Scaling in Experiments for Reactor Safety Analysis

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EASI-SMR, Scaling, Bologna, Dec.2025

Hideo NAKAMURA

born on Feb. 15, 1956

Current Major Interest

Thermo-Hydraulic (T/H) Safety & Severe Accident (SA) Research for **LWR**s
- ✓ Two-phase flow, heat transfer and measurement/instrumentation
- ✓ Integral & separate-effect tests at prototypical pressure
- ✓ Computer code development, verification and validation (V&V)
- ✓ Phenomena Scaling and Uncertainty Quantification in Reactor Safety Analysis

Engineering Ethics and **Organization Culture for Safety** (**Safety Culture**)

Preparedness for Nuclear Emergencies

Work History

2022 - *present* **OECD/NEA CSNI/WGAMA Chair**
2020 - 2021 Vice chair
2010 - *present* Bureau member (*CSNI member* 2013 - 2016)

2018 - *present* Technical Associate
2016 - 2017 Senior Associate
2012 - 2015 Deputy Director General

Nuclear Safety Research Center (NSRC)
Office for Analysis of Regulatory and International Information
Nuclear Emergency Assistance & Training Center (NEAT)

2005 - 2012 **OECD/NEA ROSA & ROSA-2 Projects** (Head of Operating Agent) with LSTF Experiments & Analyses
2001 - 2012 Group Leader of T/H Safety Research Group (**JAERI** to **JAEA** in 2005)
SARL consolidated into Thermo-hydraulic Safety Engineering Laboratory

1999 - 2001 **SA** Research Laboratory (*SARL*) in JAERI

1992 - 2001 **IFMIF** Jet Flow Target Development / ITER Project (**Fusion**)
1987 - 1988 Resident Researcher at Nuclear Research Center of **CEA** Grenoble (**CENG**) – **Cathare**/Bethsy
1981 - 1999 **ROSA-III**, **-IV** & **-V** Programs at **T/H Safety** Engineering Lab. in JAERI

Education (Nagoya University)

1992	Doctor of Engineering	Nuclear Engineering	**Thermal-Hydraulics**
1981	Master of Engineering	Crystalline Material Engineering	**Fusion** Tokamak First Wall **Material**
1979	Bachelor of Engineering	Nuclear Engineering	**Fusion** Tokamak First Wall **Material**

Contents

Prologue / Ultimate Goal ?

Ultimate Goal ?

*Accurately represent **any** of thermal-hydraulic phenomena by using computer codes with uncertainties that we may quantitatively estimate.*

What are the difficulties ?

towards the preparation of safety assessment codes

- ✓ Component Size Very Large
  - ☞ difficult --- 1:1 experiment
- ✓ Under high-Pressure & -Temperature
  - ☞ difficult --- under prototypical conditions
- ➢ Reference Data -- *Extremely limited*
  to validate computer codes, especially for local singularities (reactor design specific)

... but... Nuclear Reactors are real

Requirements for Experiments

- ✓ Equal conditions ??
  - Prototype for direct confirmation
  - Influence of radiation (ex. core)
- ✓ Scaling
  - Size & geometry
  - Fluid properties
- ✓ Type of experiments vs. IBCs
  - System-integral (ITF)
  - Separate-effect (SETF)
- ✓ Measurement
  - Parameters
  - Resolution (time, space)
  - Correspondence with *analysis*

Next Page

Computer Code and Experiment

Code Development, with Verification & Validation (V&V)

- **Safety evaluation of LWRs**
  - ✓ DBAs & beyond-DBAs including severe accidents
  - ✓ **Complex 1ø & 2ø flows** during various types of accidents & transients
- **Best-estimate (BE) code** (vs. *Conservative Evaluation Model (EM) code*)
  - ✓ Realistic evaluation of **safety margin** of reactor
  - ✓ Need **V&V** + quantification of **uncertainties**
    - Intensive utilization (power uprate, life extension, high-burnup fuel...), III+ etc.
- **Computational fluid dynamics (CFD) code**
  - ✓ In-depth understanding of local (accident) phenomena
  - ✓ **Recent methods** -- Coupling with **BE**, Extension to prototypic condition etc.
  - ✓ **Validation** with high-spatial resolution, dynamic data, up to prototypical size ... **How** ?
    - Post-accident analysis, Validation of 1-D Model, Design-by-analysis etc.

Thermal-hydraulic Experimental Data

- **Detailed data** to develop and validate BE & CFD codes
- **Mostly** with ***scaled-down*** **test facilities**
  = ***not prototypical***

Computer codes *provide*
Approximation *through*
Simplification Assumptions

But
Complementary

Code *and* ***Experiments***
Both = Incomplete

However

We may be seeing
artificially-interpreted images
as ***a reality of reactor***

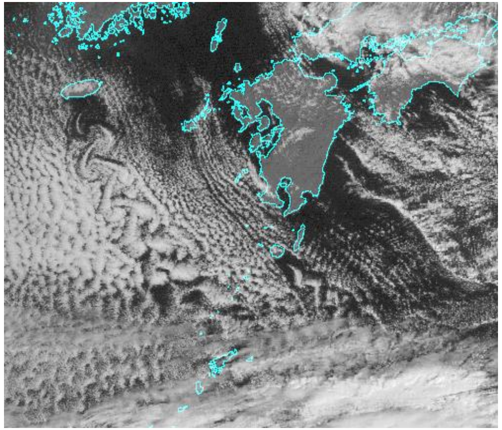
Such ... the reality ??

or just virtual ??

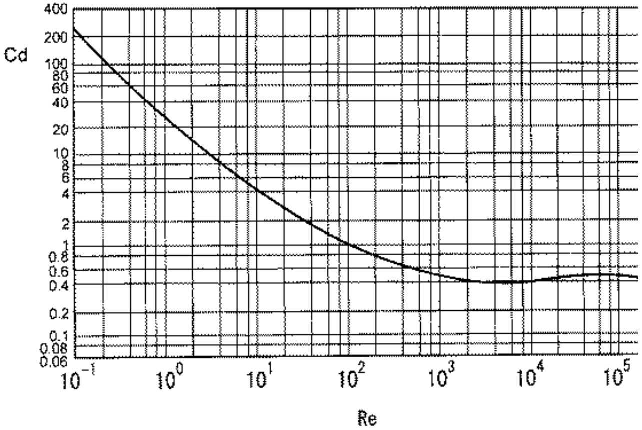
A General Method for Scaling

- **Long history for Engineering Applications**
  - Buckingham Theorem = Non-dimensionalization
  - (Many) Dimensionless or scale-independent quantities such as *Re*, *Ra*, *We*, *Fr*, *Nu*, , ,

- **Good for Local Phenomena** -- *Examples on Wake*
  - Less interaction with circumferential phenomena



(2-D ?) Kármán Vortices
Jeju and Yaku Islands
(Satellite)

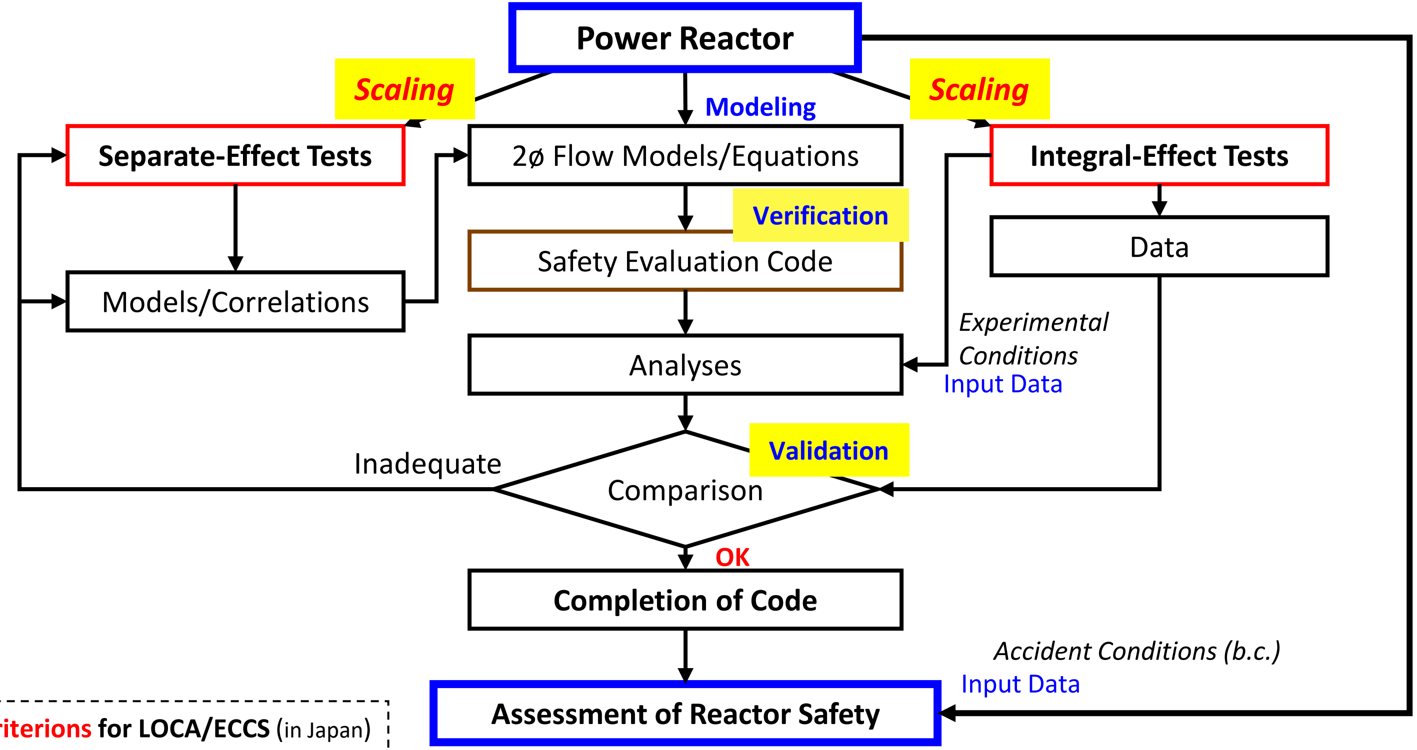


Drag Coefficient for a Sphere

(ex.) *$Re \sim 10^7$*
*for **BWR feedwater nozzle flow meter***

- **Extrapolation** or **Interpolation**
  - May cause **problem**
    (ex., *ASME V&V 20 Standard*)
- We may encounter …
  - Phenomena of **mutual interactions** during accident …,
  - Combination of **different scaling** … and …
- How about **Safety Analyses** ?

Diagram of Preparation of Computer Code for Safety Analyses



Criterions for LOCA/ECCS (in Japan)

1) T_{max} of fuel cladding = 1200 ° C
2) Cladding oxidation ≤ 15%
3) Small hydrogen generation
4) Long-term core cooling, including influences of clad deformation

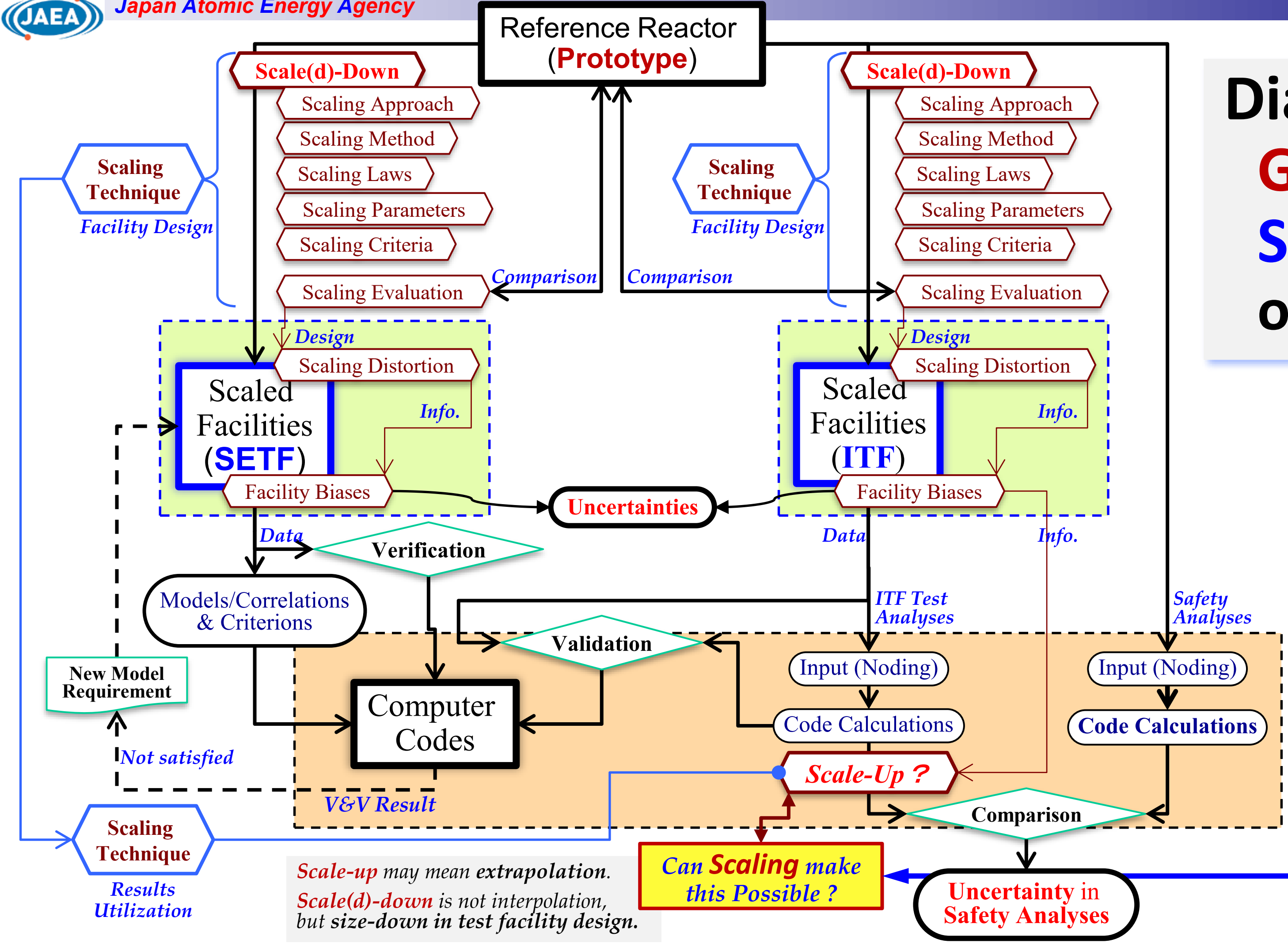
Judge

EM: *with* ***conservative assumptions***
BE: *with* ***Base case + BEPU***

We need to evaluate

- **Uncertainties w/o prototype (ref.) data**
- **Under influences of *Scaling***

Diagram of Glossary for Scaling SOAR of OECD/NEA



*Scale-up may mean **extrapolation**.*
*Scale(d)-down is not interpolation, but **size-down in test facility design**.*

BTW,

Do you know *all* the *prototype phenomena* ?

