  - All of the Primary Main Coolant Pumps tripped
    - ✓ Primary pump coast down simulated
  - Secondary pumps remain operational for the entire test
  - DHX fan speed reduced 200 seconds before start of transient

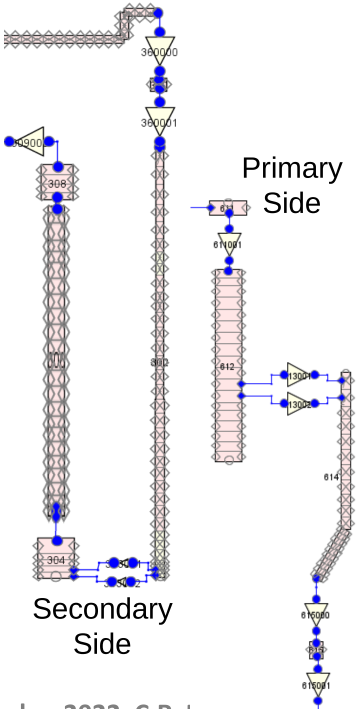


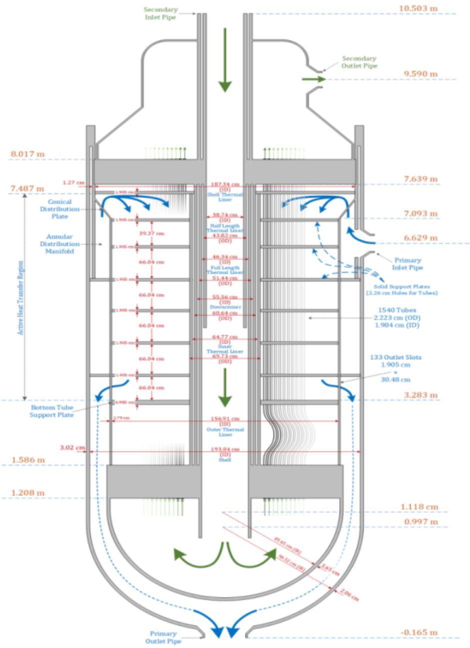
Simulation Models

- Two simulation models developed for R5-3D and TRACE
  - Same nodalization scheme adopted
- Reactor vessel modelling
  - From vessel centre to peripheral, the reactor vessel is divided into:
    - ✓ 3 radial nodes: core basket region, annular plenum region, peripheral plenum & in-vessel storage region
    - ✓ 48 axial nodes
  - The flow through the reactor core is modelled using 18 PIPE components
    - ✓ 16 PIPEs simulating the sixteen assembly flow zones
    - ✓ 2 PIPEs simulating the 2 PIOTA assemblies separately
    - ✓ Flat power imposed for all the assemblies
  - Two bypass flow paths
    - ✓ 1 PIPE connecting low-pressure plenum and peripheral plenum
    - ✓ 1 PIPE to simulate the flow path from the low-pressure plenum up to outlet plenum, passing through three 6 inches bypass pipe



- Primary and secondary loop modelling
  - Three primary and secondary loops are simulated separately with all the components
    - ✓ The pump homologous curves and coastdown velocity curves are provided in the FFTF specification
  - The DHXs are not modelled, boundary conditions imposed
    - ✓ SS cold leg inlet (DHX outlet): temperature
    - ✓ SS hot leg outlet (DHX inlet): pressure
  - Improvements from blind phase
    - ✓ **HS used to model Resistance Temperature Detectors (RTD)**
      - 5-cm ø cylindrical HS of SS-304





- IHX modelling
  - Primary side modelled with 4 PIPEs
  - Secondary side modelled with 5 PIPEs
  - Improvements from blind phase
    - ✓ The heat transfer at the IHX has been enhanced
      - Fouling Factor increased up to F = 4.0
      - The correlation used by RELAP with liquid metals in bundle geometry is reported "**to under-predict Nusselt numbers when P/D exceeded 1.2**" (P/D = 1.3072)

