IAEA Programme to Support Development and Validation of Advanced Design and Safety Analysis Codes

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IAEA

IAEA provides a forum for international collaboration on technology advances for LWRs and HWRs for improving economics and safety through

- **Information exchange meetings (“Technical Meetings”)**
- **Coordinated Research Projects (CRP)**
- **Collaborative Assessments (CA)**
- **International Collaborative Standard Problems (ICSP)**
- **Workshops and Courses**

Introduction

- **Coordinated Research Project (CRP)**
  - Article III of the IAEA Statute authorizes the IAEA to encourage and assist research on, and development and practical application of, atomic energy for peaceful purposes throughout the world.
  - The IAEA's CRP stimulate and coordinate the undertaking of research in selected nuclear fields by scientists in IAEA Member States.
- **International Collaborative Standard Problem (ICSP)**
  - Improve physical understanding on specific phenomena expected to occur in normal operation or transients of nuclear power plants
  - Facilitate the development and validation of computer codes for design and safety analysis of nuclear power plants
  - Includes an experiment to investigate interesting phenomena and the independent simulation of experiment with computer codes

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- Hydro-Mechanical Behaviour in Reactor Core with Plate-Type Fuel Assembly(ICSP)

Natural Circulation Phenomena and Modelling for Advanced Water Cooled Reactor(1)

- **Specific Objectives (IAEA-TECDOC-1677, 2012)**
  - establish the status of knowledge: reactor start-up & operation; passive system initiation & operation; flow stability, 3-D effects and scaling laws
  - investigate phenomena influencing reliability of passive NC systems
  - review experimental databases for the phenomena
  - examine the ability of computer codes to predict NC and related phenomena
  - apply methodologies for examining the reliability of passive systems
- **Reactor types examined in the CRP**
  - CANDU
  - CAREM
  - Advanced Heavy Water Reactor (AHWR)
  - AP-600
  - Multi-application Small Light Water Reactor (MASLWR)
  - Simplified Boiling Water Reactor (SBWR)
  - System Integrated Modular Advanced Reactor (SMART)
  - WWER-440 & 1000

Natural Circulation Phenomena and Modelling for Advanced Water Cooled Reactor(2)

- **Integral Test Facilities examined in the CRP**
  – CAPCN (CAREM)
  – ITL (AHWR)
  – ROSA/LSTF (AP-600)
  – OSUMTF (MASLWR)
  – PUMA (SBWR)
- **System thermal-hydraulic analysis codes examined in the CRP**
  – RELAP5-3D
  – RELAP5/Mod3.2
  – CATHENA
  – TASS/SMR
  – ATHLET-1.2A
  – DINAMIKA-97
  – TECH-M-97
  – KORSAR-V1

Benchmarking Severe Accident Computer Codes for HWR Applications (1)

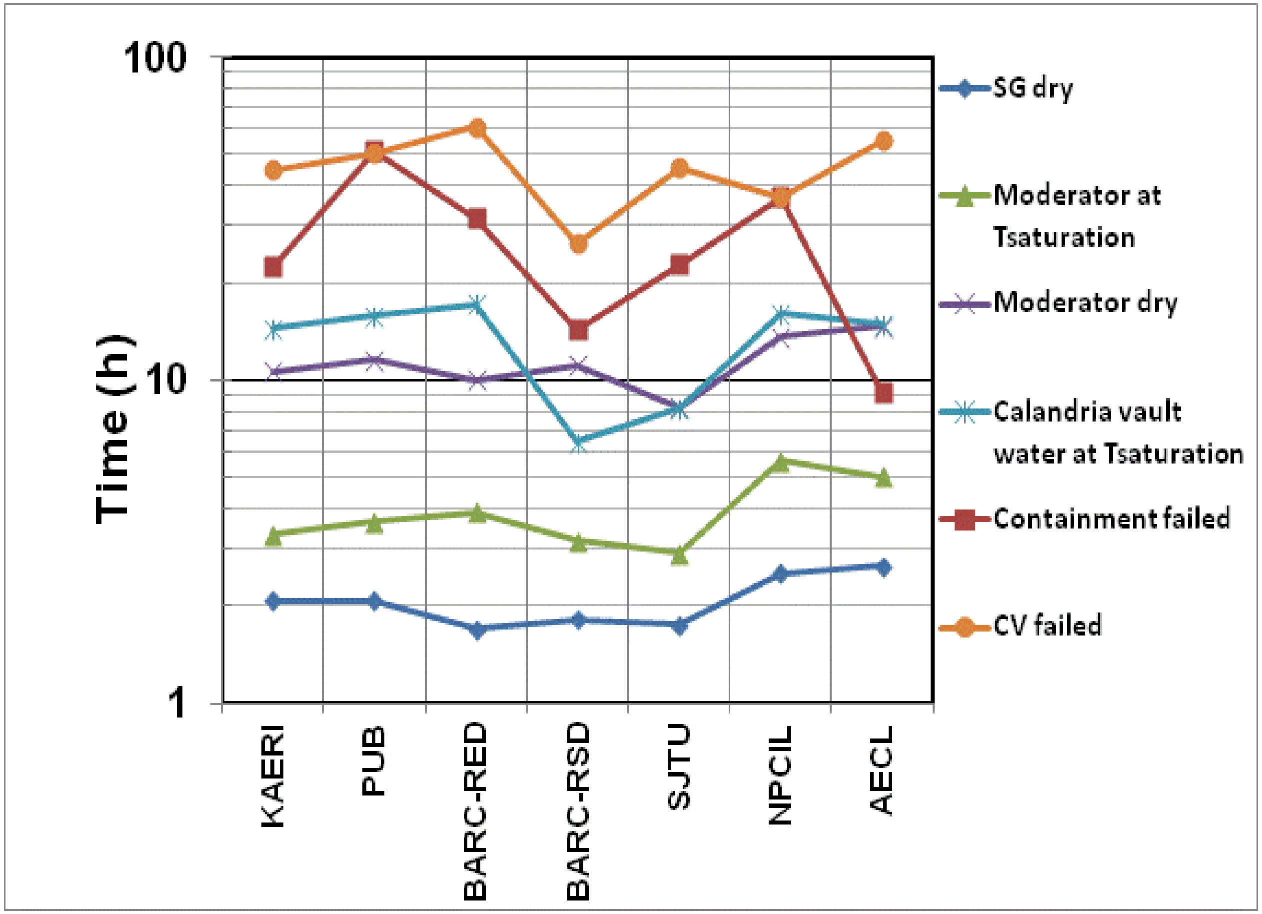
- **CRP Purpose**
  - To promote international collaboration among IAEA Member States through the benchmarking exercise to improve severe accident analysis capability for heavy water reactors
- **Expected CRP Outcomes**
  - Improved understanding of the importance of various phenomena contributing to event timing and consequences of a severe accident,
  - Advanced information on computer code capabilities for analysis of HWR severe accident
  - Improvement of emergency operating procedures or severe accident management strategies
- **Reference Design:** CANDU 6
- **Reference Scenario:** SBO

Benchmarking Severe Accident Computer Codes for HWR Applications (2)