ROSA Programs in JAEA

ROSA-I(1970-73)
Prototypical-scale tests

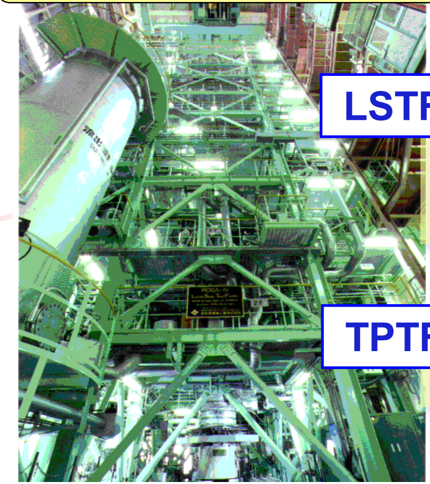
LOCA phenomena
(blowdown, core heat transfer)

ROSA-II(1974-77)
PWR LBLOCA

'79 TMI accident

ROSA-III(1978-83)
BWR LB and SBLOCA

ROSA-IV(1980-92)
PWR SBLOCA and transients



LSTF

'85- Shakedown
'89- OECD-NEA **ISP-26**
'91 **Mihama-2 SGTR** Simulation

TPTF

'82- Shakedown

ROSA-V(1991-)
Accident Management, Passive Safety of Next Generation LWRs

'92- **AP600** Simulation with USNRC
'98- **PCCS** (Horiz.Hx) for BWR with JAPC
'05- OECD-NEA **ROSA** & **ROSA-2** **Projects**

Objectives

- ✓ **To understand / find new phenomena at accident and transient with experiments under prototypical conditions**
- ✓ **To assess accident response and recovery procedures**
- ✓ **To develop & validate safety assessment codes**

- ***Prototypical*** *for both of* ***fundamental*** *&* ***system-effect tests*** *to confirm* ***ECCS performance*** *and* ***code development & validation***
- ***Always together with Scaling Considerations***

Personal Experiences in Scaling

… through the ROSA Programs in JAEA

… and some ohers

Personal Experiences in Scaling … through the ROSA Program in JAEA

(1) ROSA-II: ECCS Problem ??

(2) ROSA-III: Half-height Facility

(3) ROSA-IV TPTF: Flow Regime Transition in Horizontal Steam-Water Two-phase Flows

(4) ROSA-IV LSTF: Natural Circulation (NC)
(a) Non-homogeneous U-tube Behavior
(b) Flow Oscillation in 2Ø Flow NC mode

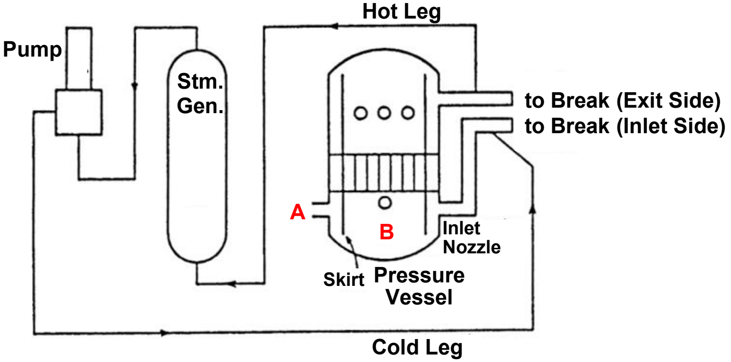
(5) ROSA-V LSTF: Counterpart Testing with PKL (OECD/NEA Joint Project ROSA-2)

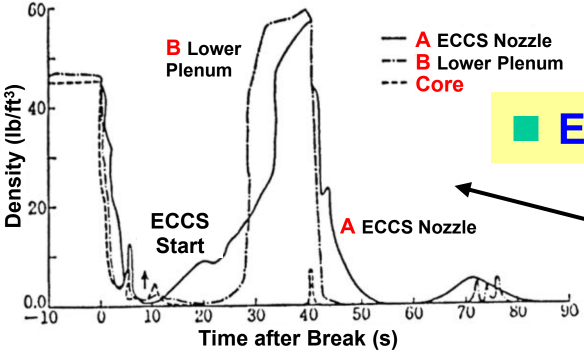
… and ***some others***

ROSA-II: ECCS Problem ??

ECCS Problem (PWR Large-break LOCA Simulation)

Semiscale (1970–71)





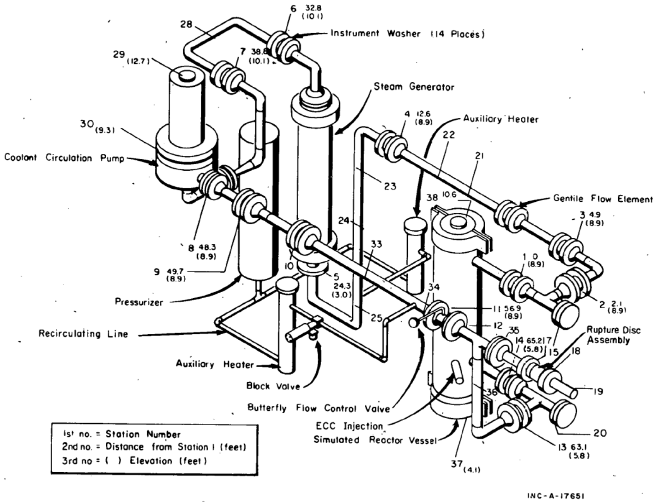


Fig. 1 Single-loop semiscale -- high inlet break configuration.

■ **ECCS coolant does not reach the core**

Because of Facility Simulation Capability

Expected Causes:

- Hot Leg Flow Resistance
- Downcomer Heat Loss

➡ Both Excessive

… but was **ECC Bypass**

- **TIETHYS** https://www.oecd-nea.org/tiethysweb/
- G.G. Loomis, "Summary of the SEMISCALE Program (1965-1986)," NUREG/CR-4945 (1987) https://inis.iaea.org/collection/NCLCollectionStore/_Public/19/018/19018882.pdf

Semiscale Series 800 (1970-71)

- Semiscale Blowdown and Emergency Core Cooling (ECC) Project --- part of Water Reactor Safety program by the U.S. Atomic Energy Commission
- Daughter to LOFT (Loss-of-Fluid Test) with nuclear fuel

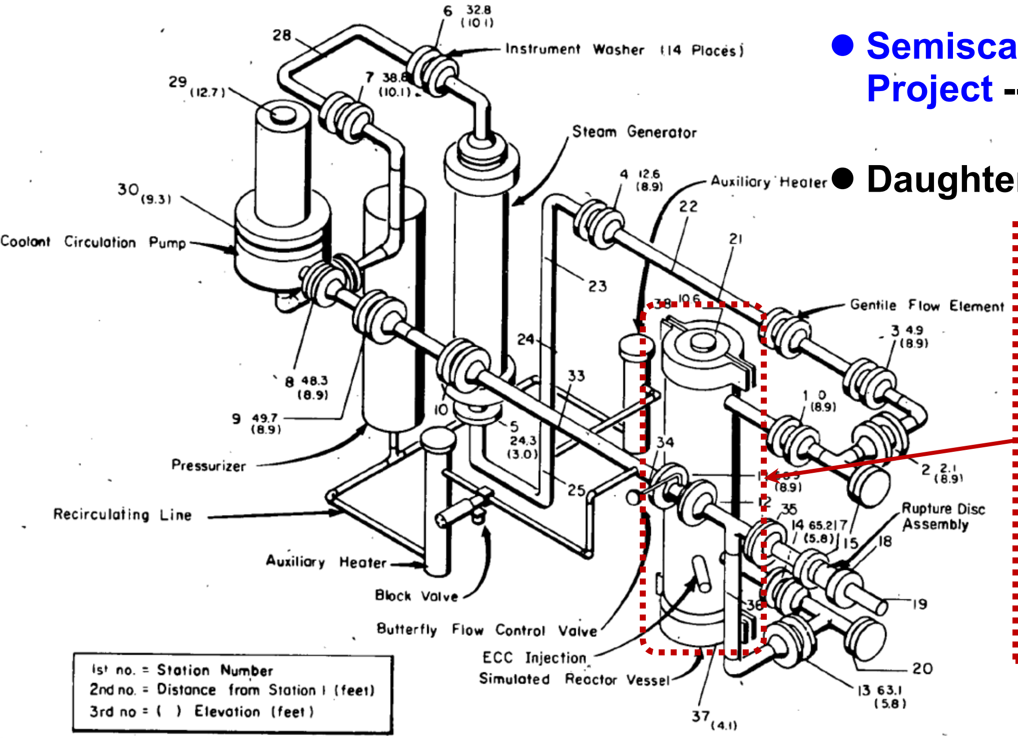
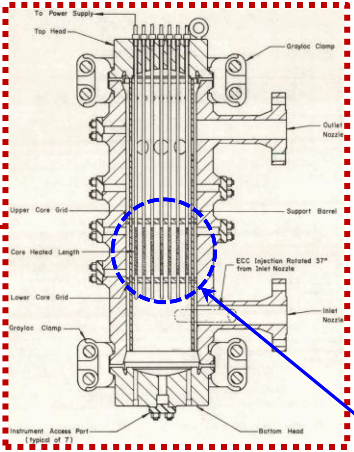


Fig. 1 Single-loop semiscale -- high inlet break configuration.



Simulated Reactor Vessel

Simulated core heated length: 0.22 m

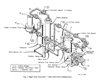
- PWR Large-break LOCA simulation
- No scaling consideration
- ECC bypass observed *first*
  - ➢ Effectiveness of ECCS became in doubt

Configuration for TESTs 845 - 850
(ECC Injection – Break Loop Only)

- TIETHYS https://www.oecd-nea.org/tiethysweb/
- G.G. Loomis, "Summary of the SEMISCALE Program (1965-1986)," NUREG/CR-4945 (1987)
- H.W. Heiselmann, et al., "Semiscale Blowdown and Emergency Core Cooling (ECC) Project Test Report -- Tests 803 through 820," IN-1404 (Oct. 1970)
- O.I. Olson, "ibid.-- Tests 848, 849, and 850 [ECC Injection]," ANCR-1036 (June 1972)

Evolution of Semiscale

Series 800-900 (1969-72)



- Small-height core (0.22 m)
- Full-pressure Full-temperature

Mod-1 (1974-77)

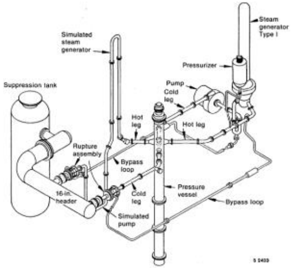


Figure 4. Semiscale Mod-1 system for cold leg break configuration.

- Half-height core (1.68 m)
- Full-pressure Full-temperature

Mod-2C

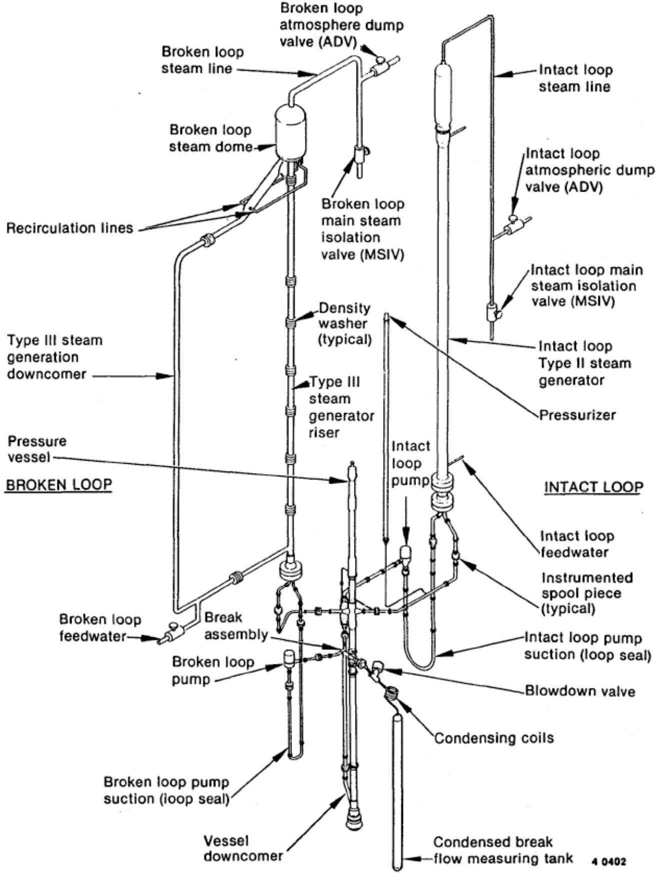


Figure 8. Semiscale Mod-2C system configuration.