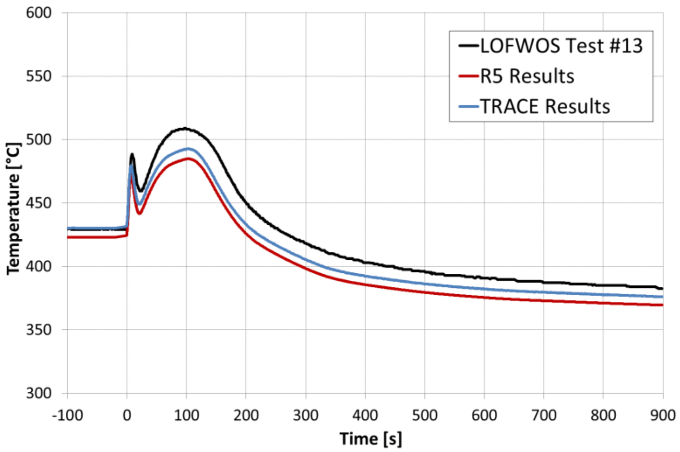
- Demonstration of steady-state achievement
  - The reactor power is provided by the benchmark team
  - The mass flow rates through different regions of reactor vessel are compared
    - ✓ Maximum discrepancy is found to be about 2.0%
  - The initial flow rates for each of the sixteen assembly flow zones are provided in the specification
    - ✓ The discrepancy is found to be lower than 2% except for 2 flow zones

Parameter	Units	LOFWOS Test #13	R5-3D	TRACE	Discrepancy (%)	
					R5-3D	TRACE
Power (BC in the SM)	MWt	199.2	199.2	199.2	0.00%	0.00%
Core Inlet Temperature	°C	317.2	317.1	323.9	0.02%	2.11%
Flow Through All Assemblies	kg/s	1988.4	1986.0	2003.8	0.12%	0.77%
Shield Flow Rate	kg/s	56.3	56.4	57.6	0.24%	2.36%
Leakage and Bypass Flow Rates	kg/s	157.5	156.4	158.5	0.71%	0.68%
Primary Loops Flow Rate	kg/s	736.9	736.8	744.2	0.02%	0.99%
		735.7	735.0	742.3	0.09%	0.90%
		729.6	727.0	734.1	0.36%	0.61%

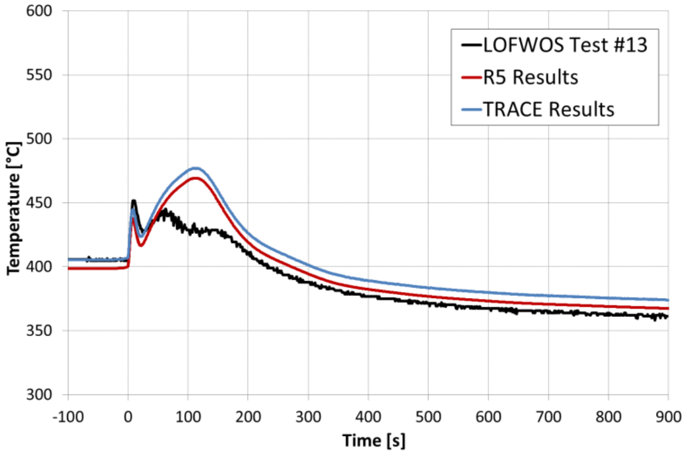
Parameter	Mass Flow Rate (kg/s)			Discrepancy (%)	
	LOFWOS Test #13	R5-3D	TRACE	R5-3D	TRACE
Flow Zone 1	558.60	560.92	554.61	0.41%	0.71%
Flow Zone 2	289.72	289.94	295.07	0.08%	1.85%
Flow Zone 3	458.08	457.66	464.12	0.09%	1.32%
Flow Zone 4	304.66	302.36	309.01	0.75%	1.43%
Flow Zone 5	99.84	98.18	101.41	1.66%	1.57%
Flow Zone 6(1)	74.45	74.93	75.32	0.65%	1.18%
Row 2 PIOTA	24.82	24.96	25.21	0.60%	1.58%
Flow Zone 7(2)	41.00	40.58	40.82	1.03%	0.45%
Row 6 PIOTA	20.50	20.29	20.50	1.03%	0.00%
Flow Zone 8	1.787	1.791	1.91	0.20%	6.82%
Flow Zone 9	2.043	2.049	2.05	0.28%	0.34%
Flow Zone 10	40.91	40.73	40.43	0.44%	1.18%
Flow Zone 11	0.650	0.645	0.66	0.79%	1.68%
Flow Zone 12	19.65	19.46	20.40	0.98%	3.80%
Flow Zone 13	13.36	13.45	13.46	0.70%	0.77%
Flow Zone 14	17.46	17.19	17.59	1.55%	0.77%
Flow Zone 15	9.525	9.458	9.68	0.70%	1.58%
Flow Zone 16	11.38	11.40	11.56	0.18%	1.60%

Simulation Results

- Transient PIOTA outlet temperature at Row 2 and Row 6
  - Initial rapid increase of temperature is caused by the increase of power-to-flow ratio following the pump trip
  - The power start to decrease faster than primary mass flow rate due to TH feedback
    - ✓ The temperature starts to decrease
  - TH negative feedback effect start to slow down
    - ✓ Temperature starts to increase again
  - Natural circulation is established and PIOTA outlet temperature start to decrease

