Knowledge Management from Joyo and Monju to Next SFR and Researchers

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Contents

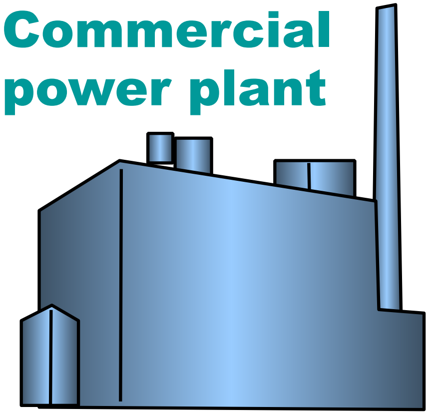
- Experiences and Data in
  - Joyo
  - Monju
- Ideas on Knowledge Management

Joyo and Monju toward Commercial Reactors

Prototype fast breeder reactor power plant MONJU 280MWe (714MWt)

To confirm the performance of sodium cooled FBR power plant

Commercial power plant

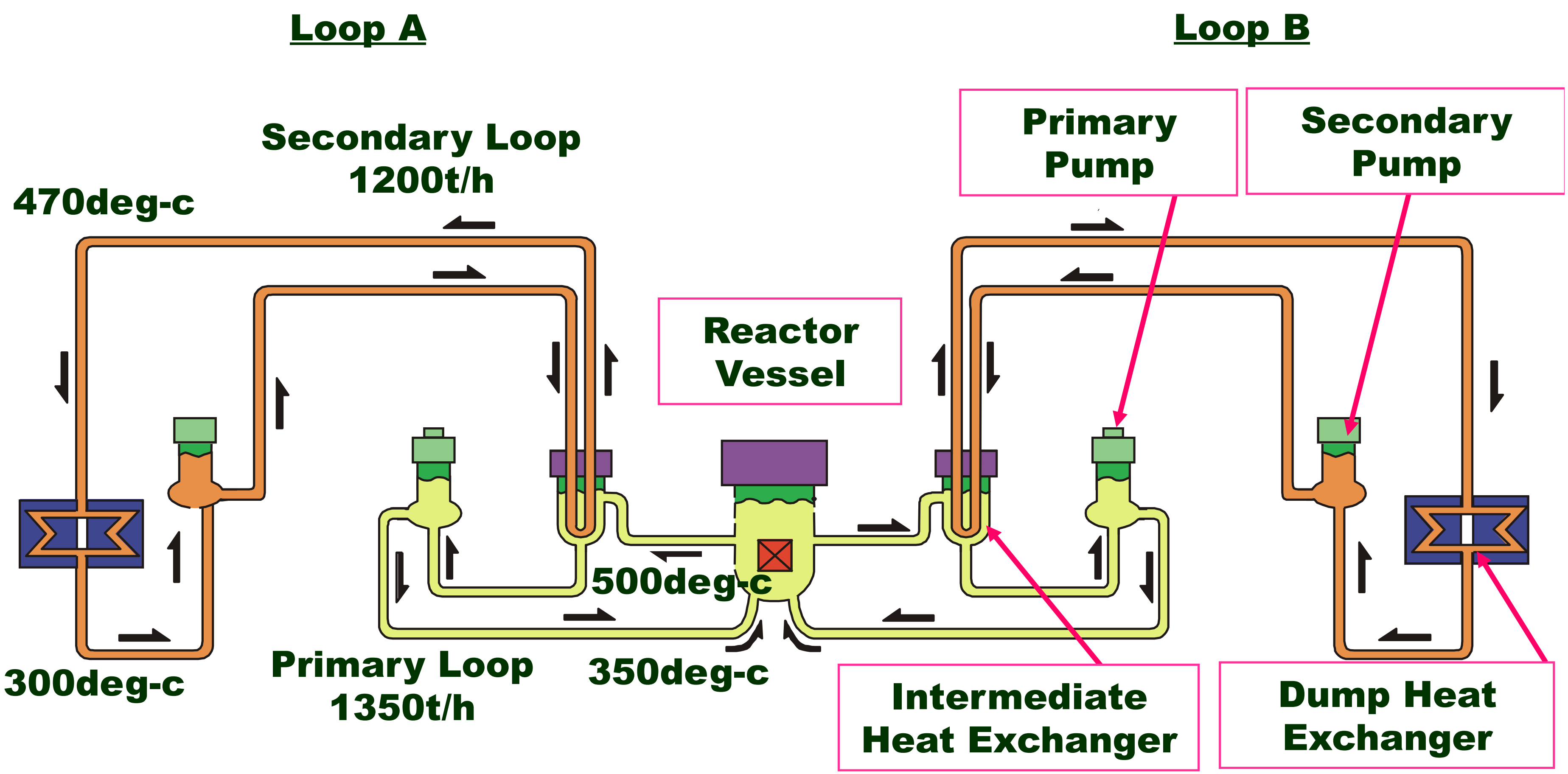


Experimental fast reactor JOYO

140MWt

- **To confirm the principle of sodium cooled FBR**
- **To establish operation, maintenance technology**
- **Irradiation for fuels and materials development using fast neutron field**