Final Configuration (1985-86)

- Full-height core (3.66 m)
- Full-pressure Full-temperature
- Volume: 1/1500
- Ref. Reactor: Zion (4-loop)

- G.G. Loomis, "Summary of the SEMISCALE Program (1965-1986)," NUREG/CR-4945 (1987) https://inis.iaea.org/collection/NCLCollectionStore/_Public/19/018/19018882.pdf

ROSA-II Program (1974 - 77)

The 1st IET in JAEA for PWR Large Break LOCAs

- **Volume** = 1/400, **Height** = 1/2, **2 Loops**
- **Full-Pressure and -Temperature**
- **Core Power** = 2.4 MW

Objectives

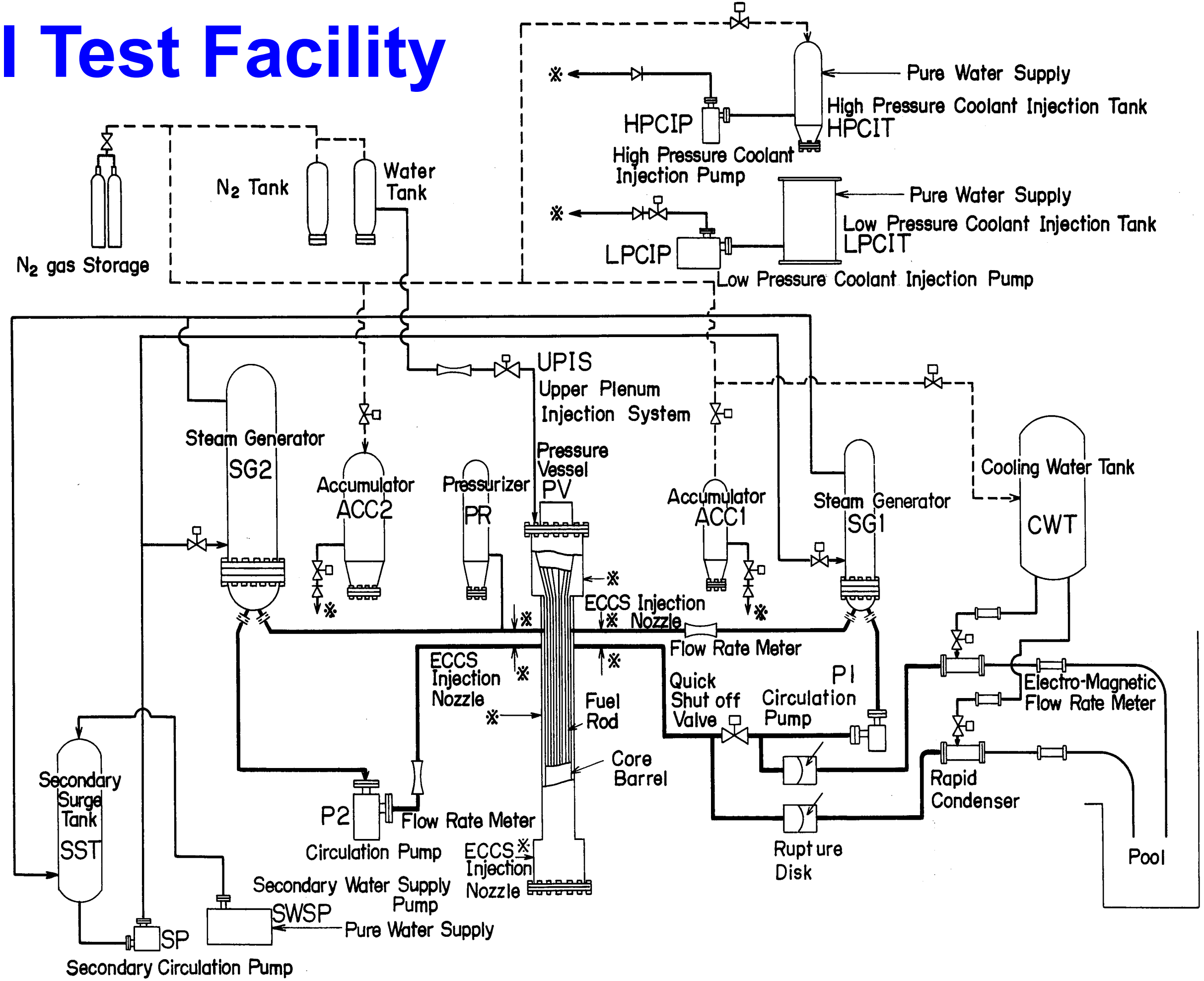
- To understand T/H response during PWR LB-LOCA (Blowdown to Reflooding) in a systematical way
- To clarify parameters relevant to core cooling by ECCS injection

Major Results

- Several effects that may influence core cooling clarified, such as condensation onto ECCS water, stored-heat of reactor structure, Counter-Current Flow Limiting (CCFL)
- Detailed investigation on
  ECCS Problem (= **ECC bypass**) in **Semiscale** experiments (1971)
- Proposal of alternative (improved design) ECCS
- Validation of RELAP3 and RELAP4 codes

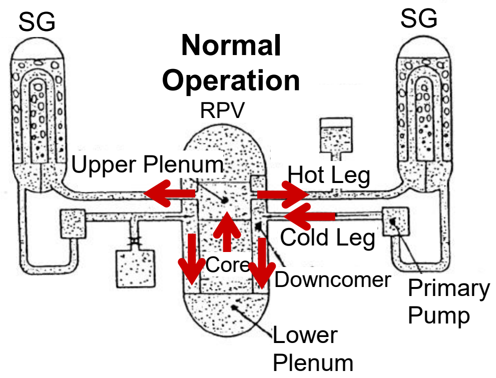
● H. Adachi et al., “ROSA-II Experimental Program for PWR LOCA/ECCS Integral Tests” JAERI-1277 (1980)

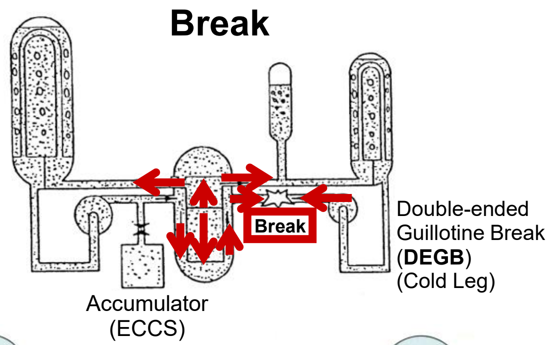
ROSA-II Test Facility



- H. Adachi et al., “ROSA-II Experimental Program for PWR LOCA/ECCS Integral Tests,” JAERI-1277 (1980)

PWR Large Break LOCA





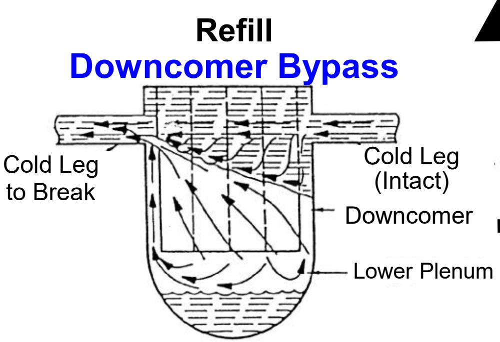
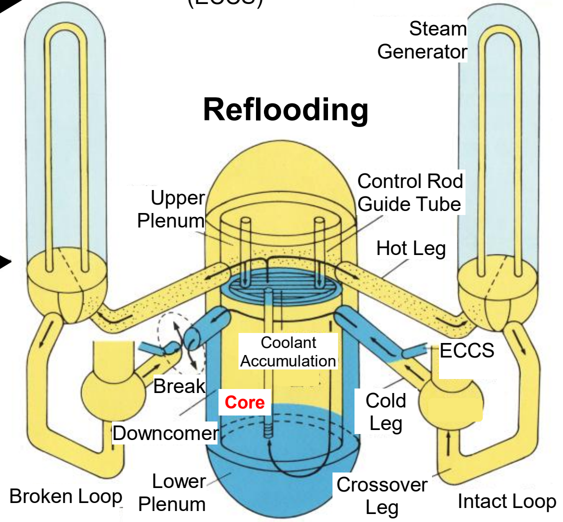


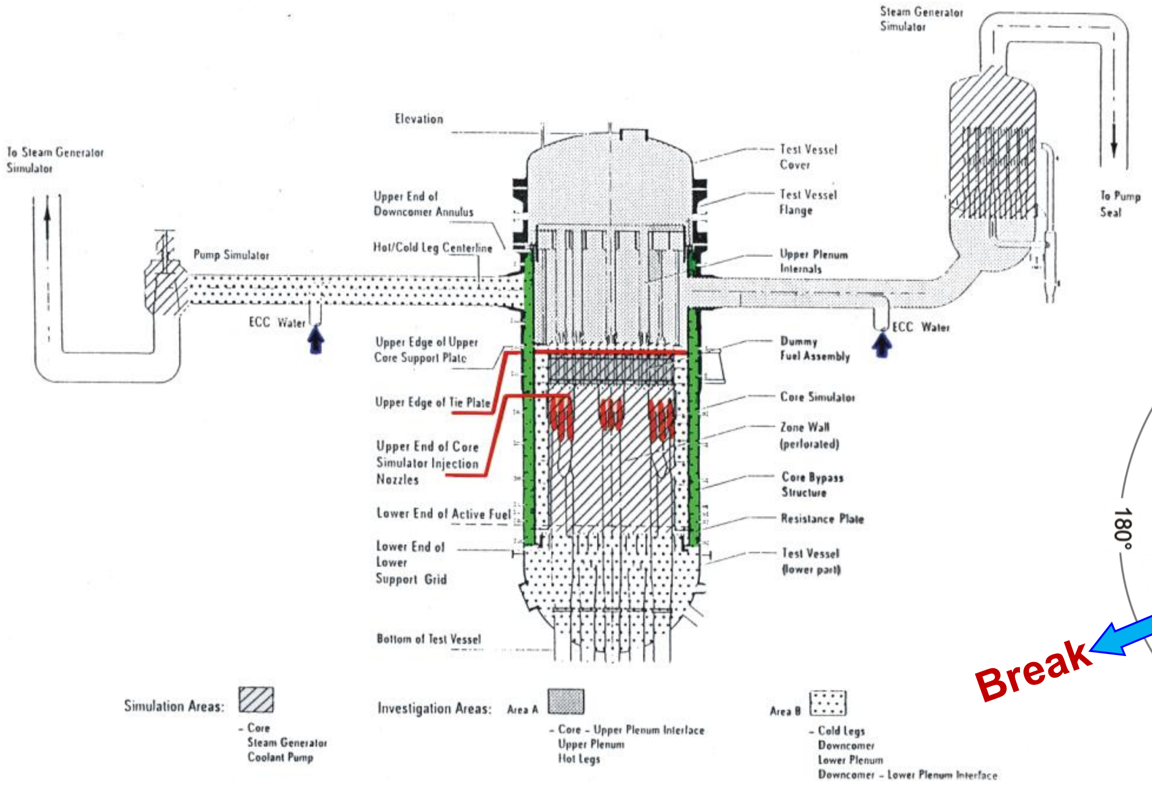
Image from
CREARE Experiment (US)



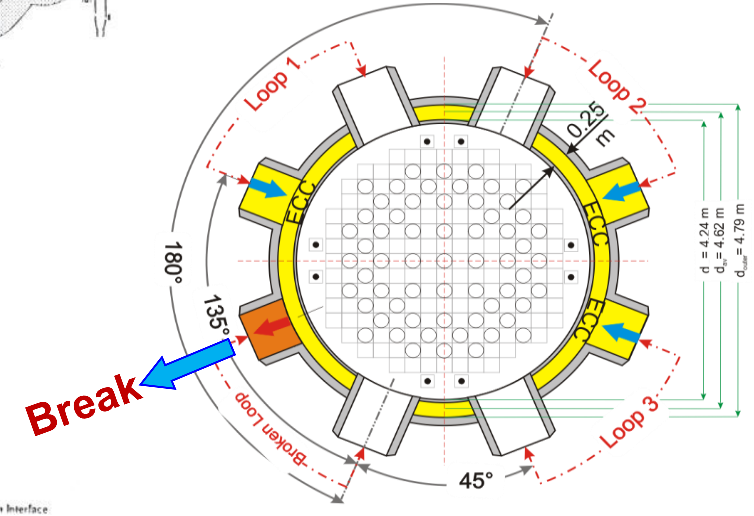
- **PWR LB-LOCA = DBA (Design Basis Accident)**
- **High PCT**
- **Occurrence Frequency of DEGB $< 10^{-7}$**

UPTF Experiment

Downcomer Bypass during PWR LB-LOCA Refill Phase



- **German-type 4-loop PWR**
- **Equal-size Components**
- **Steam** from **Coal Power Station** into **Core**

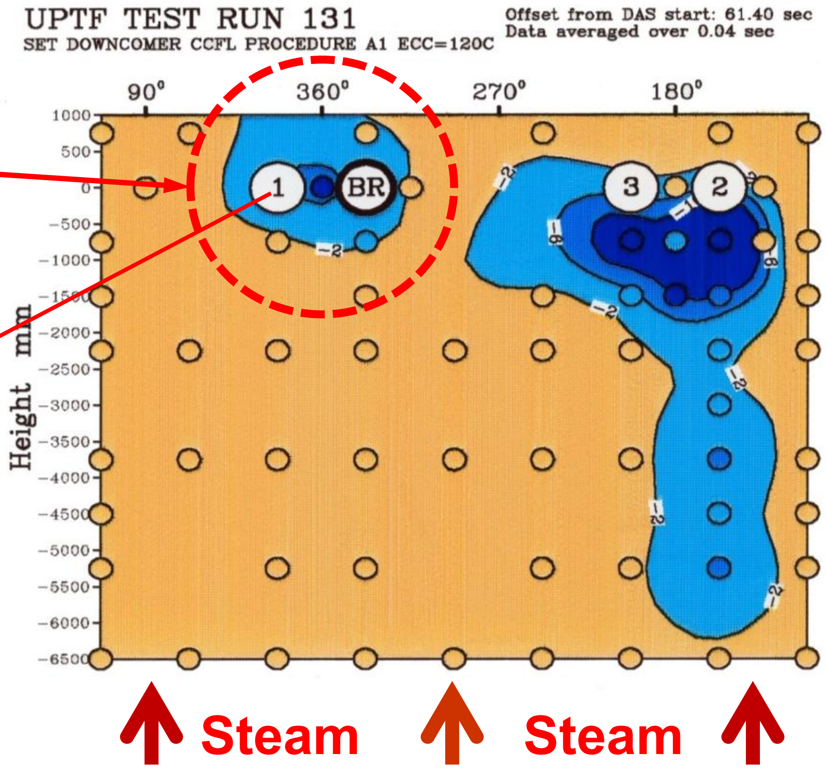


UPTF Observation

Temperature profile in downcomer observed from outside

Downcomer Bypass in Limited Region

3 Cold Legs with ECC Injection into downcomer



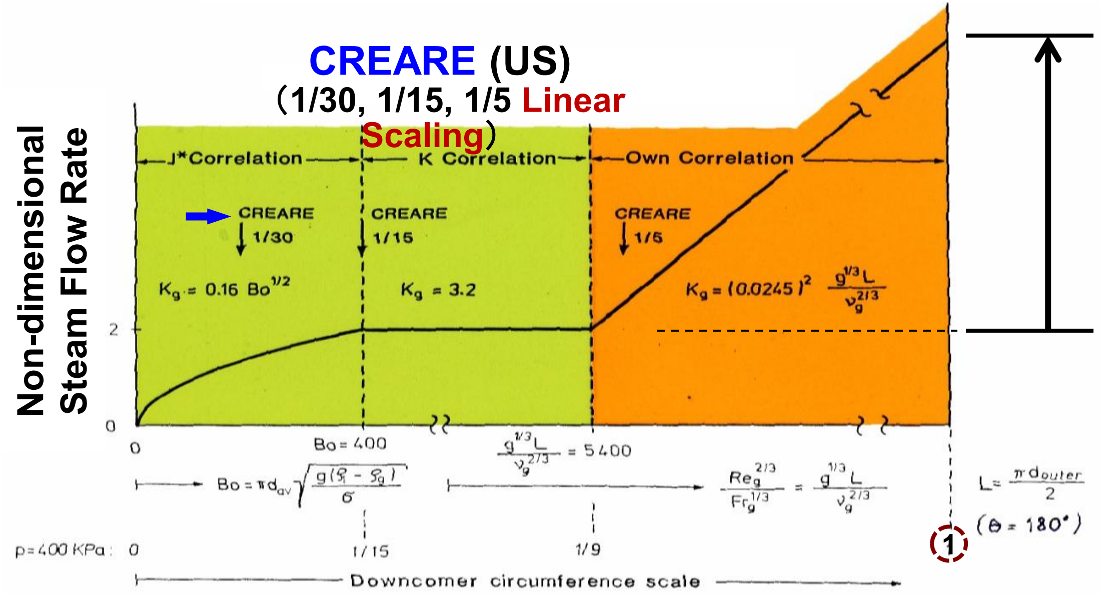
Subcooling in Downcomer

Above		−2
−6	−	−2
−10	−	−6
Below		−10

UPTF Result

Downcomer Flooding Correlation for Total Bypass (Zero Penetration of Liquid)