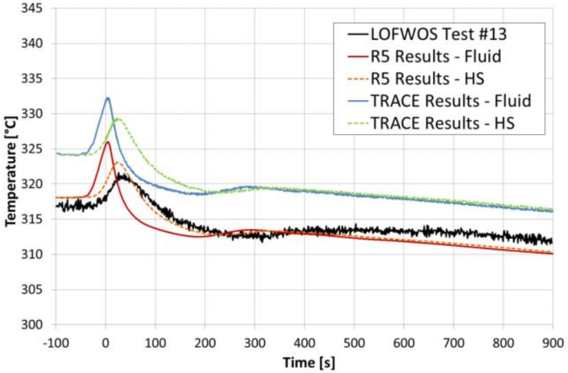


Row2 PIOTA Outlet Temperature

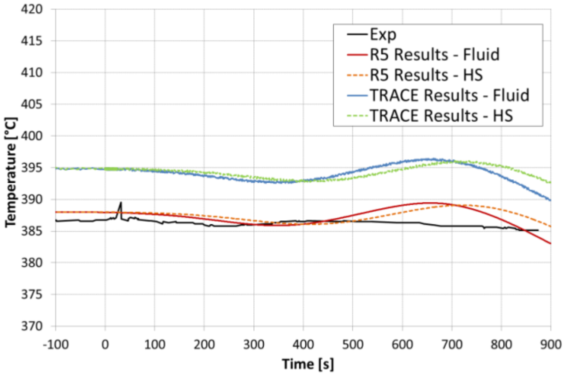


Row6 PIOTA Outlet Temperature

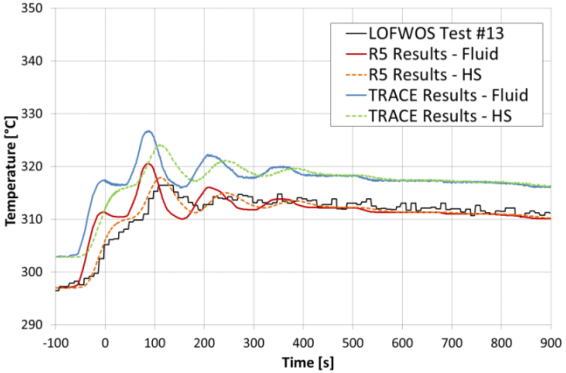
- Primary side temperatures
  - CL trends showed a faster rise, reaching a higher peak value
    - ✓ Oscillations occurred about 30 s earlier
  - HL oscillation may be due to sodium mixing and thermal stratification phenomena
    - ✓ Difficult to simulate using SYS-TH code
- Secondary side temperatures
  - CL temp followed the time trends of the DHX sodium outlet temp
  - HL temp decreased faster at the beginning of the transient
    - ✓ Oscillations occur earlier
- Better prediction with modelling of RTD thermal inertia