Joyo Heat Transport Systems



Cooling system type : Loop type
Number of main cooling system : 2
Heat Removal : Air Cooling

Progress History of Joyo

MK-III

2004.5~ Rated Power Operation

2003.7 First Criticality of MK-III Core

MK-III Renovation

MK-II

2000.6

- Carbide and Nitride Fuel Irradiation (Collaboration with JAERI) □
- Power-to-Melt Test (PTM) □
- Fuel Failure Simulation Test
- High Burn-up Test (Peak Burn-up of 144 GWd/t, Collaboration with CEA France)
- Served Mainly as an Irradiation Facility for FBR Fuel and Material

1982.7 First Criticality of MK-II Core

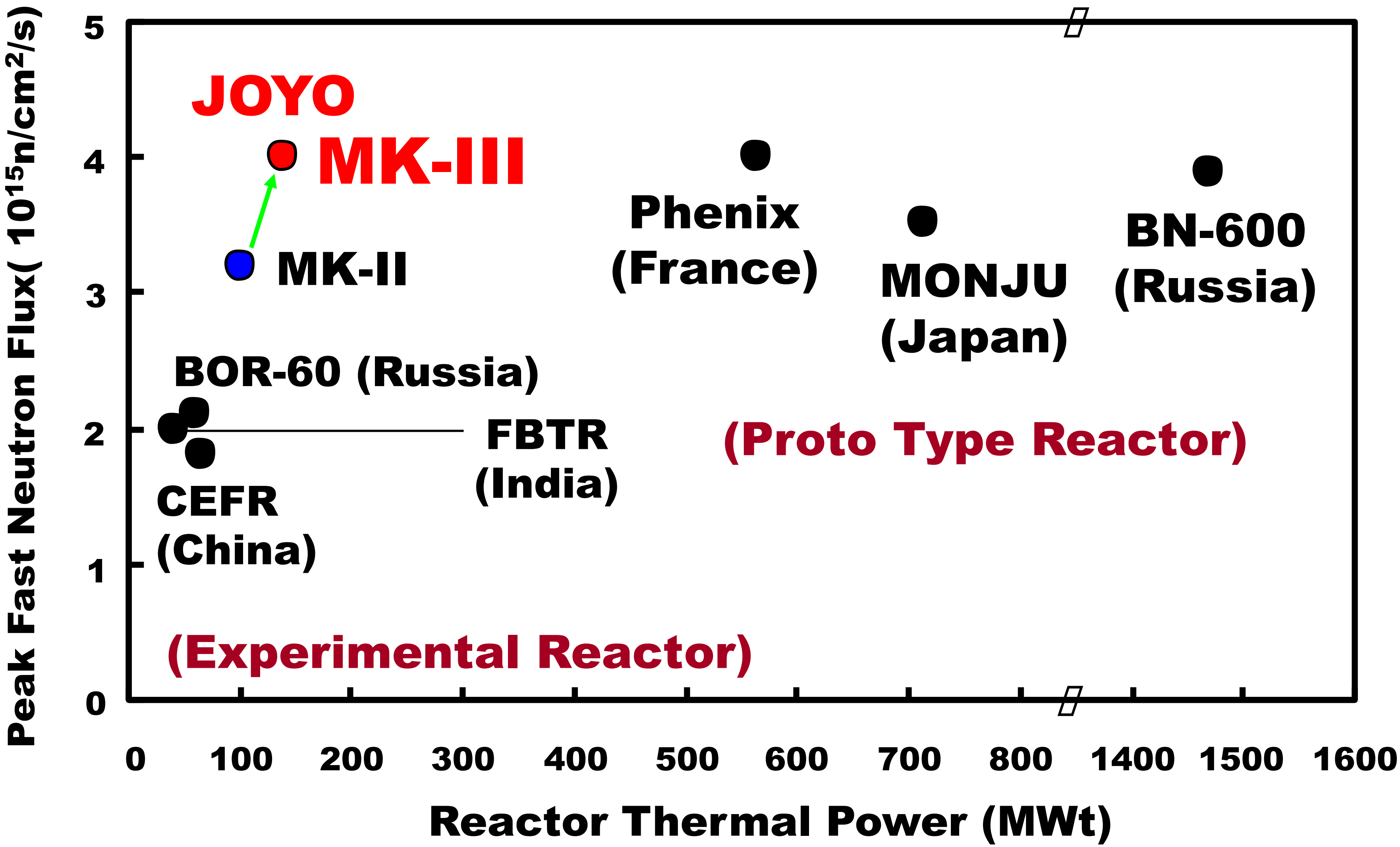
MK-I

1981

- Natural Circulation Test
- Confirm Breeding Ratio
- Accumulate Technical Experience Through Planning, Construction and Operation

1977.4 Attain First Criticality

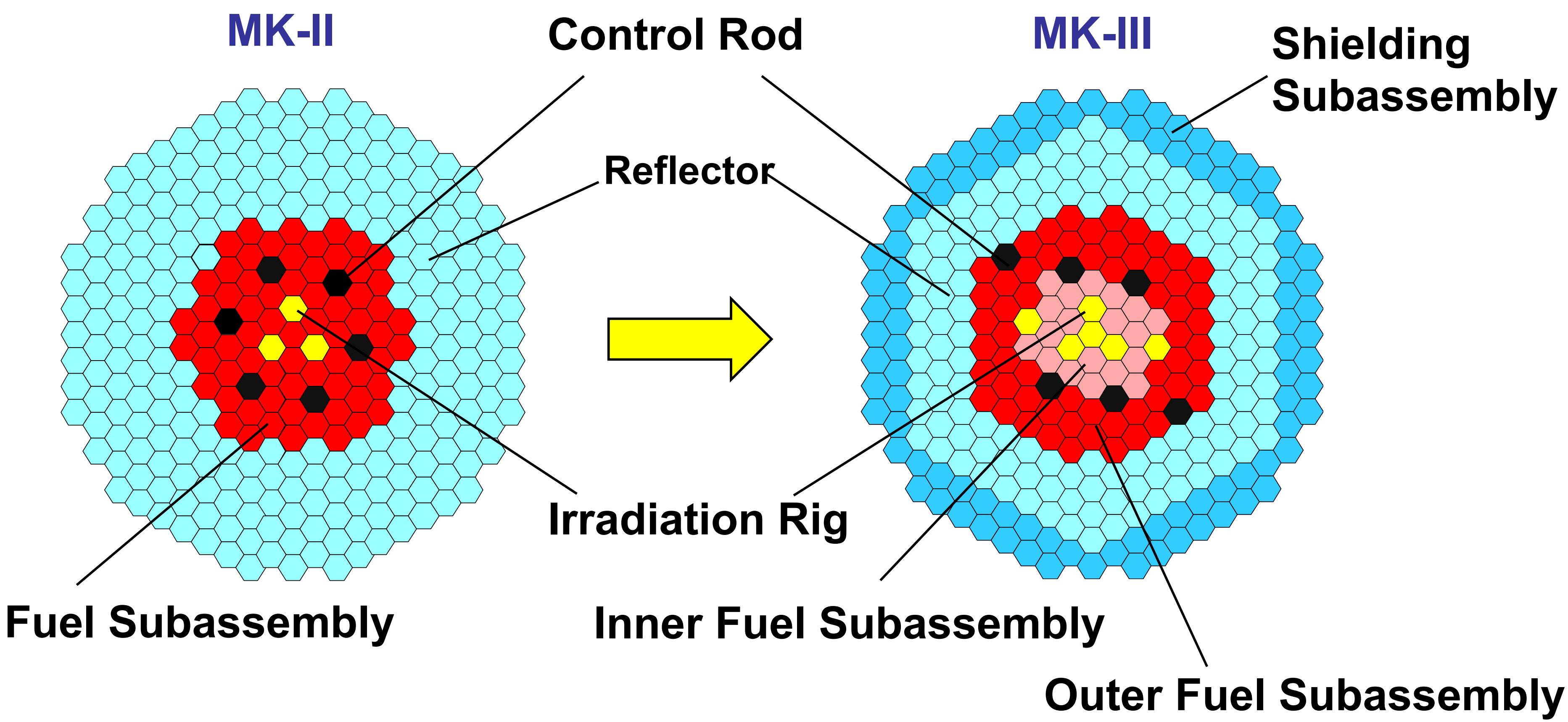
Reactor Power and Neutron flux for Irradiation