- **Participants and Computer Codes**

Institute	Computer Code
AECL (Canada)	MAAP4-CANDU v4.0.6A
BARC-RED (India)	RELAP5 Mod 3.2, ANSWER, CAST3M, MELCOOL
BARC-RSD (India)	SCADAP/RELAP5 Mod 3.2, PHTACT, ASTEC
NPCIL (India)	ATMIKA.T, CONTACT, SEVAX, PACSR/STAR, ACTREL
KAERI (Rep. of Korea)	ISAAC 4.02
PUB (Romania)	SCDAP/RELAP5 Mod 3.4
SJTU (China)	SCDAP/RELAP5 Mod 3.4

Benchmarking Severe Accident Computer Codes for HWR Applications (3)

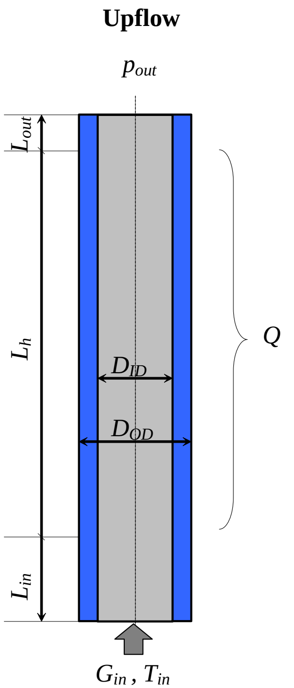


- SG secondary dryout; μ = 2.1 h, σ = 0.4 h
- Moderator reaching saturation temperature; μ = 3.9 h, σ = 1.0 h
- Moderator in the calandria vessel becoming dry; μ = 11.4 h, σ = 2.2 h
- Calandria vault water reaches saturation temperature; μ = 13.3 h, σ = 4.2 h,
- Containment failed μ = 26.9 h, σ = 14.2 h,
- Calandria vessel failure; μ = 45.7 h, σ = 11.4 h

Heat Transfer and Code Testing for Super Critical Water Cooled Reactors (1)

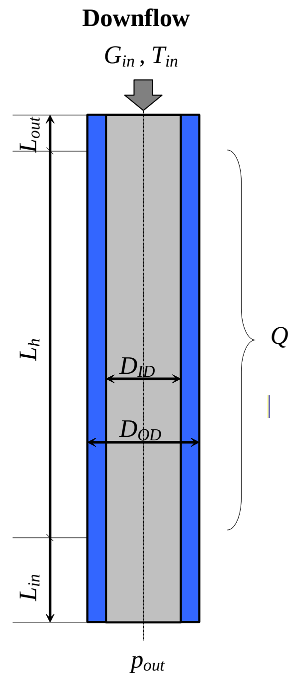
- **Specific Research Objectives:**
  - to establish a base of accurate data for heat transfer, pressure drop, blowdown, natural circulation and stability for conditions relevant to super-critical fluids,
  - to test analysis methods for SCWR thermo-hydraulic behaviour, and to identify code development needs.
- **Two Code Testing Benchmarks (CTB) were conducted**
  - Steady state flow in a heated pipe
  - Benchmark on flow stability

Heat Transfer and Code Testing for Super Critical Water Cooled Reactors (2)



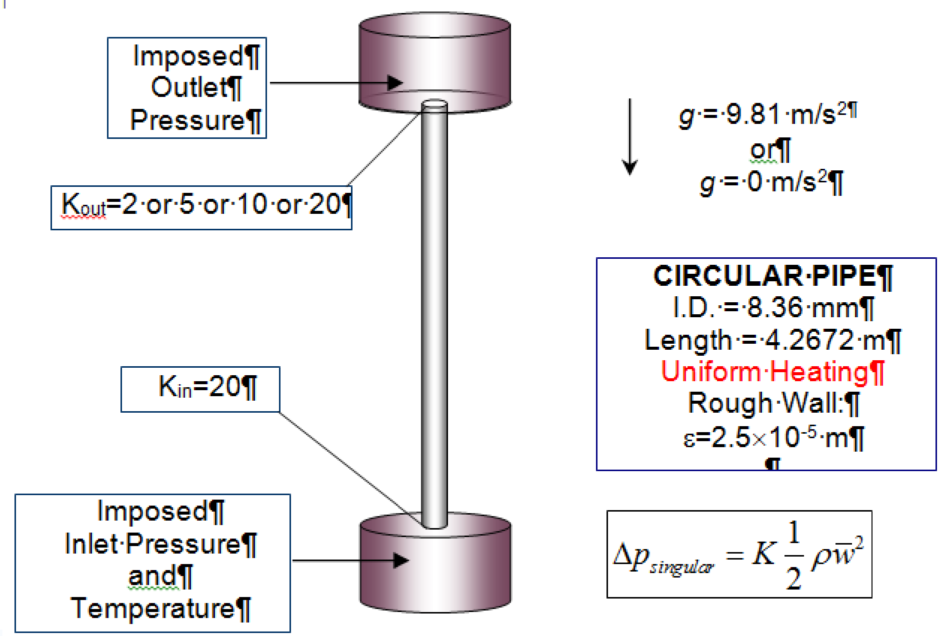
Nomenclature:

- D_{ID} – pipe inside diameter, m
- D_{OD} – pipe outside diameter, m
- G_{in} – inlet mass flow rate, kg/s
- L_{in} – length of the bottom unheated part of the pipe, m
- L_h – pipe heated length, m
- L_{out} – length of the top unheated part of the pipe, m
- p_{out} – output pressure, Pa
- Q – heat power, W
- T_{in} – inlet temperature, ºC