CREARE (US)
(1/30, 1/15, 1/5 Linear Scaling)



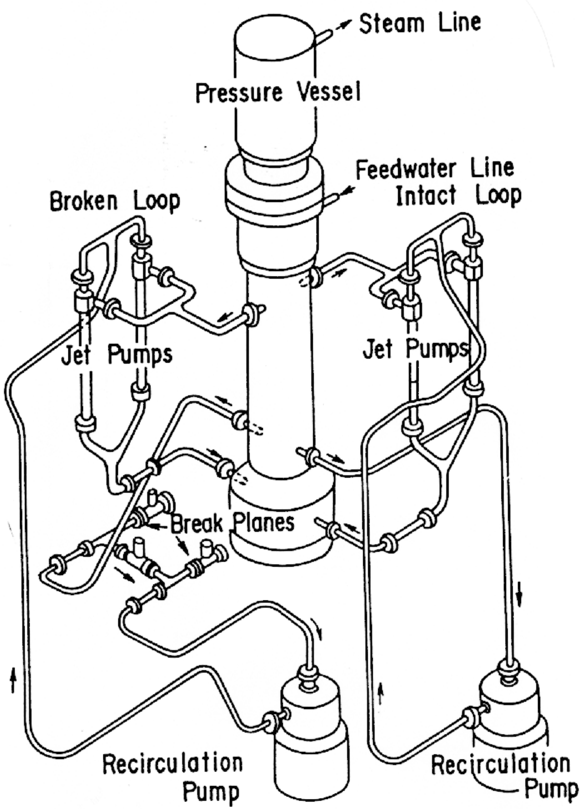
UPTF (Real Size)

- **Significant effect** of **scaling** onto **the key phenomena** related to **flooding initiation time**

1. H. Glaeser, "Downcomer and tie plate countercurrent flow in the Upper Plenum Test Facility (UPTF)", Nucl. Engng. and Des. 113 (1992) 259-283
2. 2D/3D Summary Report, GRS-100 (1992), 2D/3D Program Work Summary Report, NUREG/IA-0126 (1993)

ROSA-III: Half-height Facility

ROSA-III Program (1978～1983)



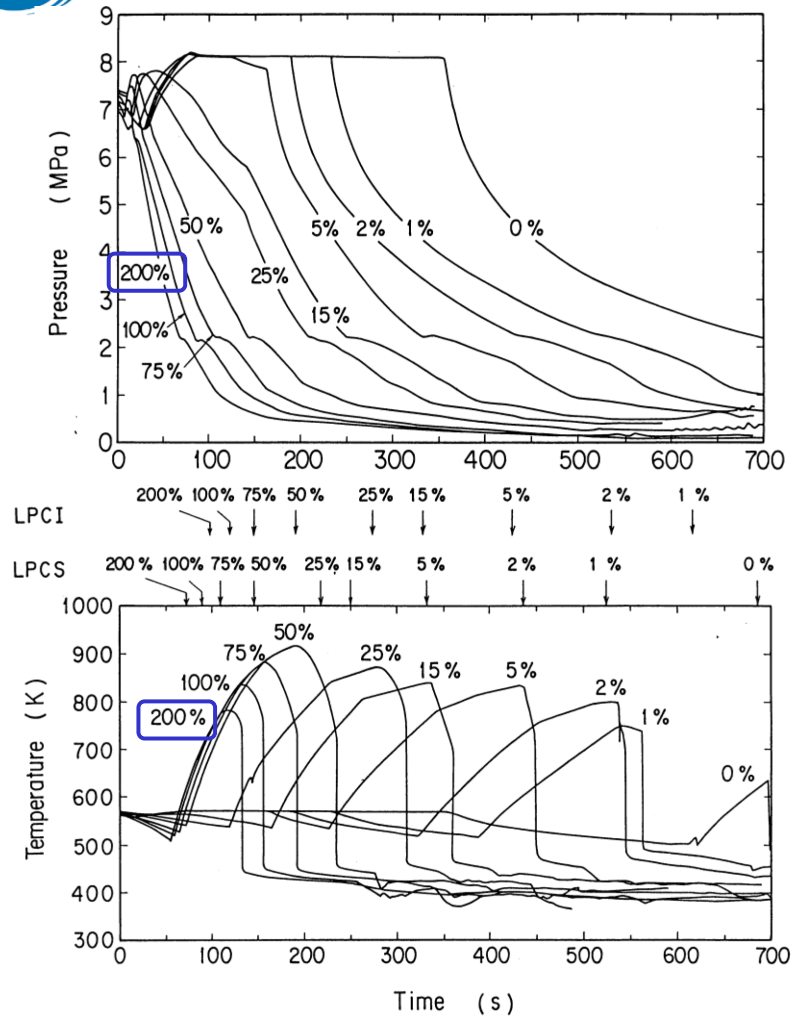
- BWR/6 Simulation – Full-Pressure/Temperature
- Half-height + Volume 1/424
- 4 Core Bundles (8 x 8) + Core Power ≤ 44%
- ECCS -- Full simulation
- Jet Pump – outside of RPV
- TMI-2 Accident (1979)
  LB-LOCA → SB-LOCA

Facility Characteristics

① Single-failure of ECCSs (DBA) + BDBA
② Beyond DBA (multiple-failure)
③ Break area parameter exps. (0 to 200%) for Recirculation pump suction break LOCA
④ ISP-12 (OECD/NEA)
⑤ Main Steam Line Break LOCA
⑥ Natural Circulation
⑦ Various types of parameter experiments etc..

K. Tasaka et al., “ROSA-III Experimental Program for BWR LOCA/ECCS Integral Simulation Tests,” JAERI-1307 (1987)

ROSA-III Program
(1978～1983)



- Recirculation pump suction break LOCA
- Break area parameter experiments
- Assumptions
  - 0 to 200%
  - SBO simultaneous at break
  - Total failure of high-pressure core spray (HPCS)
    = Beyond Design-Basis (BDBA)

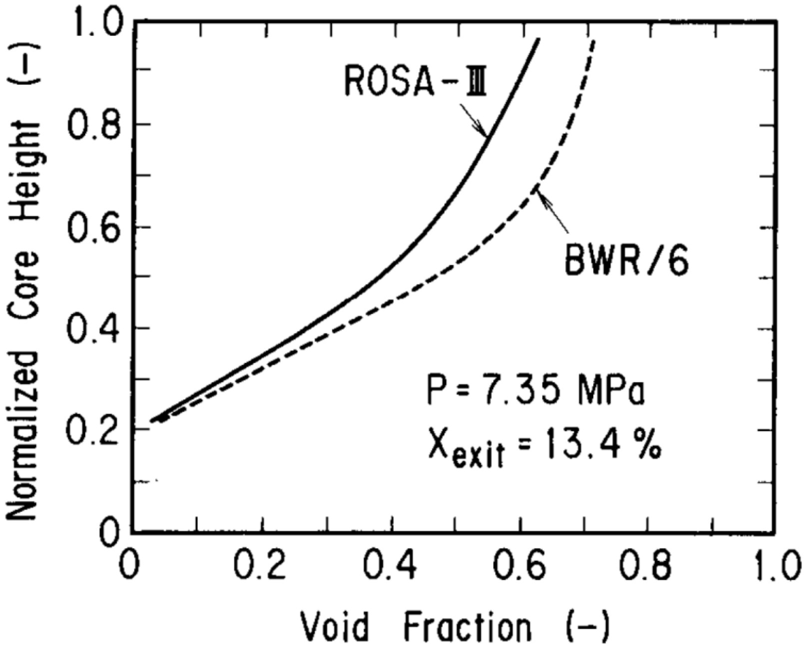
- Peak Cladding Temperature (PCT) as FOM rather low
  (far lower than PWR cases for LBLOCA)
- Fast transient of accident
  = operator intervention d
- Automatic Depressurization System (ADS) crucial to start Low-Pressure ECCSs to duly cool the core

Influences of half-height (core) ??

K. Tasaka et al., "ROSA-III Experimental Program for BWR LOCA/ECCS Integral Simulation Tests," JAERI-1307 (1987)

ROSA-III Program (1978～1983)

Effect of Height Scaling on Void Fraction Profile in the Core during Steady State Operation



Atypicality in the ROSA-III Experiment

- Height of Core = Half of BWR/6 Core
- Low Core Power = 44%
- Low Core Inlet Flow Rate = 44%
  to result the same core exit quality

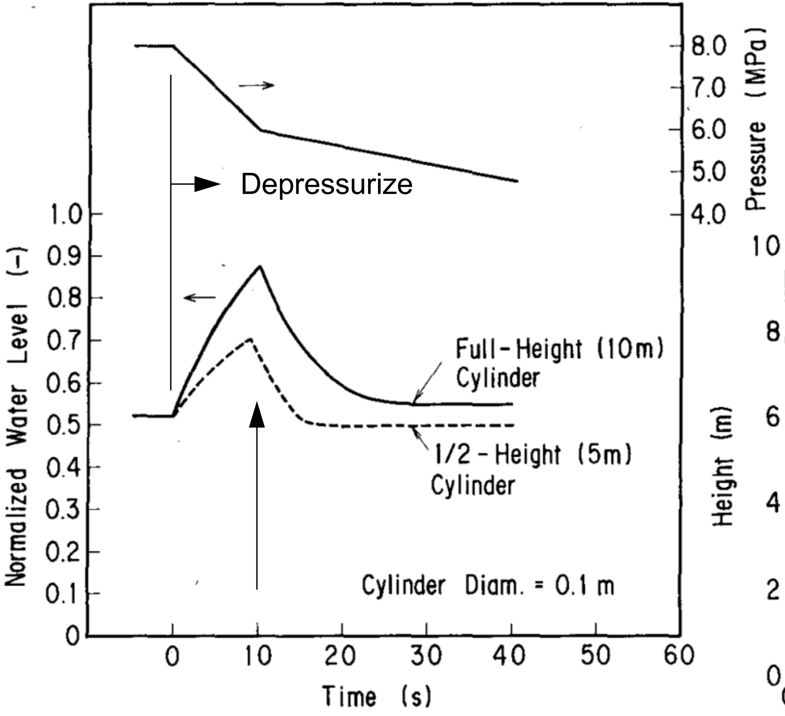
Calculated Void Fraction Profile

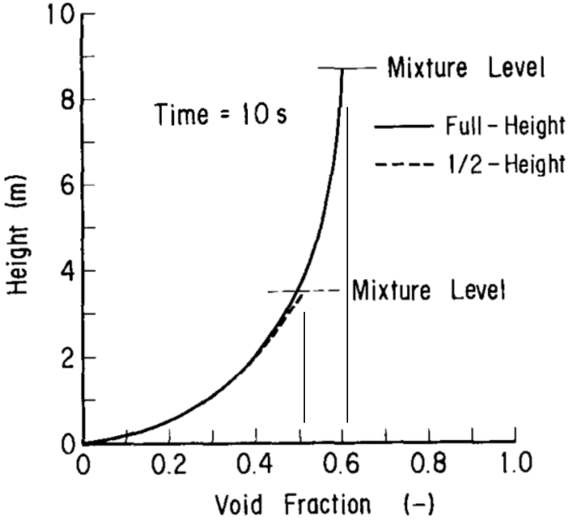
- RELAP4/MOD6/U4/J3 code
- Low void fraction in the ROSA-III (half-height) case than in the BWR/6 (full-height) case

K. Tasaka et al., "ROSA-III Experimental Program for BWR LOCA/ECCS Integral Simulation Tests," JAERI-1307 (1987)

ROSA-III Program (1978～1983)

Effect of Height Scaling on Mixture Level Transient under Depressurization





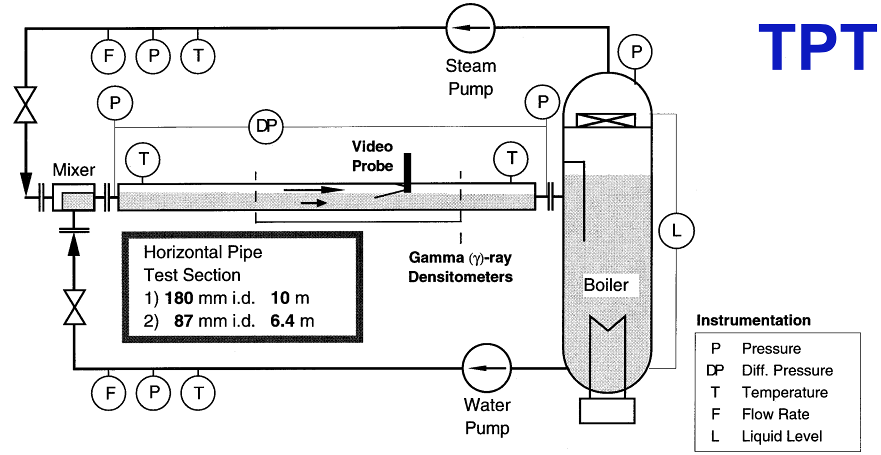
- Comparison of mixture level transient in two vertical cylinders of 10 m-high and 5 m-high
- Saturated steam-water two-phase mixture
- **RELAP4/MOD6**/U4/J3 Code calculational testing