Modification work from Mk-II to Mk-III as significant Knowledge

- Core
  - Full Power: 100MW to 140 MW
  - Higher Irradiation Capability
- Cooling System
  - Exchange of Main Components
    - IHX
    - Air cooler of Main heat sink (Damp Heat Exchanger)
    - Motor of Pumps in primary and secondary loops

Core Modification



• Reactor Thermal Power	100MWt	→	140MWt
• Number of Fuel Subassemblies (Max.)	67	→	85
• Arrangement of Control Rod	6 Control Rods in 3rd Row	→	4 Control Rods in 3rd Row 2 Control Rods in 5th Row
• Core Height	55 cm	→	50 cm

Modification in Cooling System

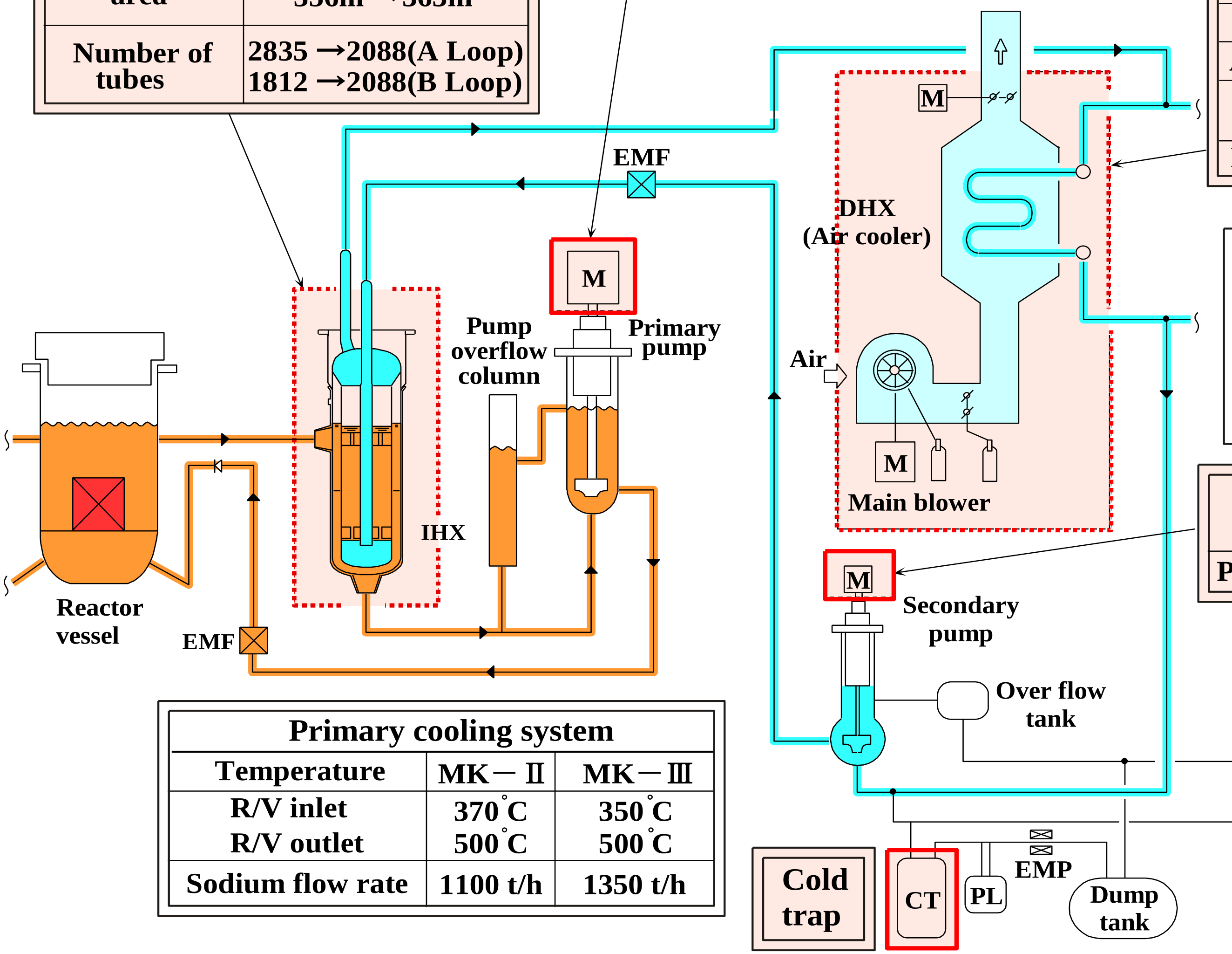
IHX	
Capacity	50MWt→70MWt
Heat transfer area	$352m^2$→$363m^2$ $356m^2$→$363m^2$
Number of tubes	2835 →2088(A Loop) 1812 →2088(B Loop)