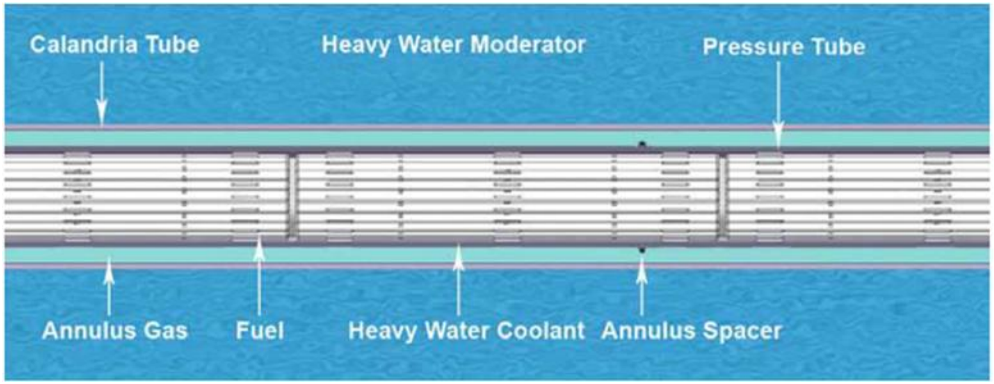


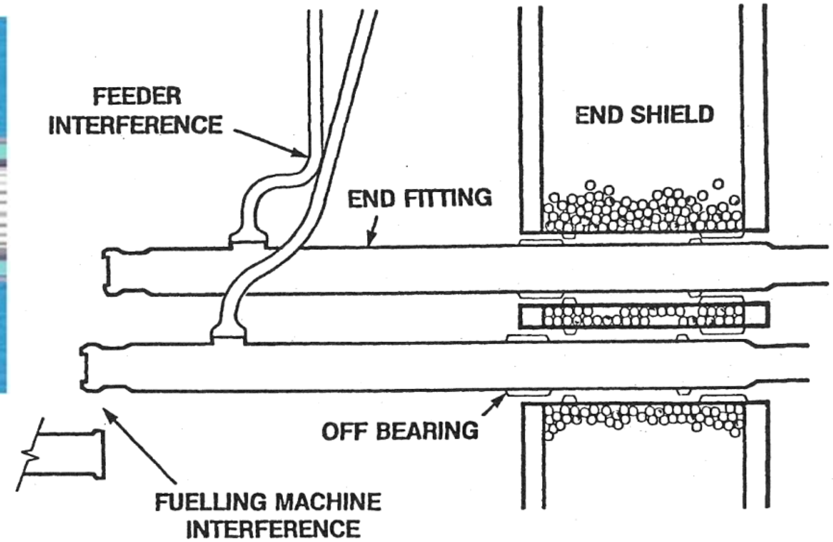
Steady state flow in a heated pipe

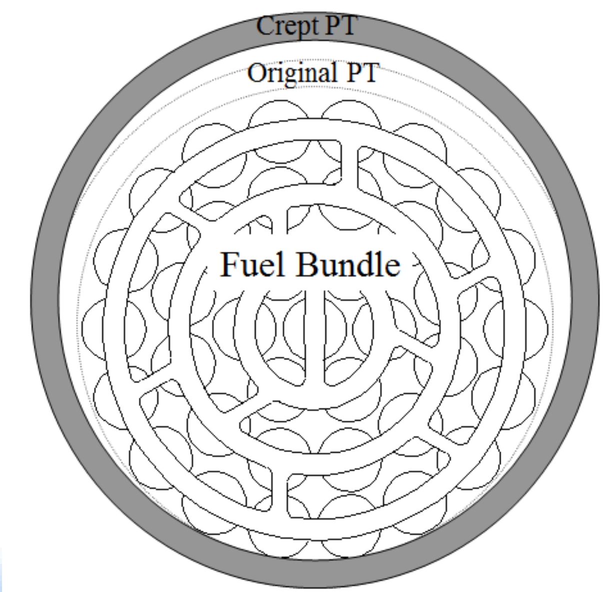
Benchmark on flow stability

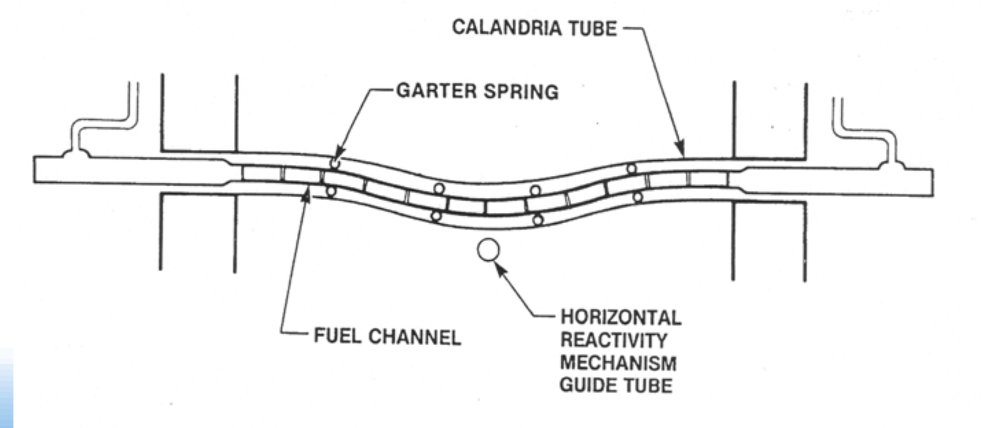


Diametral and Longitudinal Creep in Pressure Tubes (1)