- **Half-height Case**:
  - Lower void fraction = Lower mixture level
  - **Conservative** Higher core temperature excursion
  - Mixture level in the core, however, is influenced by coolant distribution = flow between downcomer/core through lower plenum

K. Tasaka et al., "ROSA-III Experimental Program for BWR LOCA/ECCS Integral Simulation Tests," JAERI-1307 (1987)

ROSA-IV TPTF:
Flow Regime Transition in Horizontal Steam-Water Two-phase Flows

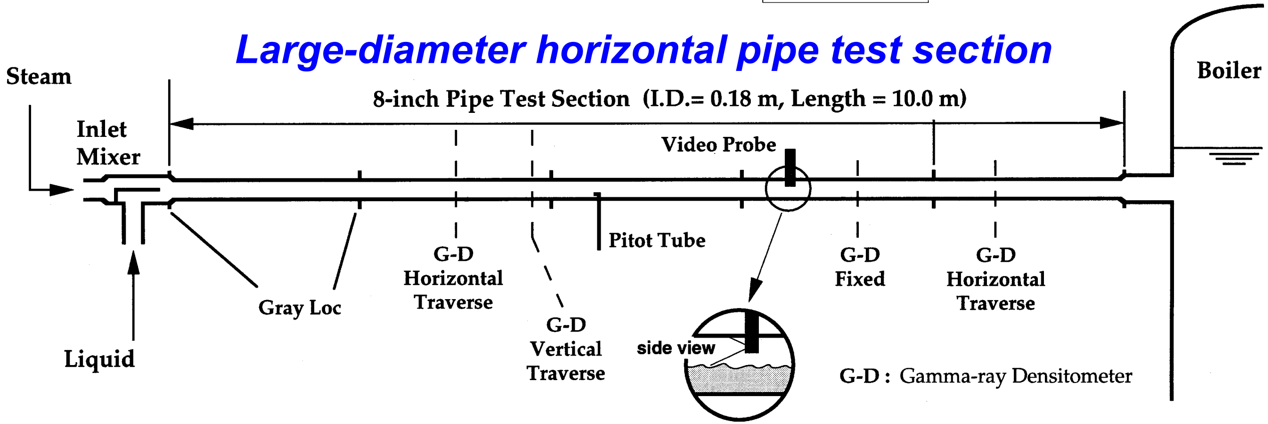
Influences of Pipe Diameter & Fluid Property



TPTF (Two-Phase Flow Test Facility)

- ✓ **Flow regime transition**
- ✓ Horizontal pipe
  Inner Diameter = 0.18 m (8-inch case)
- ✓ Steam-water two-phase flows
- ✓ Pressure ≤ 12 MPa
- ✓ Saturated temperature ≤ 325 ºC (598 K)

Large-diameter horizontal pipe test section



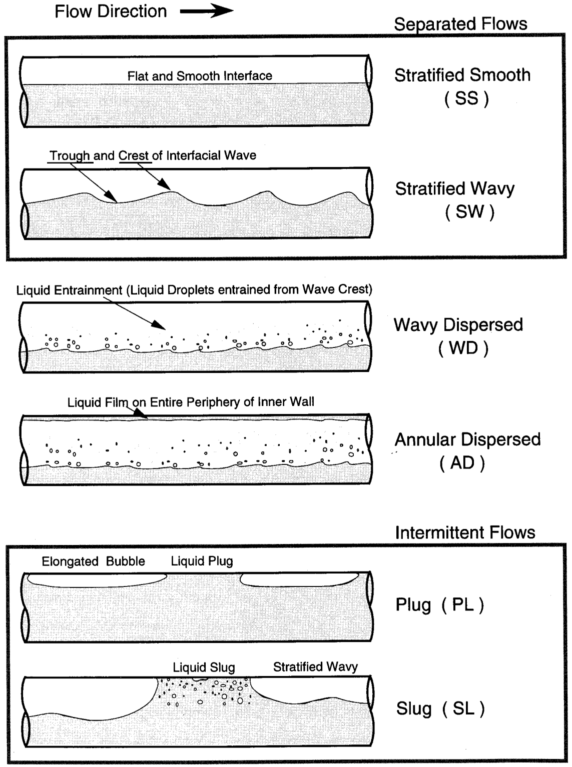
Special Meas. Instruments

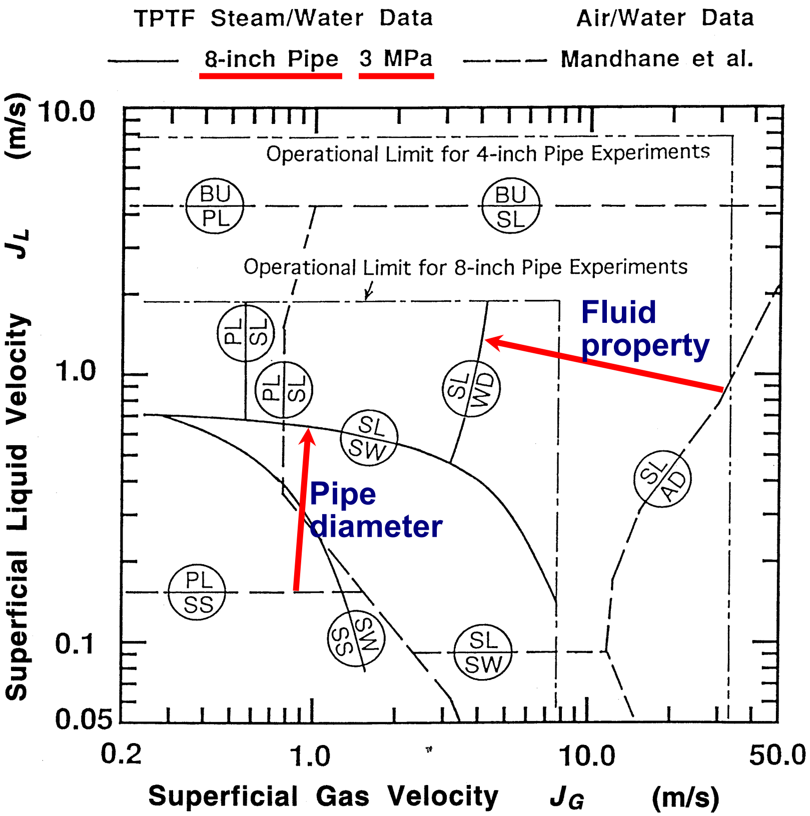
- ✓ **Video Probe**
  for flow visual observation
- ✓ **γ-ray Densitometers** (G-D)
  for liquid level measurement

H. Nakamura, "Slug Flow Transitions in Horizontal Gas/Liquid Two-phase flows," JAERI-Research 96-022 (1996)

TPTF Flow Regime Map

Typical Flow Patterns





Significant influences of Pipe Diameter and Fluid Properties
= f (System Pressure)

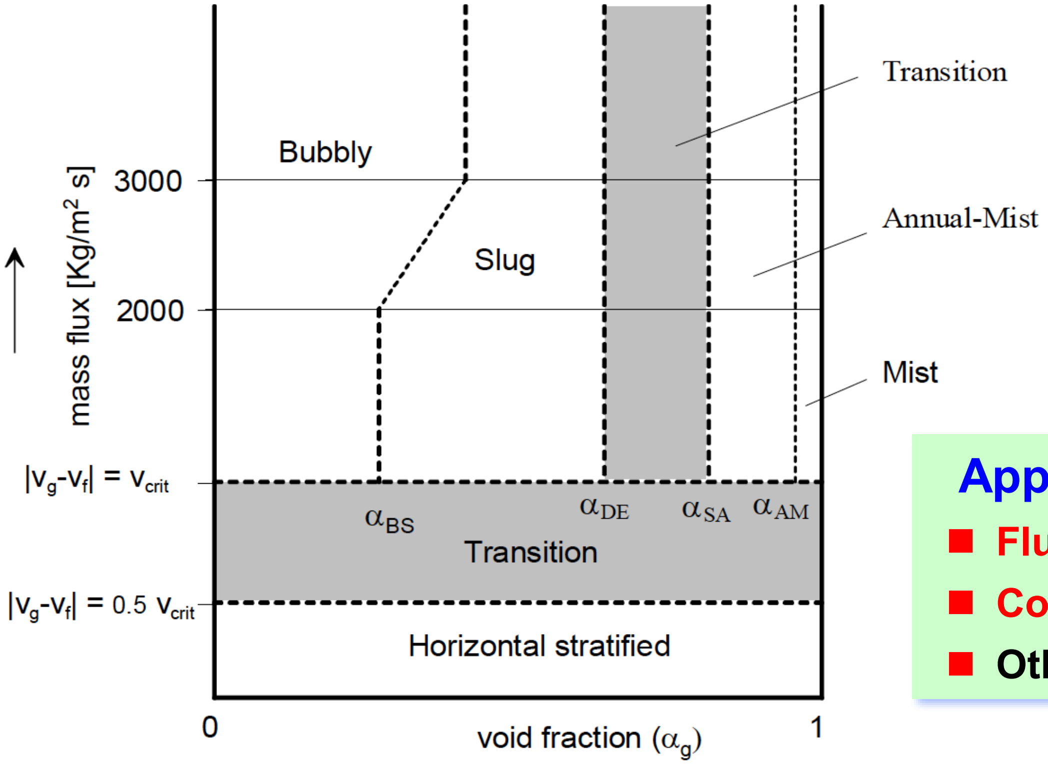
- **Changes in transition boundaries**
- **Disappearence of slug flow at P > 8.6 MPa**

How to extrapolate to NPP conditions ?

H. Nakamura, "Slug Flow Transitions in Horizontal Gas/Liquid Two-phase flows," JAERI-Research 96-022 (1996)

Simplification for Computer Code

RELAP5/MOD3 Flow Regime Map for Horizontal Flows



Applicability Universal ?

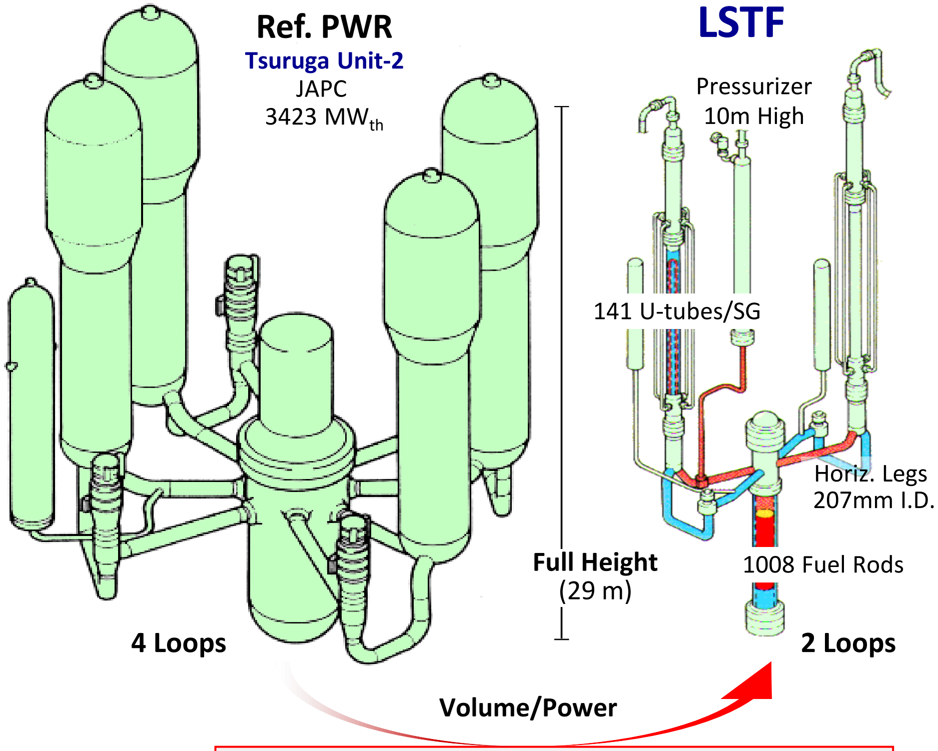
- Fluid Physical Properties
- Conduit Size & Geometry
- Other parameters

ROSA-IV LSTF: Natural Circulation (NC) (a) Non-homogeneous U-tube Behavior

Flow Reversal Tubes together with Forward Flow Tubes

LSTF

Large Scale Test Facility



LSTF for ROSA Programs in JAEA

ROSA-IV (1980-92)

SBLOCA + Transient

- '85- Shakedown
- '89- OECD **ISP-26**
- '91 **Mihama-2 SGTR** Simulation

ROSA-V (1991-)

Accident Management (AM)

Passive Safety of Next Generation LWRs

- '92- **AP600** Simulation with USNRC
- '98- **PCCS** for BWR with JAPC
- '05- **OECD-NEA ROSA Project**

Unique Features

· **Full Pressure**	· 10 MW **Core Power** (14%)
· **Full Height**	· Full + passive **ECCS**s
· **1/48 Volume**	· 20 **Break** Points
· Large-D Pipes	· 1900 **Instruments**

- Designed after **TMI-2** (1979) >> Full Height
- **SBLOCA**s & **Operational Transients**
- **SESAR/SFEAR** (2007) - ranking 2.9 (highest group)