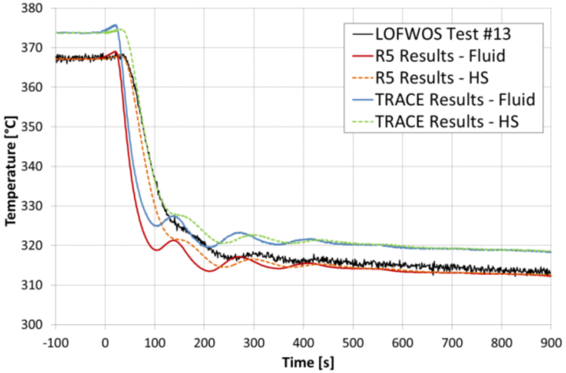
Primary side cold leg temperature



Primary side hot leg temperature



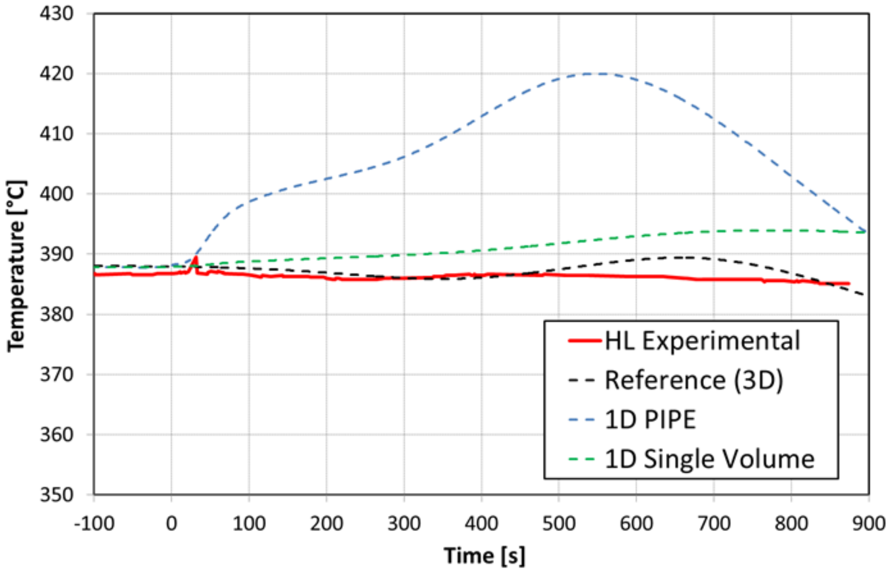
Secondary side cold leg temperature



Secondary side hot leg temperature

Sensitivity Analysis

- Sensitivity analysis on outlet plenum nodalization (R5-3D)
  - To study the effect of the modeling choice made for the reactor vessel Outlet Plenum (OP) and its impact on sodium mixing and thermal stratification phenomena
    - ✓ Reference (3D): OP modelled with a cylindrical multi-dimensional component
      - Thermal stratification with 2 main mixing flow paths
        - Hot leg connection is at the bottom of the outlet plenum, just above the core outlet
      - Axial flow path: part of the hot sodium reaches the top of the reactor through the central zone and then recirculates downwards from the lateral region
      - Radial flow path: part of the hot sodium exiting from the core flows directly towards the hot legs
    - ✓ 1D PIPE: OP modelled with vertical pipe component
      - No thermal stratification occurs
      - Hot sodium exiting the core flows directly towards the hot legs
    - ✓ 1D Single Volume: OP modelled with a 1D single-volume
      - Hot sodium exiting from the core is completely mixes with all the sodium in the outlet plenum



Outcomes from EBR-II Benchmark

- Natural phenomena, such as expansion of the sodium coolant and thermal inertia of the primary sodium pool, can be effective in successful cooling of the EBR-II type reactor
  - Some parameters are in good agreement with the experimental data (i.e. pumps flow , inlet plena temperature)
  - Others parameters still have some margins for improvement (e.g. Z-pipe and IHX primary side inlet temperature)
- Several open issues remaining at the end of the benchmark
  - The effect of thermal inertia and γ-heating during the transient
  - The possible thermal stratification in the upper plenum and Z-pipe
  - The heat losses into the pool
  - The axial conduction in the coolant

→ **Further investigation are needed to improve the validation of simulation tools and models for the safety analysis of SFR**

- Finally, the qualification procedure has been applied
  - From the geometrical fidelity, higher errors in some parameters due to the lack of information
  - The FFTBM results show major discrepancies in those time trends related to plant region with less geometrical information available and therefore more difficult to model
    - ✓ The inlet flow holes regarding the subassemblies mass flow rate
    - ✓ The IHX primary side (baffle plate) for the coolant temperatures

It is the first time that such methodology is applied to a SFR in the framework of an international benchmark

Outcomes from FFTF Benchmark

- The RELAP5 SM results shows good agreement with the experimental results
  - Applicability of NEMM methodology in simulating SFR designs confirmed
  - TRACE SM requires further improvement
    - ✓ Heat transfer diameter of each junction to compute the heat transfer coefficient instead of using a same lump heat transfer diameter for the entire heat structure
- Open issues
  - Sodium mixing and the thermal stratification phenomena in the outlet plenum
    - ✓ Sensitive to the nodalization scheme adopted
  - Estimation of the core pressure drop and the flow distribution during the transients
  - Need for an accurate modelling of the mixing of coolant flows in the assembly
  - Need for a proper approach for the simulation of the heat transfer between adjacent subassemblies

Conclusions

- The paper summarizes the activity carried-out, presents the results and discusses the main outcomes of the mentioned benchmark exercises
- The following items have been identified as meriting particular attention in the future
  - The need for an accurate modelling of the mixing of coolant flows in the assembly
  - The estimation of the pressure drop and the flow distribution during the transients
  - The sodium mixing and the thermal stratification phenomena which play a crucial role during the transients
    - ✓ Also sensitive to the nodalization scheme adopted and cannot be accurately predicted by the current existing SYS-TH codes
  - The need for a correct and comprehensive simulation of the heat transfer between adjacent subassemblies
  - The suitability for improvement of the calculation of inlet-plenum flow distribution
- The importance of the representativeness, exhaustiveness and comprehensiveness of the data adopted for validation of the computation tools
  - It is considered worth complementing the existing data base with the results of ad-hoc experimental programs, accurately designed, and engineered to match some specific validation needs

Thank you for your attention

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