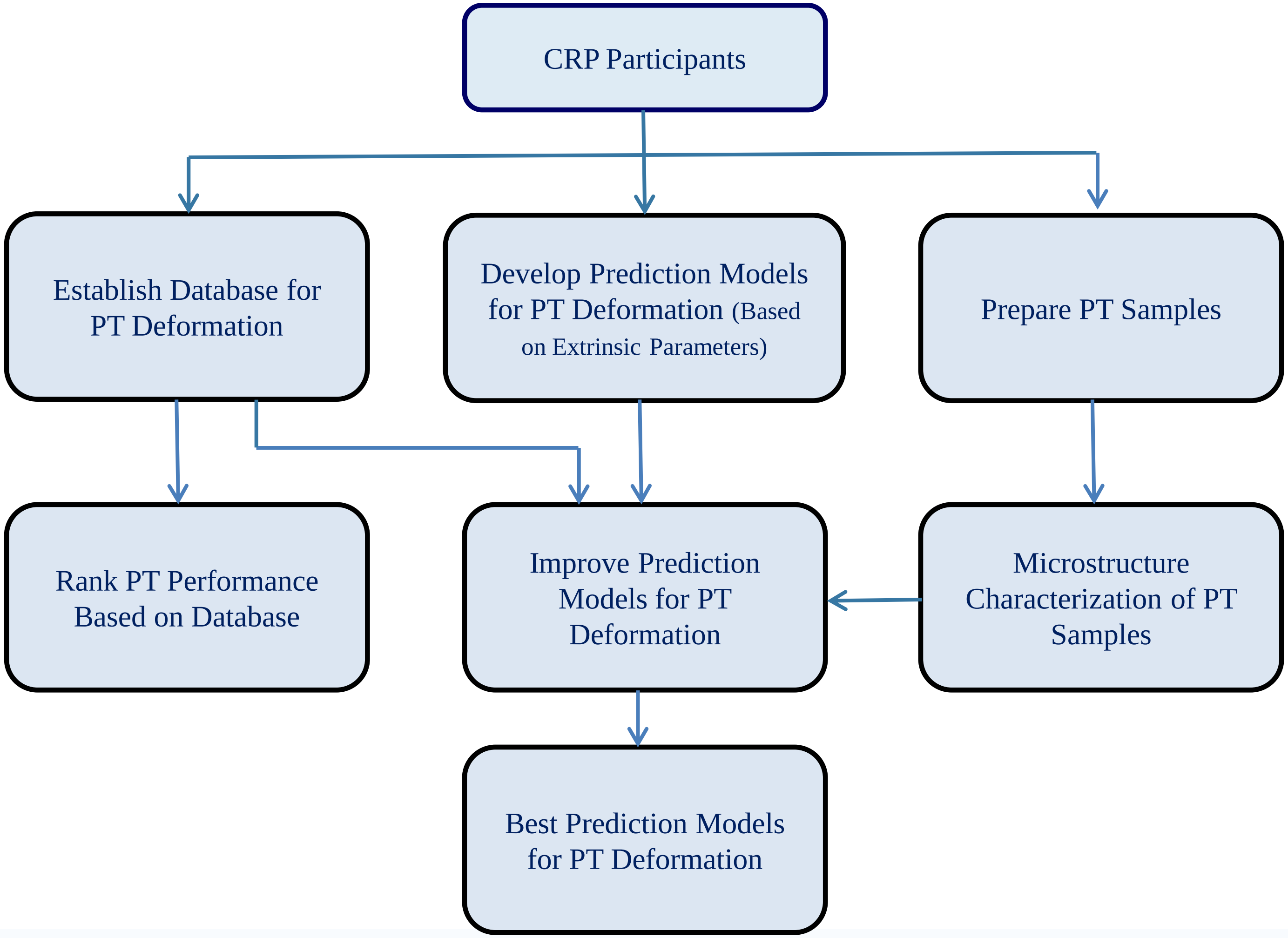




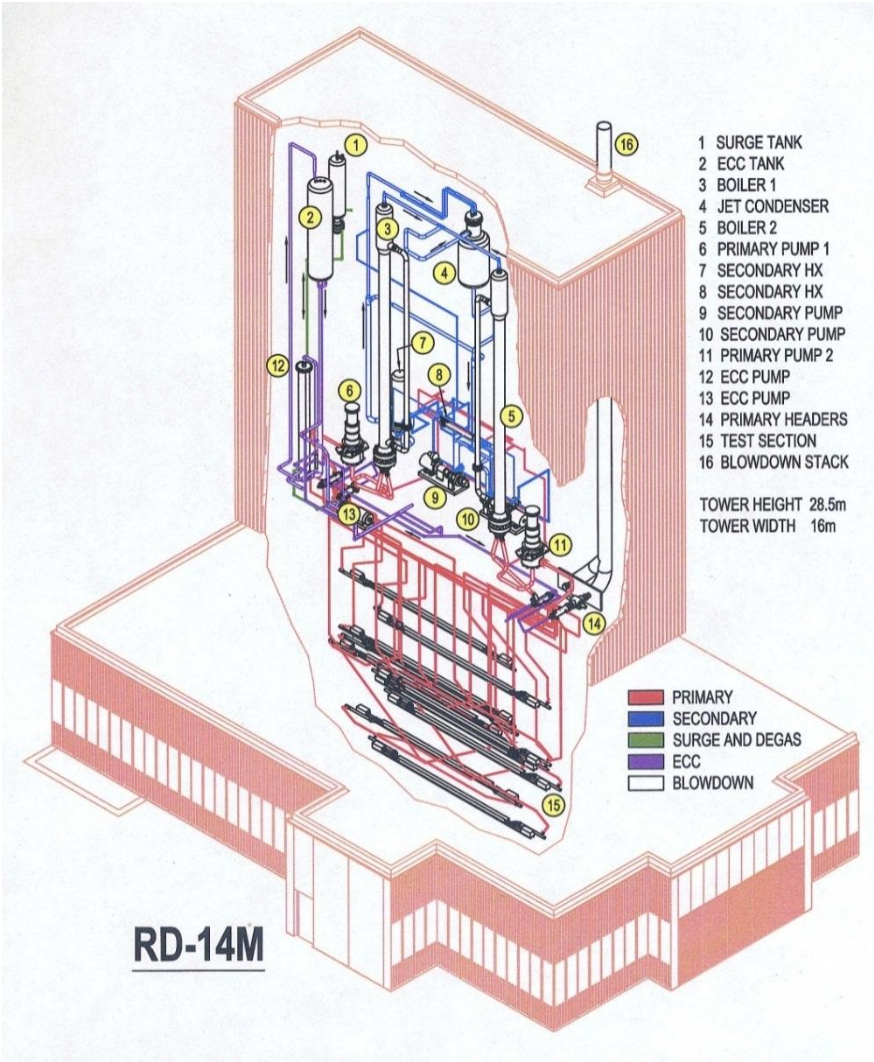




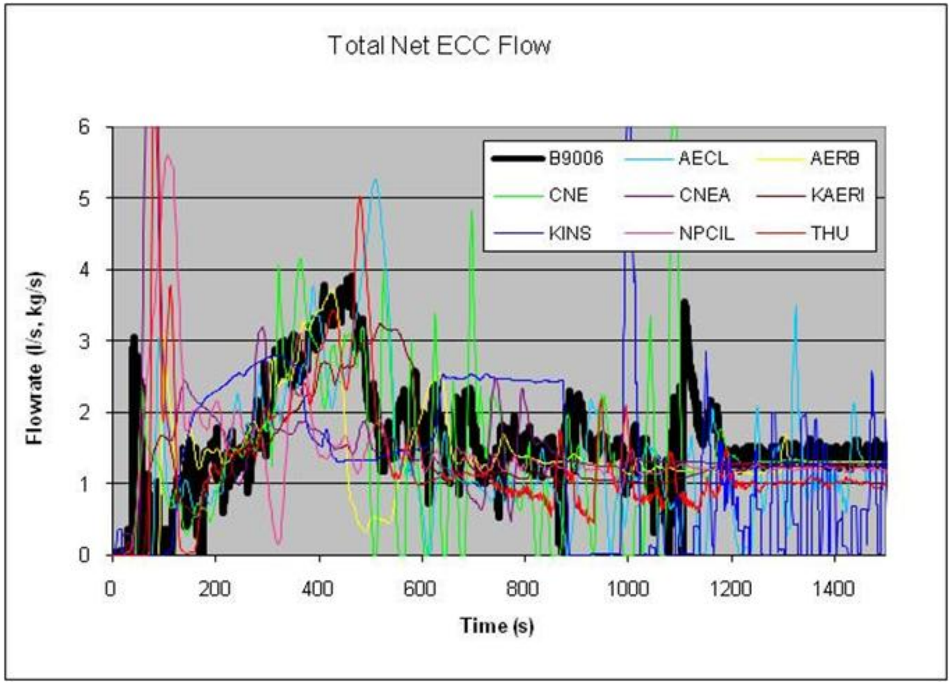
Diametral and Longitudinal Creep in Pressure Tubes (2)



HWR Thermalhydraulic Code Evaluation with SBLOCA Experimental Data (1)



- 7 mm Inlet Headr SBLOCA
- 3 mm Inlet Headr SBLOCA



HWR Thermalhydraulic Code Evaluation with SBLOCA Experimental Data (2)