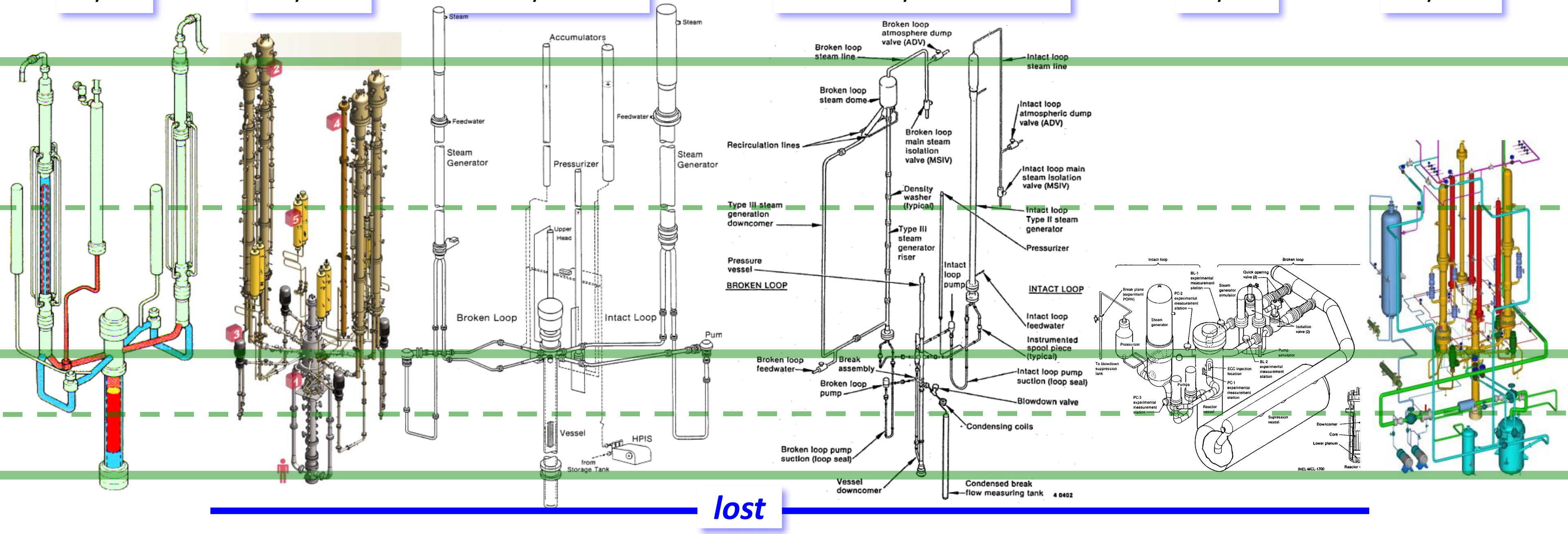
OECD NEA

The ROSA-V Group, "ROSA-V Large Scale Test Facility (LSTF) System Description for the 3rd & 4th Simulated Fuel Assemblies," JAERI-Tech 2003-037

Integral Effect Test (IET) Facilities with different Scaling Methods

➢ **Linear, Volume/Power, H2TS, Three-Level**

LSTF	PKL-III	LOBI-MOD2	Semiscale Mod 2C	LOFT	ATLAS
1/48	1/145	1/700	1/1530	1/50	1/288



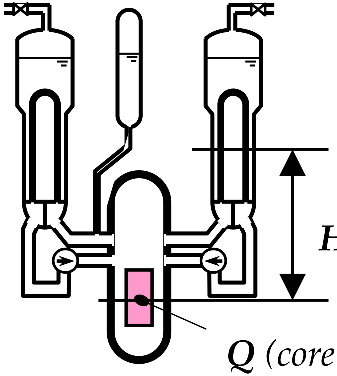
❖ ***Different scales may co-exist in one facility.*** (*distortion, compromise, ..*)

❖ ***Direct extrapolation of the results into reference LWR is NOT feasible.***

General Correlation of Loop Flow Rate

PWR Natural Circulation

- **Key mechanism for core cooling when forced circulation is lost.**
- **TMI-2 accident (**severe accident in 1979**) made us notice the importance.**
- **Three modes: single-phase liquid, two-phase, reflux cooling**
  depending on primary coolant inventory
- **Time Scaling**: $Time_{transient}$ in proportion to $\sqrt{Facility\ Height\ Scaling}$ **<< Ishii's 3-level scaling**



Single-phase liquid natural circulation

✓ Flow rate may be correctly predicted by this correlation.

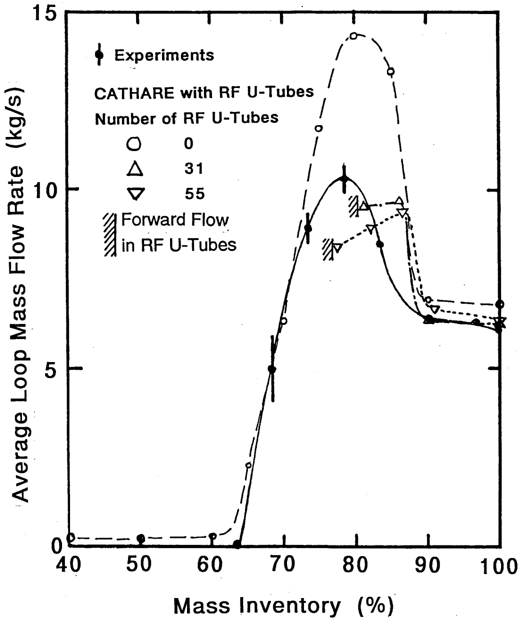
$$W = \left(\frac{2gQ\beta HA^2}{c\rho_0 K} \right)^{1/3}$$

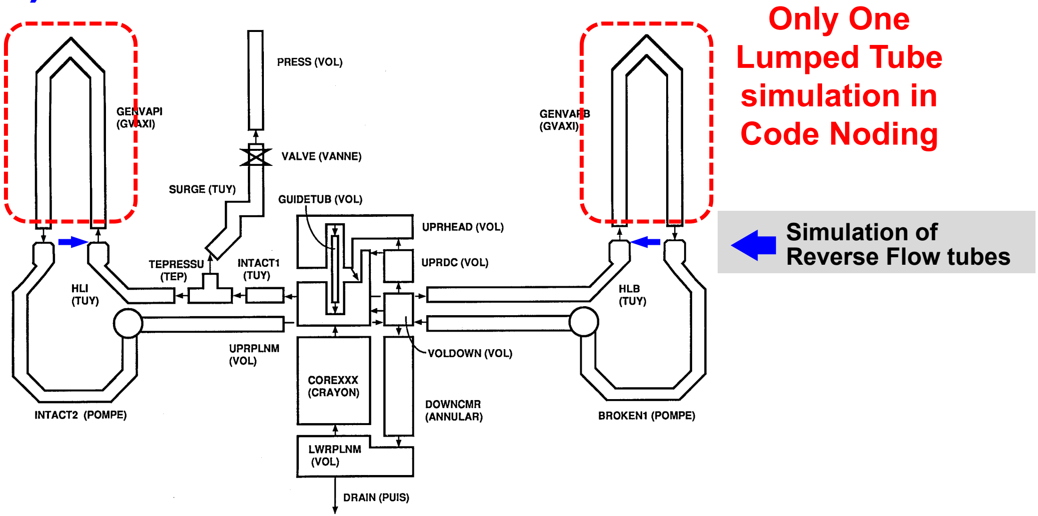
✓ However, calculated flow rate is usually over-estimated because **reverse flow tubes** exist. ➡ **Next Page**

Y. Zvirin,"A review of natural circulation loops in pressurized water reactors and other systems," Nucl. Eng. Des. 67 (1982), pp. 203–225.

An Example of Computer Code Analysis

ROSA/LSTF (core power = 2%)





LSTF Noding Schematic

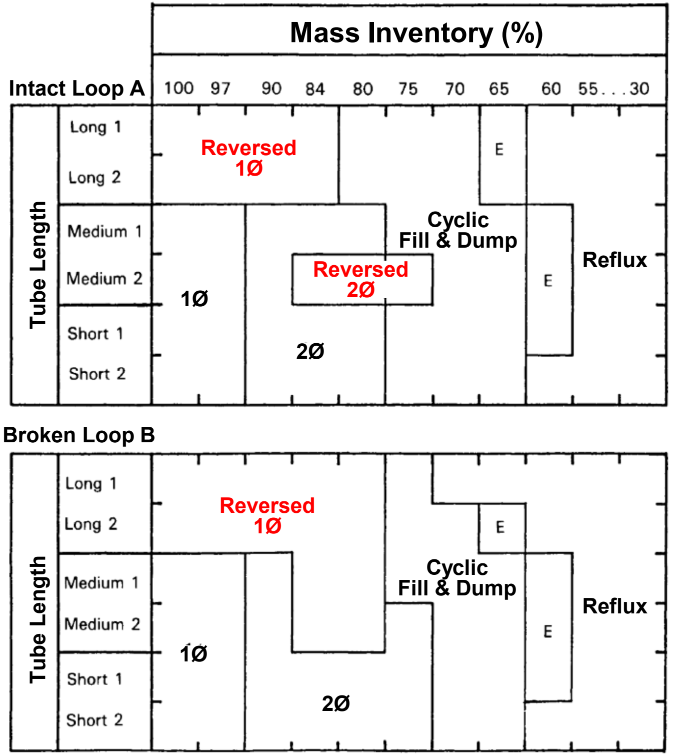
CATHARE-1 Trial Calc./ 1987

- ✓ **Flow reversal** in small fraction of tubes
- ✓ Two modes: Single-phase liquid & two-phase flow NCs
- ✓ Number of reverse-flow tubes may change during transient

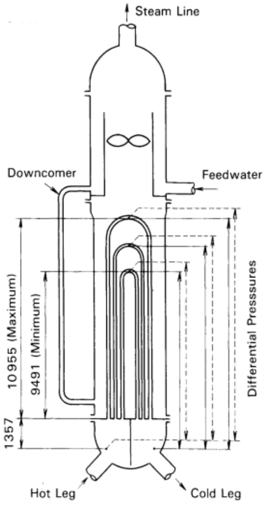
- **Loop flow rate** decreases
- **Multiple-tube simulation** necessary to properly simulate whole transient

LSTF Experiment Observation

U-tube Flow Modes



- **2% Core Power – Stepwise Mass Inventory Reduction**
- **Flow Reversal** in **Long** & **Medium tubes**
- **1Ø reverse** coexisted with **2Ø forward**
- **Coolant temp.** and/or **void fraction** in **SG inlet plenum** decreases, causing
  decrease in the natural circulation flow rate
- **Flow reversal** also appeared in **Semiscale, LOBI, PKL**
  - ➢ **Multiple-tube model** necessary in **Code Noding**



LSTF SG Measurement

- **6 instrumented / 141 U-Tubes/SG in both loops**
  Long x 2, Medium x 2, Short x2

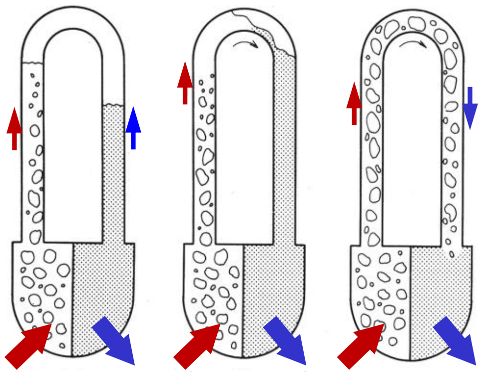
Number of U-Tubes in SG:

Ref. PWR	3384 tubes/SG in Ref. 4 Loop PWR	
LOBI	8 + 24 tubes/SG 1- & 3-loop	2-loop simulation of 4-Loop PWR
Semiscale	2 + 6 tubes/SG 1- & 3-loop	

Y. Kukita et al., “Nonuniform Steam Generator U-Tube Flow Distribution During Natural Circulation Tests in ROSA-IV Large Scale Test Facility,” Nucl. Sci. Engng. 99 (1988) 289

ROSA-IV LSTF: Natural Circulation (NC)
(b) Flow Oscillation in 2Ø Flow NC mode

Observation in ROSA/LSTF Experiments

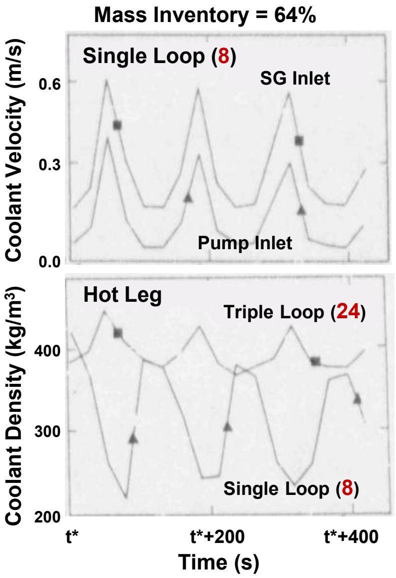


- **Fill and Dump phenomena**
  ("syphon condensation")
- **Oscillation happens in each U-tube,**
  ***inherent to* 2Ø natural circulation mode**
- **Narrow range of primary mass inventory**
  = around 55 to 80 %

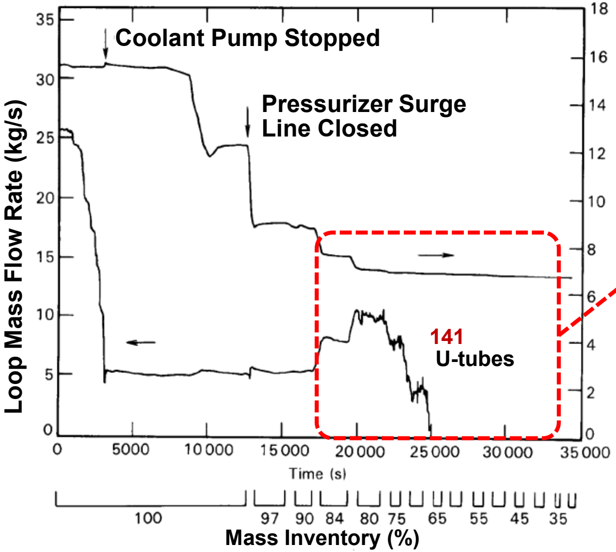
- **Fluctuation is resulted in natural circulation flow**
- **In-phase oscillation among U-tubes may cause large fluctuation in loop flows.**
  When the IET facility has *a small number of U-tubes*,
  even *loop-to-loop oscillation* may emerge.
- **However, the oscillation will be greatly attenuated** (even to negligible) **in the reactor,** *as observed in **LSTF** NC experiments*, **because fill and dump in individual tube is mutually compensated among *many U-tubes*.**

Y. Kukita et al., "Nonuniform Steam Generator U-Tube Flow Distribution During Natural Circulation Tests in ROSA-IV Large Scale Test Facility," Nucl. Sci. Engng. 99 (1988) 289

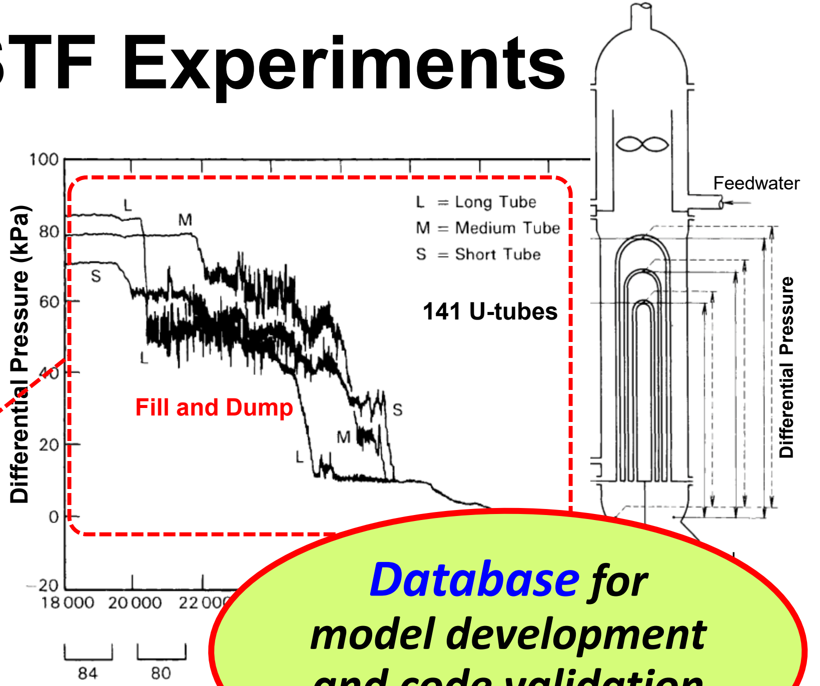
Observation in ROSA/LSTF Experiments



LOBI NC Data
(A2-77A)



Primary Pressure and **Loop A Flow Rate**
(2% core power natural circulation)



Empty... SG Inlet P...

Database* for model development and code validation *extremely important

Oscillation during 2Ø NC mode:

- **Significant (LOBI) vs. Small (LSTF)**
- **Narrow Range (around 55 to 80%) in Primary Mass Inventory**
- **Structure of SG Inlet and Hot Leg Nozzle**

*We are trying to **extrapolate to Reference Reactor (3400 tubes)** with **this info.-based** safety assessment code*

- Y. Kukita et al., “Nonuniform Steam Generator U-Tube Flow Distribution During Natural Circulation Tests in ROSA-IV Large Scale Test Facility,” Nucl. Sci. Engng. 99 (1988) 289
- F. D'Auria and G. M. Galassi, “Relevant Results in Analysis of LOBI/MOD2 Natural Circulation Experiment A2-77A,” NUREG/IA-0084, NT 163(90), U.S. NRC (1992)

ROSA-V LSTF / OECD/NEA Joint Project ROSA-2:

Counterpart Testing with PKL

IETs *(Integral Effect Tests) have made substantial contributions* ***to promote understanding*** *NPP accident responses.*

However, Typicality of observed responses *at* ***one facility*** *may be questioned due to inherent* ***scaling distortions*** *resulted from construction compromises and simulation constraints.* ***So, ...***

Counterpart Tests

NEA/CSNI/R (96) 17

Code assessment is performed, to assure ***adequacy*** *of the system code, over a wide range of data from test facilities having* ***different scaling*** *and/or* ***design concepts****.*

It is beneficial to ***validate code scalability*** *against a set of data from different facilities with* ***similar initial & boundary conditions (IBCs)****.*

- ***Rigorous definitions*** *made for* ***IBCs*** *of tests at* ***different facilities***
- ***Many counterpart tests have thus been performed worldwide so far ...***

Counterpart Tests in OECD/NEA ROSA-2 Project

Collaboration with PKL-2 Project

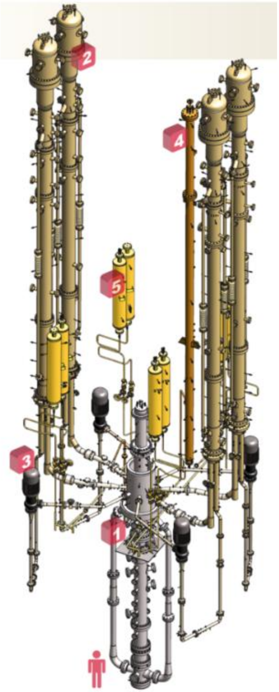
1. **Differences** in facility design concept, size and pressure range
2. **Mutual interest** on validity of **CET** (core exit temperature) as the key parameter to start AM action
3. **In-depth discussions** on synchronization of IBCs to observe core uncovery adequate to confirm effectiveness of AM

Test Conditions

1. **Hot leg SB-LOCA (** 1.5%, total-failure of HPI, for **CET** response **)**
   - Superheated steam during core boil-off
   - No reflux coolant
2. **A set of data from PKL & LSTF to observe influences from Differences**
   - **Two-pressure approach for LSTF** under High + Low pressures
   - **Similar IBCs** at Low-Pressure portion
   - **PKL**: G 7.1 in PKL-2 Project vs. **LSTF**: Test-3 in ROSA-2 Project

Counterpart Tests in OECD/NEA ROSA-2 Project

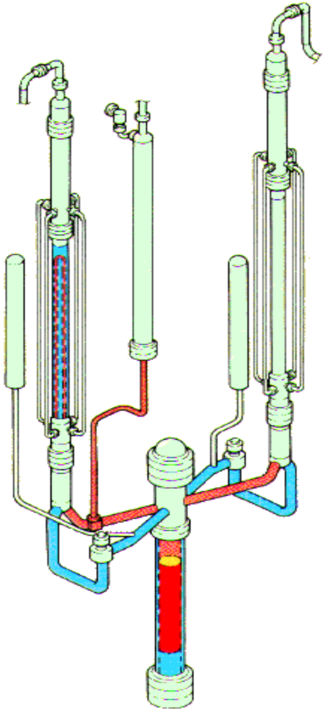
PKL-III



Facility Comparison

Item	PKL	LSTF
Height	Full	Full
Volumetric Scaling	1/145	1/48
No. of Loops	4	2
PV downcomer	**Double-pipe**	**Cylindrical**
SG U-tubes / SG	30	141
Pressure	**≤ 4.5 MPa**	**Full**
Core Power (max)	10%	14%
Axial profile	**Flat**	**Chopped cosine**
Radial profile	3-region	3-region
ECCS	Full	Full
Special Measurement	Boron sensor	Video probe O_2 gas sensor
No. of Instrument	about 1070	about 1600

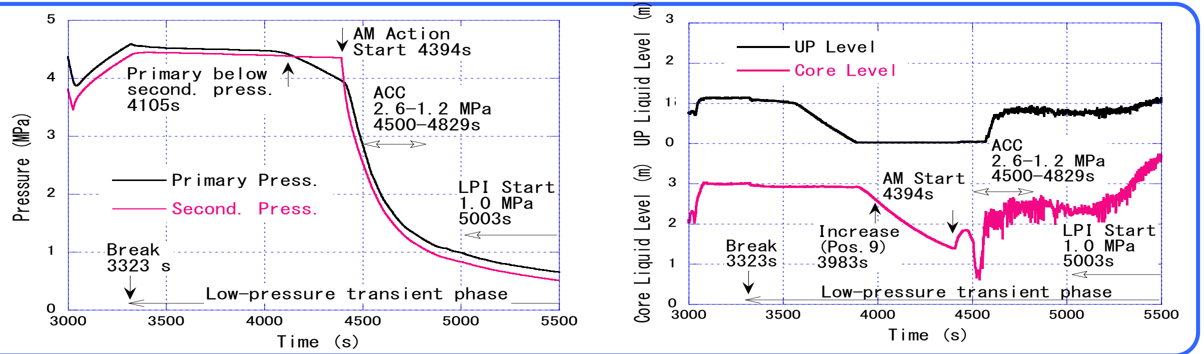
LSTF



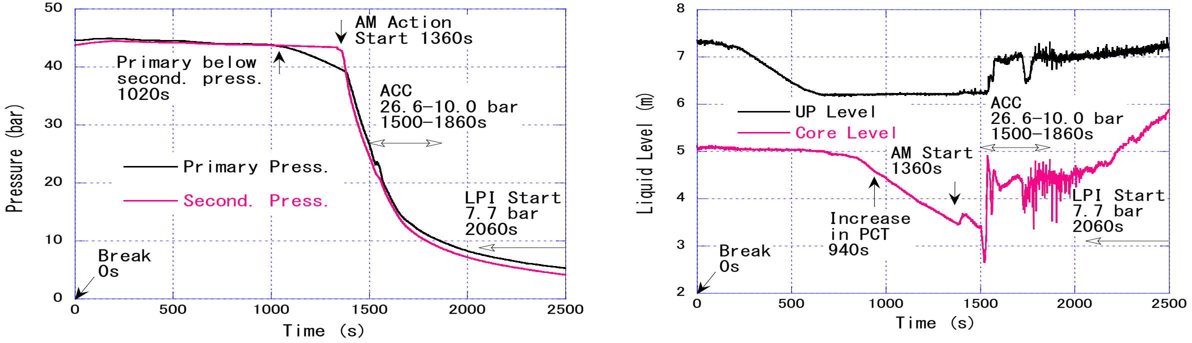
Counterpart Tests in OECD/NEA ROSA-2 Project

Primary & Secondary Pressures | **Liquid Levels in Core & Upper Plenum**

LSTF



PKL

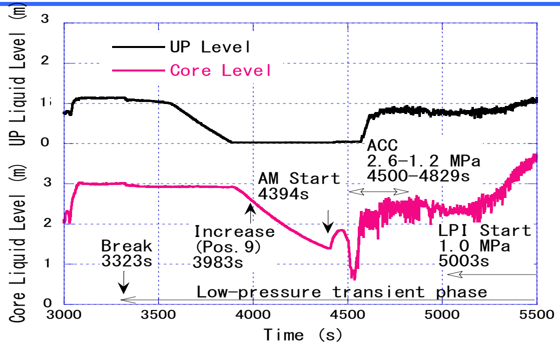


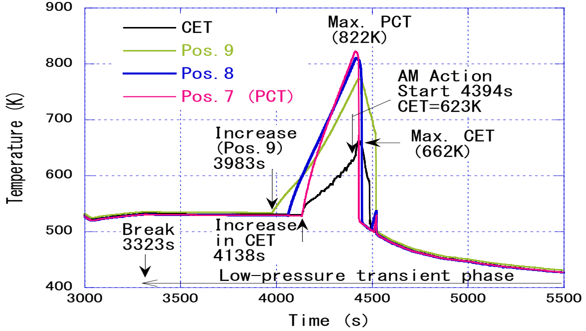
- **Almost equal IBCs** at Low-Pressure portion
- **Very similar liquid level response** in core

Counterpart Tests in OECD/NEA ROSA-2 Project

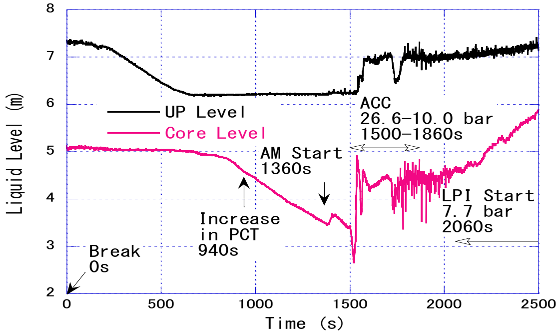
Liquid Levels in Core & UP | **Simulated Fuel Cladding Temperature**

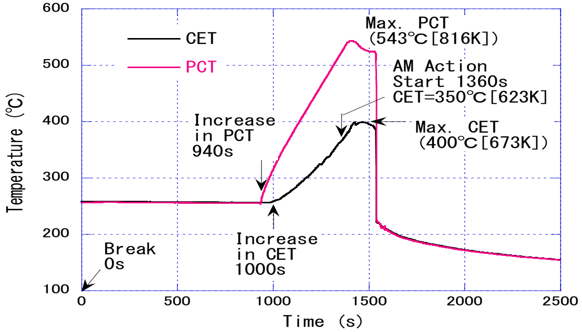
LSTF





PKL



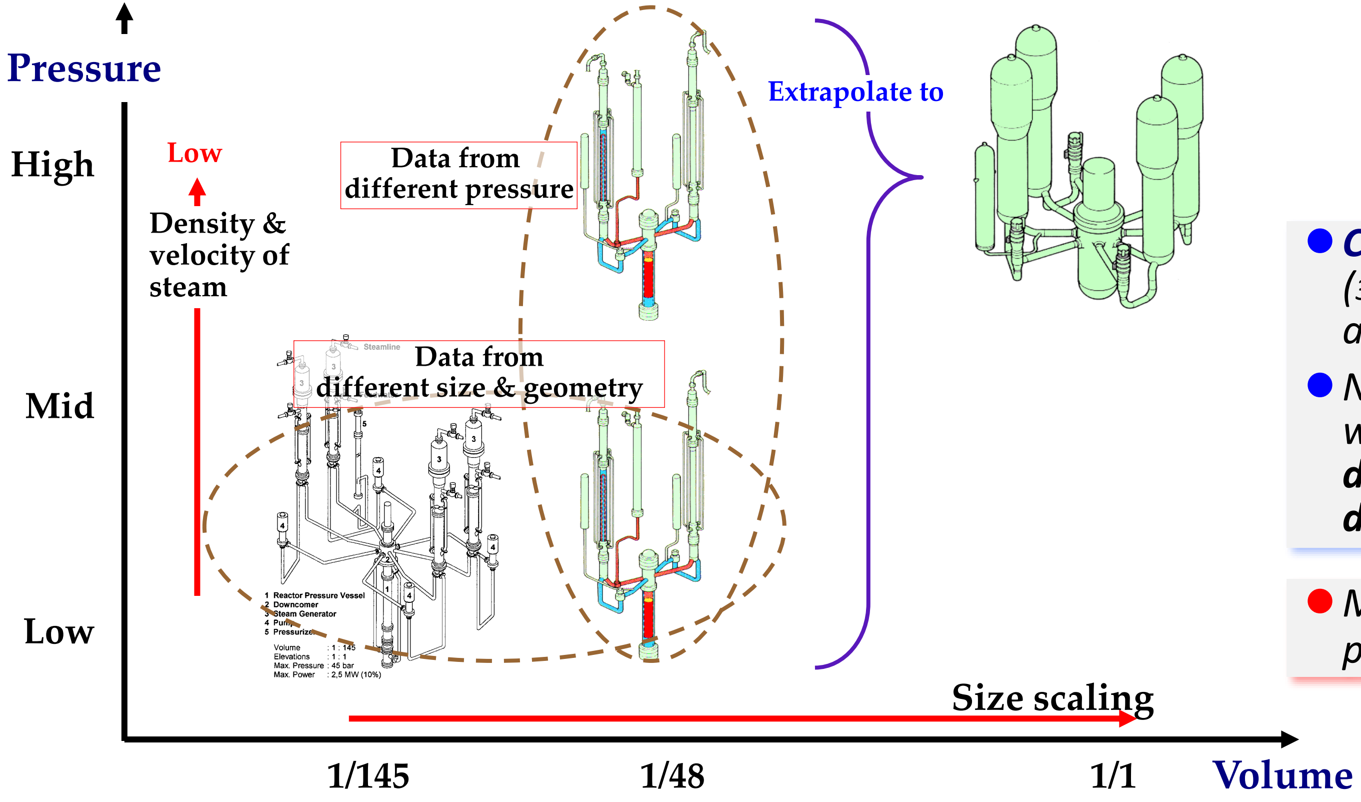


PCT: Peak Cladding Temperature

CET: Core Exit Temperature (T/C)