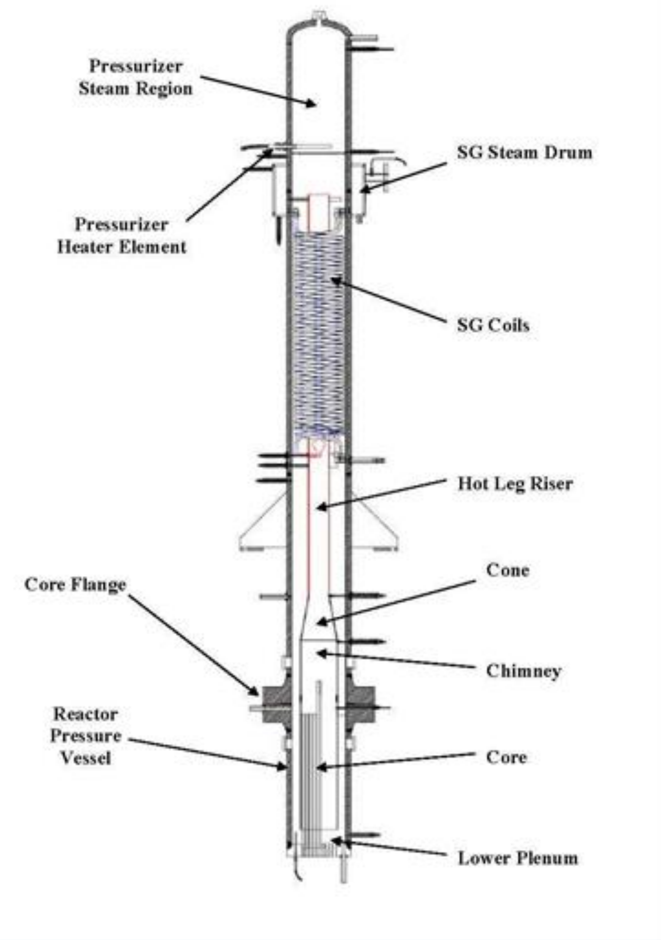
- All calculations were capable of achieving steady state conditions consistent with the experimental data apart from deviations in flow distribution among the individual parallel channels in each pass.
- For SBLOCA, channel voiding is not a concern as much as for LBLOCA.
- All main phenomena (e.g. break discharge, coolant voiding, pressure drop, boiling and condensation heat transfer, and temperature excursion in the heated sections) are qualitatively captured by the participants.
- The application of codes developed outside the HWR technology did not show any special deficiency in the comparison with the present experimental database. However, the existence of parallel channels, and the potential of flow reversal in some of them, is untypical in other PWR reactor systems and requires special attention in the modeling.
- The performed activity is relevant in assessing the capabilities of codes and permitted the quantification of the amount of discrepancy between measured and calculated values.

Evaluation of Advanced Thermalhydraulic System Codes for Design and Safety Analysis of Integral Type PWR (1)

- ICSP Objectives
  - To compare best-estimate computer code calculations to the experimental data obtained from the integral test facility representing an integral type reactor
  - To improve the understanding of thermal-hydraulic phenomena expected to occur in normal operation and transients in an integral reactor
  - To evaluate the capability of computer codes to adequately predict the occurrence of important phenomena, and the corresponding behaviour of nuclear systems during operating, upset and accident conditions, which are represented in experiments
- Scope
  - A. Normal Operating Conditions at Different Power Levels
  - B. Loss of Feedwater Transient with Subsequent ADS Operation and Long Term Cooling
- Host: Oregon State Univ. (OSU) of USA
- Three phases: Double-blind(Pre-test), Blind, Open
- Experiment was conducted in July 2011

Evaluation of Advanced Thermalhydraulic System Codes for Design and Safety Analysis of Integral Type PWR (2)

Experimental Facility in OSU (MASLWR)



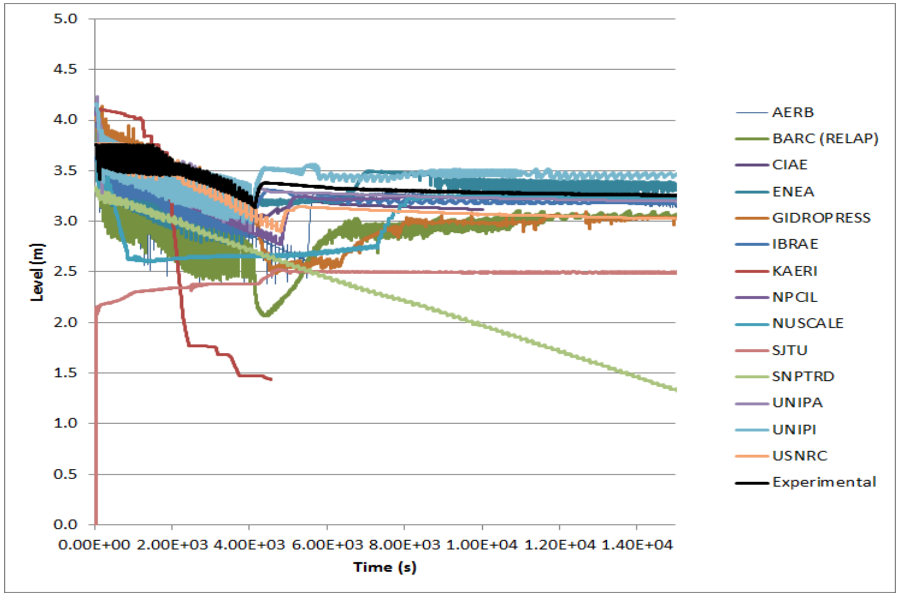
Evaluation of Advanced Thermalhydraulic System Codes for Design and Safety Analysis of Integral Type PWR (3)

- **Participants and Computer Codes**

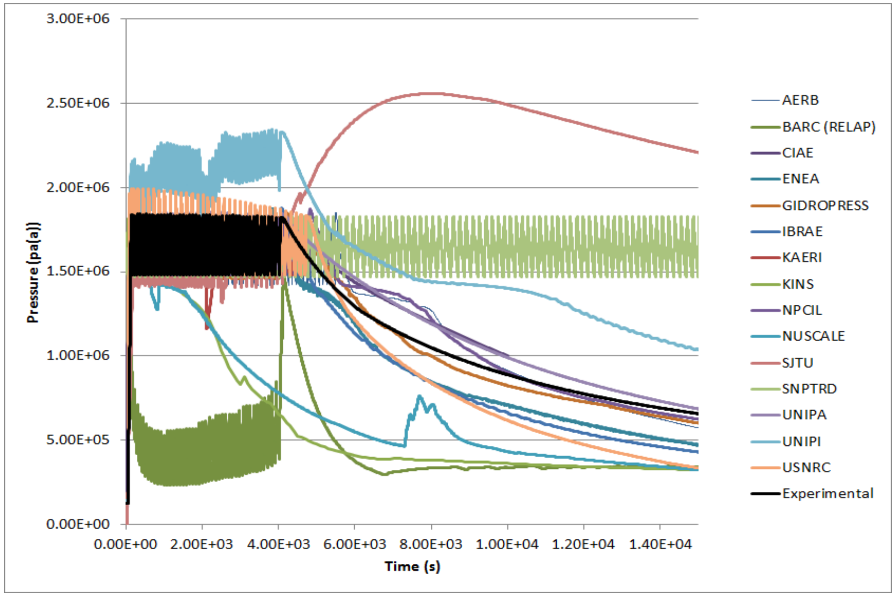
No	Country	Institutes	Codes
1	China	China Institute of Atomic Energy	RELAP5/Mod3.3
2		Shanghai Jiao Tong Univ.	RELAP5/Mod3.4
3		SNPTRD	RELAP5/Mod3.2
4	India	BARC	RELAP5/Mod3.2, CATHARE
5		Atomic Energy Regulatory Board	RELAP5/SCDAP/Mod3.4
6		NPCIL	RELAP5/Mod3.2
7	Italy	Univ. of Palermo	TRACE/v5
8		Univ. of Pisa	RELAP5-3D/v2.4.2
9		ENEA	RELAP5/Mod3.3
10	Rep. of Korea	Korea Atomic Energy Research Institute	TASS/SMR-S
11		Korea Institute of Nuclear Safety	MARS/KS-002
12	Russian Fed.	OKB Gidropress	KORSAR/GP
13		IBRAE	SOCRAT
14	United Kingdom	Serco	RELAP5-3D/v1.2.2
15	USA	NuScale	RELAP5/Mod3.3
16		Oregon State Univ.	RELAP5-3D
17		US Nuclear Regulatory Commission	TRACE/v5.505

Evaluation of Advanced Thermalhydraulic System Codes for Design and Safety Analysis of Integral Type PWR (4)

Comparison of Blind Calculation (LOFW)



RPV Level



HPC Pressure

Evaluation of Advanced Thermalhydraulic System Codes for Design and Safety Analysis of Integral Type PWR (5)

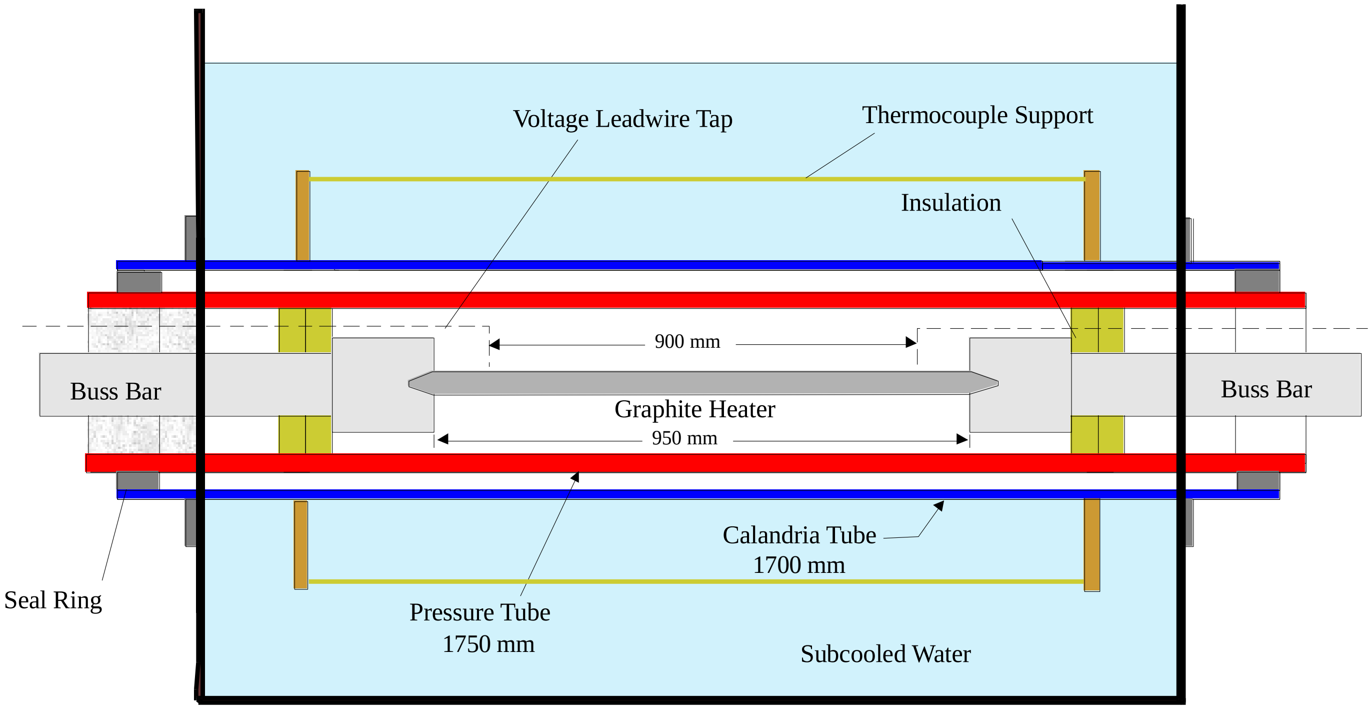
- Current heat transfer correlations applied for simulation of conventional U-tube bundle SG underestimates the heat transfer for the MASLWR helical coil SG.
- The heat transfer and pressure drop in helical coil are further complicated due to the fact that the inlet condition for the helical coil steam generator corresponds to single-phase flow while the outlet condition corresponds to superheated flow requiring not only the estimation of heat transfer and pressure drop under single- and two-phase flow conditions, but also the dryout and post-dryout heat transfer.
- Helical coil SG in MASLWR test facility consists of 14 helical coils (4 inner, 5 middle and 5 outer coils). In general a lumped SG tube model showed more stable behaviour, however this did not allow for study of parallel channel instabilities which cannot be ruled out.
- User effects could be evaluated by comparing the results from the same computer code. Several versions of RELAP5 codes were used by 9 participants. The predictions from the same code show a wide range of spectrum. That means the results are quite sensitive to the nodalization, SG modeling, initial & boundary conditions, and etc.

Fuel Channel Behaviour and Moderator Subcooling Requirements in LOCA with LOECC (1)

- **Objectives of ICSP**
  - to assess the subcooling requirements for a heated pressure tube, plastically deforming into contact with the calandria tube during a postulated large break loss of coolant accident condition.
  - to assess safety analysis computer codes simulating radiation heat transfer to the pressure tube, pressure tube deformation or failure, pressure tube to calandria tube heat transfer, calandria tube to moderator heat transfer, and calandria tube deformation or failure.
- **Safety analysis computer code(s) will be assessed for the following phenomena:**
  - Radiation heat transfer to the pressure tube
  - Pressure tube deformation or failure
  - Pressure tube to Calandria tube heat transfer
  - Calandria tube to moderator heat transfer
  - Calandria tube deformation or failure