- **Similar liquid level response** resulted in **similar core heat-up response**,
- **Similar steam flow** may prevail in **different-scaled pressure vessel**

Counterpart Testing between PKL & LSTF



CET: Core Exit Temperature (T/C)

- ***CET**-related phenomena (3-D steam flow at core exit) at rather high pressure*
- *Need data from facilities with **different size scaling** and **different structure shape***

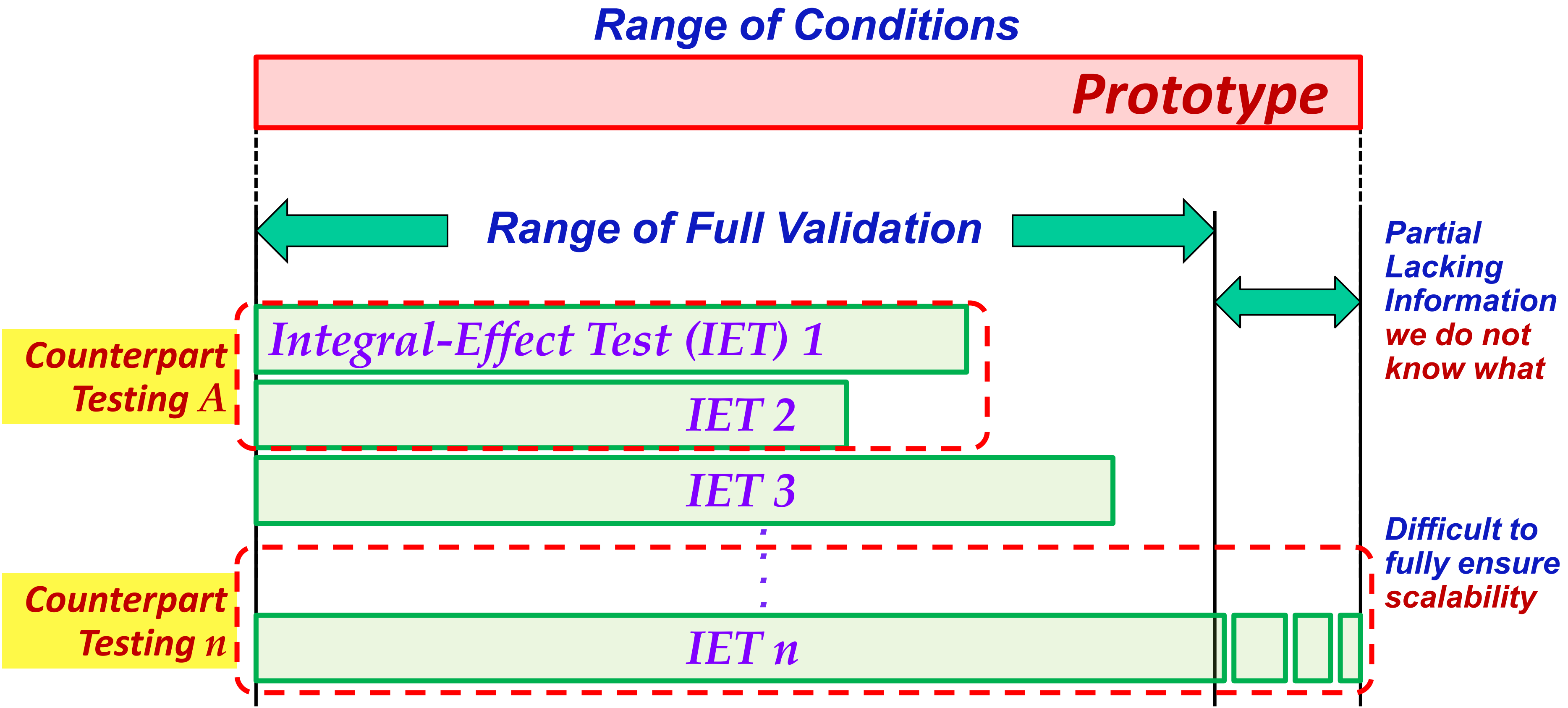
- *More counterpart testings planned at past **PKL-3***

Notes on Counterpart Tests

- *Permits to experimentally characterize observed phenomena at different scales, thus to approach **scaling of phenomena** via **phenomenological considerations.***
- *However, the **characterization is "restricted"** within the experimental range and facility maximum scale. Thus, **Direct extrapolation** of the data to prototype is in general **not feasible**.*
- ***PKL-ROSA/LSTF counterpart tests** performed further on SBLOCAs etc. in OECD/NEA **PKL-3 Project**, as well as **ATLAS-ROSA/LSTF**.*
- ***SET-SET** and even **IET-SET** should be possible as counterpart testing, good to extend the possibility of experiment range.*

- ***F. Mascari et al.**, "Scaling Issues for the Experimental Characterization of Reactor Coolant System in Integral Test Facilities and Role of System Code as Extrapolation Tool," Proc. of **NURETH-16, 13916, Chicago (2015)***
- *"Scaling in System Thermal-Hydraulics Applications to Nuclear Reactor Safety and Design: a State-of-the-Art Report," **NEA/CSNI/R(2016)14***

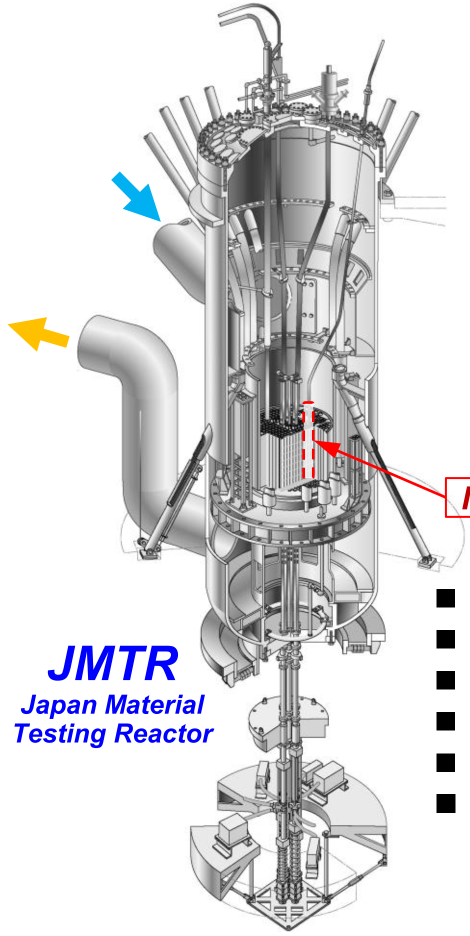
Scaling (Validation) Limit



Personal Experiences in Scaling

… and some others

Radiation-induced Surface Activation (RISA)



- **Gamma ray photochemically activates metal surface** (oxidized)
  = **Surface** to be covered by **radical,** greatly enhances **wettability**
- **Surface** becomes **"super wet"** for **water**
  = **Condition** of **Nucleate Boiling should change**
  = **Surface superheating** during boiling may **rise very high**
- **Nuclear fuel** emits **intense gamma ray** ***during*** **&** ***after*** **the reactor operation** **>> Fuel boiling condition may change**
- **JMTR was used to confirm** ***the last experiment/operation for JMTR***

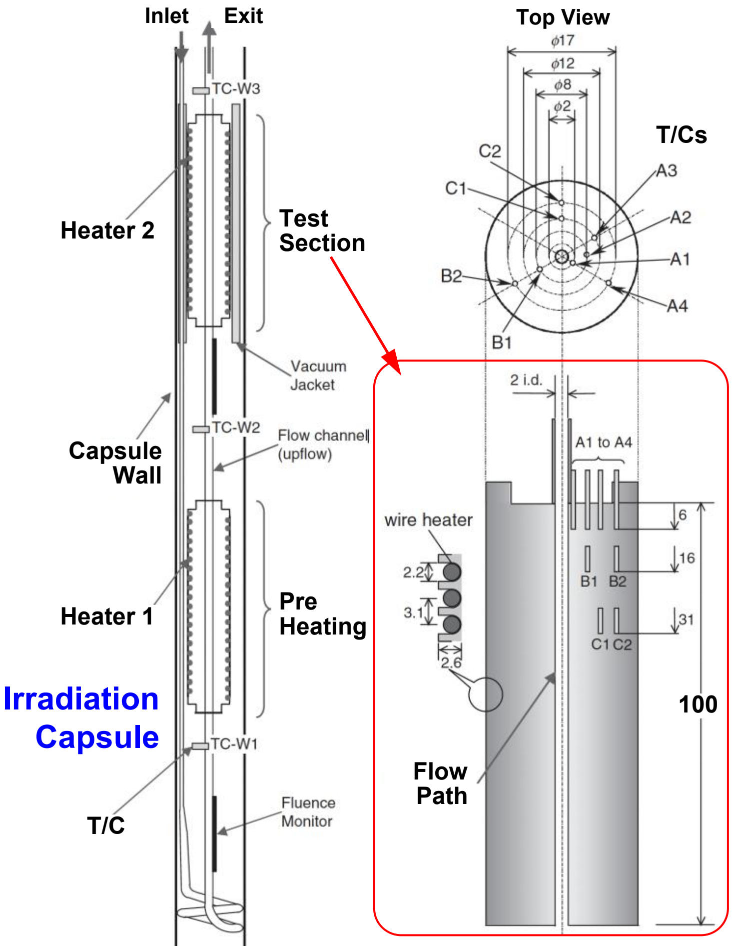
Core Power	**50 MW**
Pressure	**1.5 MPa**
Temperature (Inlet/Outlet)	**49/56 ºC**
Coolant flow rate	**6000 m^3/h**
Neutron flux (thermal)	about **1.2 x 10^{13} cm^{-2}/s**
Gamma-ray dose rate	**540 kGy/h***

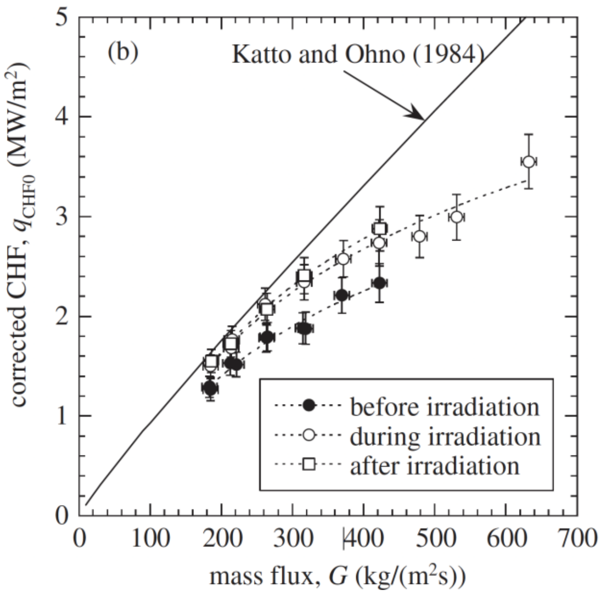
at *Capsule*

*estimated from gamma heating rate

Y. Sibamoto et al., "In-pile Experiment in JMTR on the Radiation Induced Surface Activation (RISA) Effect on Flow-boiling Heat Transfer," J. Nucl. Sci. and Technol. 44 (2) (2007) 183

Radiation-induced Surface Activation (RISA)





Obtained CHF Data
(Compensated for inlet subcooling effect)

- **RISA effect** was directly evaluated **before, during and after** the intense gamma-ray irradiation.
- **During & after irradiation, CHF increased by 17%** on average, from **CHF before irradiation.**
- **Wall superheat** at **subcritical heat fluxes became greater** than that before irradiation, in general.

Material	**SUS361L**
Flow Path i.d.	**2 mm**
Pressure	**0.7 - 1.0 MPa**
Outlet Temperature	**about 147 ºC**
Inlet Mass Flux	**186 - 637 kg/(m²s)**
Heat Flux	**0 - 6.4 MW/m²**

Y. Sibamoto et al., "In-pile Experiment in JMTR on the Radiation Induced Surface Activation (RISA) Effect on Flow-boiling Heat Transfer," J. Nucl. Sci. and Technol. 44 (2) (2007) 183

Note: **JMTR** was closed eternally after this experiment

Epilogue

Do you know what you do not know ?

Summary

Experiences on Scaling

- **Ultimate Goal**: ***Reactor Safety Assessment** under **Prototypical Conditions***
- **Computer Code & Experiment = *Complementary* ➡ *Both Necessary***
- **Safety Assessment Codes go even *beyond Scaling Limit (Virtual ?)***
  ➡ ***need to remind modeling basis at Goal Conditions***

Examples

Prototypical Phenomena in ROSA Programs

- **ECCS Problem**
  – PWR ECC Bypass (ROSA-II)
- **Half-height Facility (ROSA-III)**
- **Horizontal Flow Regime Transition**
  (*Influences of Fluid Physical Properties*
  + Channel Geometry, **ROSA-IV TPTF)**
- **PWR Natural Circulation**
  (ROSA-IV LSTF – *Flows in SG U-tubes*)
  (a) Non-homogeneous / Flow Reversal
  (b) Flow oscillation in 2Ø NC mode
- **Counterpart Testing with PKL / LSTF**
  Scaling Limit ?? (NEA ROSA-2 JP)

Other Subject (Prototypical)

- **CHF increase** due to **Radiation-induced Surface Activation (RISA)**

Final Notes

Experiences on Scaling

- **Phenomena Scaling is one of the major Uncertainty Sources in safety assessment results, *while noticed or not* ..**
- **Methods to go beyond the scaling limit** (*the world beyond the knowledge range*) → ***NOT suggested***
- **A Solution (possibility ?)**

  Combination of **Prototype Experiments** and **T/H & CFD Analyses**

  - ***Mutual interactions in an experiment with different scaling may need detailed validity confirmation (*** *in a 3-D phenomenology-oriented way* ***)***
  - ***T/H codes can go beyond the limit.*** → ***Firm Confirmation Necessary !!***
  - ***CFD may have a possibility to provide an insight to go beyond the limit with detailed/precise local flow analyses.***

Thank you for your kind attention !!

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