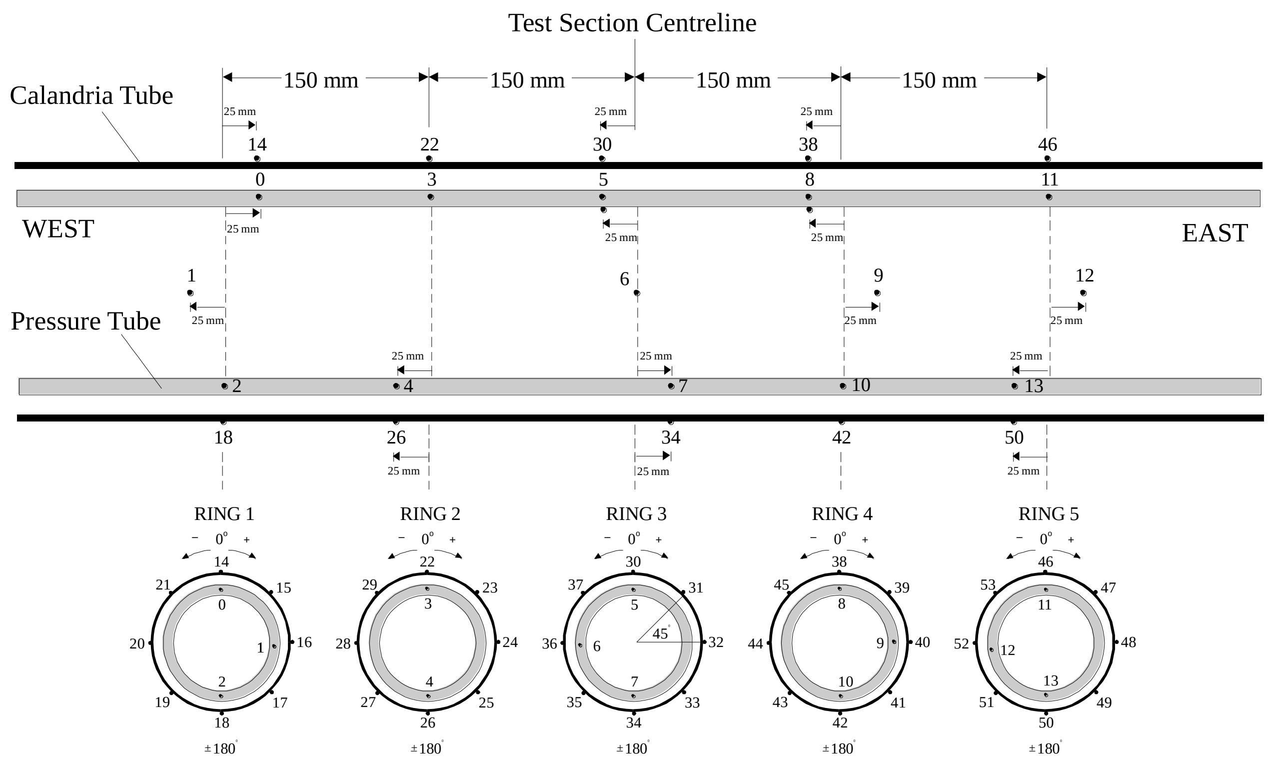
Fuel Channel Behaviour and Moderator Subcooling Requirements in LOCA with LOECC (2)

- **Experimental Facility**



Fuel Channel Behaviour and Moderator Subcooling Requirements in LOCA with LOECC (3)

- **Instrumentation**
  - Heater voltage to measure power supplied to graphite heater
  - Pressure transducers
  - 54 Thermocouples at 5 axial rings to measure PT/CT temperature
  - 4 RTDs to measure water temperature
  - 3 Video cameras



Third Angle Projection

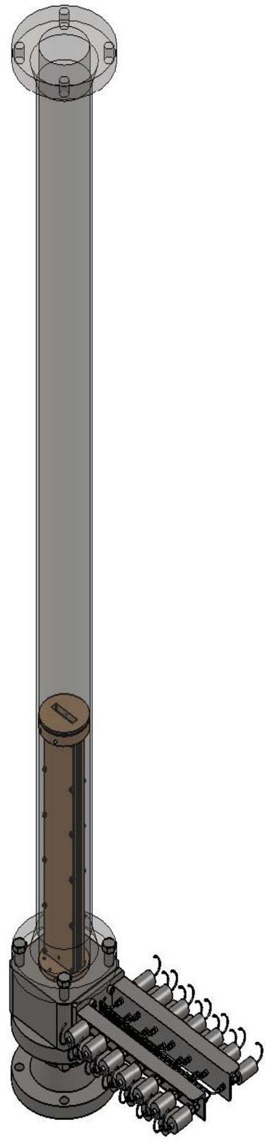
Fuel Channel Behaviour and Moderator Subcooling Requirements in LOCA with LOECC (4)

- **Parameters for comparison**
  - Pressure tube temperature at thermocouple locations
  - Pressure tube deformation - Strain (%) at thermocouple locations
  - Calandria tube temperature at thermocouple locations
  - Water temperature at RTD locations
  - Heater temperature
  - Total heat transfer between heater & pressure tube, pressure tube & calandria tube, from calandria tube to moderator
  - Pressure tube to calandria tube contact heat transfer coefficient
  - Heat transfer coefficient at calandria tube surface

Hydro-Mechanical Behaviour in Reactor Core with Plate-Type Fuel Assembly (1)

- **ICSP hosted by OSU, USA**
- **Purpose of ICSP**
  - Evaluate of computer codes used for fluid-structure interaction in reactor design and safety analysis
  - Improve understanding of fluid-structure interaction phenomena in reactor core with plate-type fuel assembly
- **Objectives of ICSP experiment**
  - Identify the flow conditions which induce the onset of elastic plate deformation for an aluminum clad uranium-molybdenum fuel alloy,
  - Identify the flow conditions which produce plate failure of each fuel plate material.
  - Experimental data may be utilized to assess multi-physics tools simulating the following specific phenomena:
    - Flow disparity within a fuel plate assembly,
    - Bulk computational thermal hydraulic characteristics,
    - Influence of pressure boundary condition on solid domain,
    - Fuel plate plastic deformation and vibration.
- **The 1st Meeting:** 27-30 August 2013, Oregon State University, Corvallis, Oregon, USA

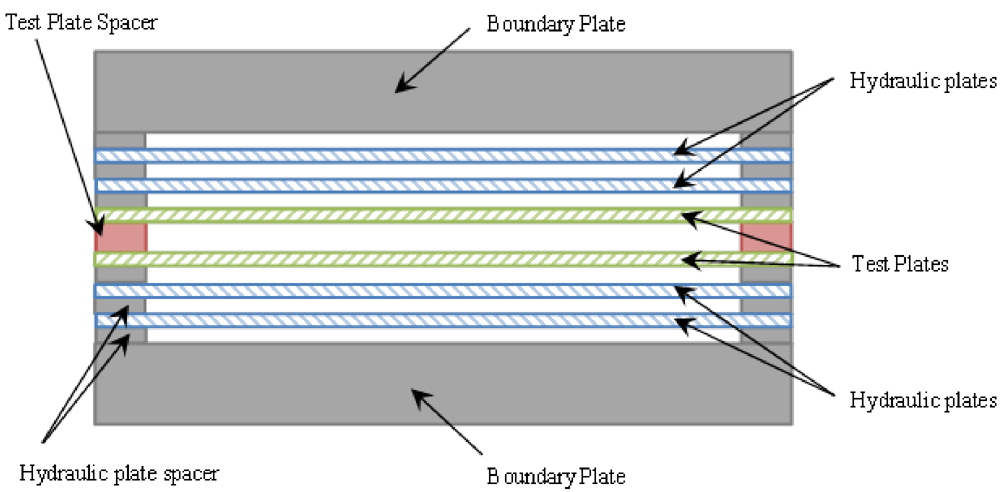
Hydro-Mechanical Behaviour in Reactor Core with Plate-Type Fuel Assembly (2)



HMFTF fluid operating range

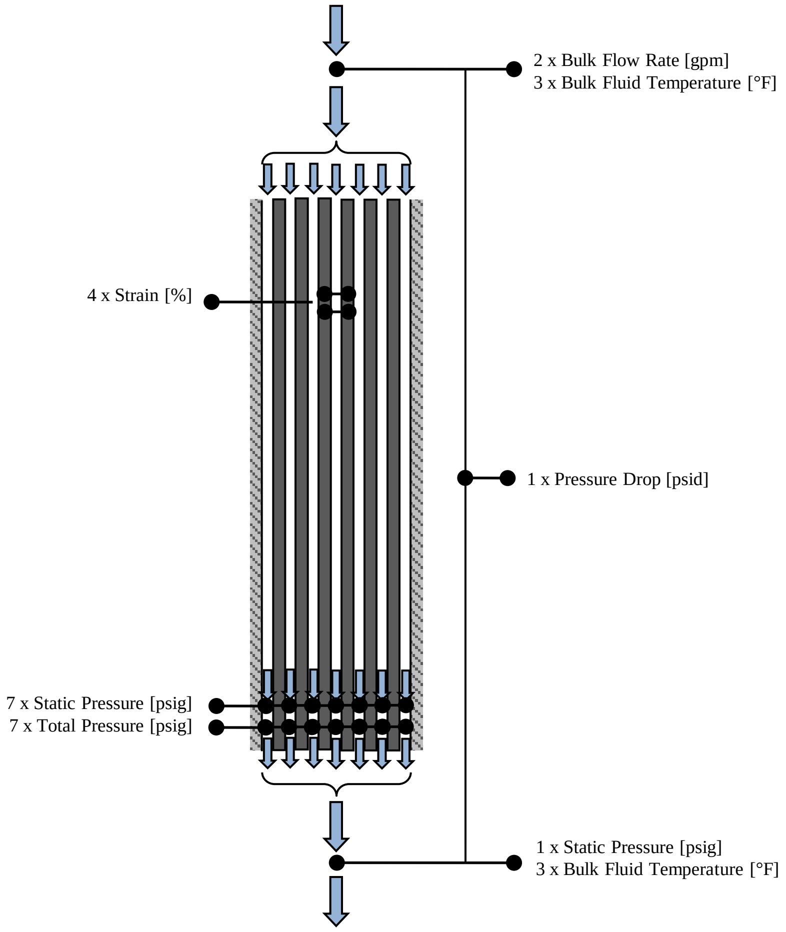
Parameter	Value
Flow Rate Range (liters/sec)	0 – 100.94
Pressure Range (MPa(a))	0.101 – 4.137
Fluid Temperature Range (°C)	20 – 238
Conductivity Range (micromhos)	1 – 3
pH Range	4 – 8

Test Plate Assembly (TPA)



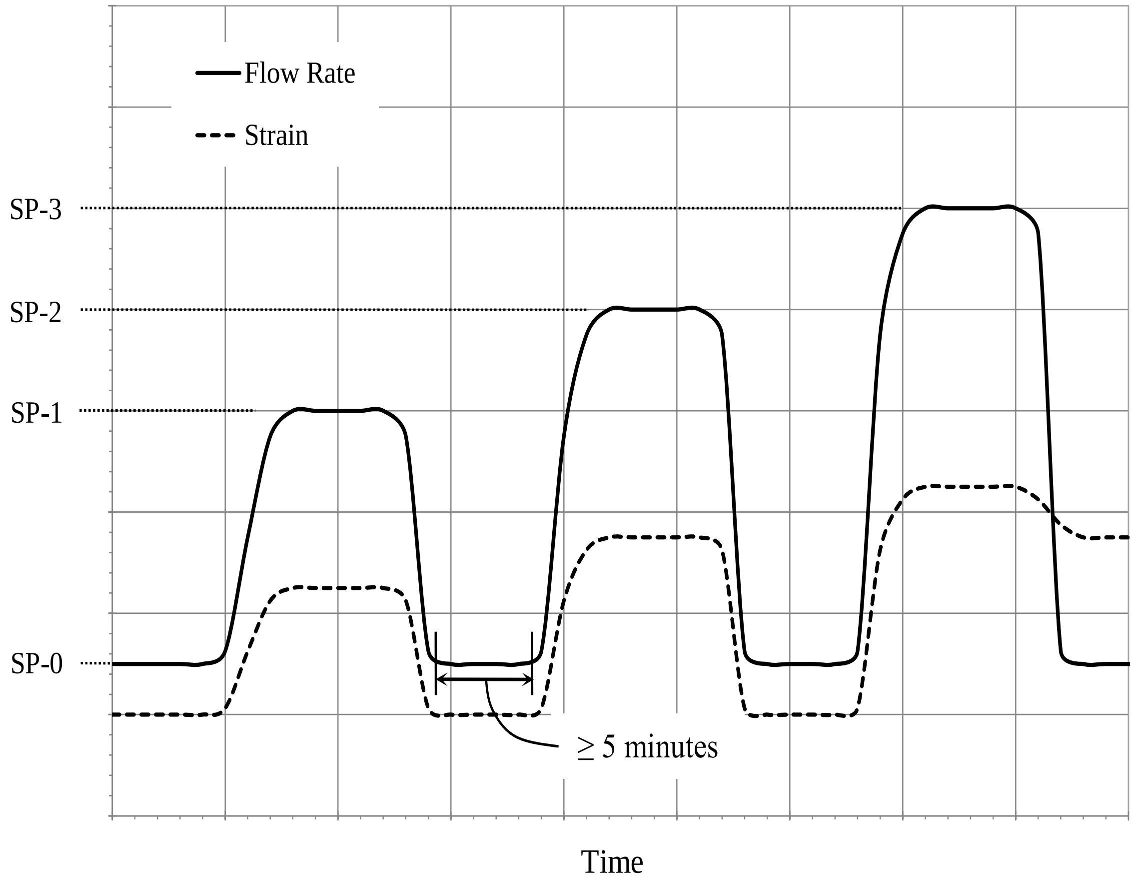
Hydro-Mechanical Behaviour in Reactor Core with Plate-Type Fuel Assembly (3)

- **Instrumentation plan**
  - Bulk fluid pH
  - Bulk fluid conductivity
  - Bulk fluid volumetric flow rate upstream of the test element
  - Bulk fluid temperature upstream of the test element
  - Bulk fluid temperature downstream of the test element
  - Bulk fluid pressure downstream of the test element
  - Local static fluid pressure in every flow channel near the outlet of the test element
  - Local total fluid pressure in every flow channel near the outlet of the test element
  - Local plate strain on two plates of interest in the test element
  - Local plate strain-temperature compensation on two plates of interests in the test element.



Hydro-Mechanical Behaviour in Reactor Core with Plate-Type Fuel Assembly (4)

Experimental Plan



Summary & Conclusions

- The International Atomic Energy Agency (IAEA) organizes Coordinated Research Projects (CRPs) and International Collaborative Standard Problems (ICSPs) to assist research and development in IAEA Member States.
- Many of CRPs and almost all ICSPs are focused to facilitate the development and validation of computer codes for NPP design and safety analysis.
- By participating in CRPs or ICSPs, participants may identify any weaknesses or strengths of their prediction tools, search a way to overcome any limitations, and also suggest any further experiments or analytical models to fill the gap identified in the specific CRP or ICSP.