Primary main pump motor and flow control system	
Pump motor capacity	330kW→330kW
Pony motor capacity	2.5kW → 2.5kW

DHX	
Capacity	25MWt→35MWt
Tube type	⊂→Σ
Heat transfer area	$1250m^2$→$2400m^2$
Air flow rate(m^3/min)	7400→7700
Air temperature (DHX outlet)	220 ˚C→280 ˚C
Blower motor power	400kW→710kW

Secondary cooling system		
Temperature	MK－Ⅱ	MK－Ⅲ
DHX inlet DHX outlet	470 ˚C 340 ˚C	470 ˚C 300 ˚C
Sodium flow rate	1100 t/h	1200 t/h

Secondary main pump motor and flow control system	
Pump motor capacity	180kW → 220kW

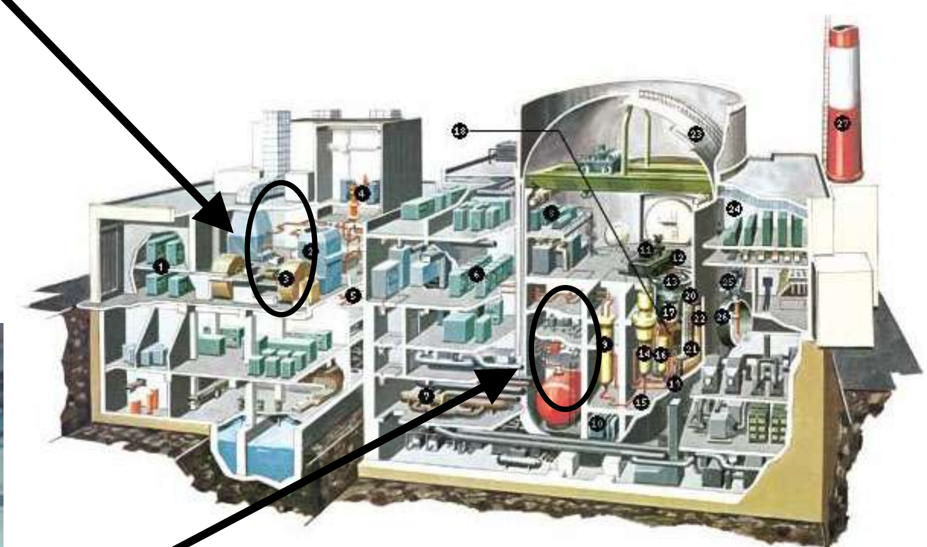


Primary cooling system		
Temperature	MK－Ⅱ	MK－Ⅲ
R/V inlet R/V outlet	370 ˚C 500 ˚C	350 ˚C 500 ˚C
Sodium flow rate	1100 t/h	1350 t/h

Renewal components in cooling system

Large Components replaced

Dump Heat Exchanger
(DHX) × 4



Intermediate Heat Exchanger
(IHX) × 2

Difficulties and Management points in Cooling System Modification

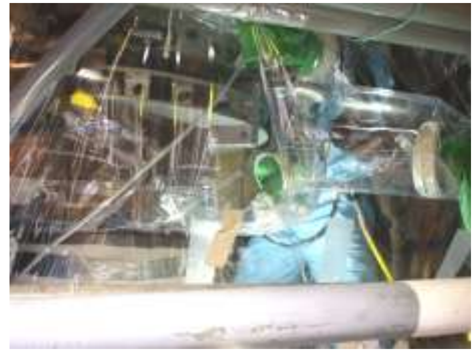
- Renewal of the large sodium components such as IHX / DHX was the first experience in Japan.
- Radioactive sodium remained in IHX after Draining.
  - Radioactive sodium and CP such as ^{60}Co and ^{54}Mn
- Need to manage modification work under high radiation dose rate environment.
- Sodium purity control
  - Prevention of impurity (oxygen) ingress to sodium system

Cutting the double walled primary loop piping connected to IHX

Set a tool

Cut off
the Outer Piping

Cut the surface
of inner piping



Filled by Ar Gas

Cut off the Inner Piping

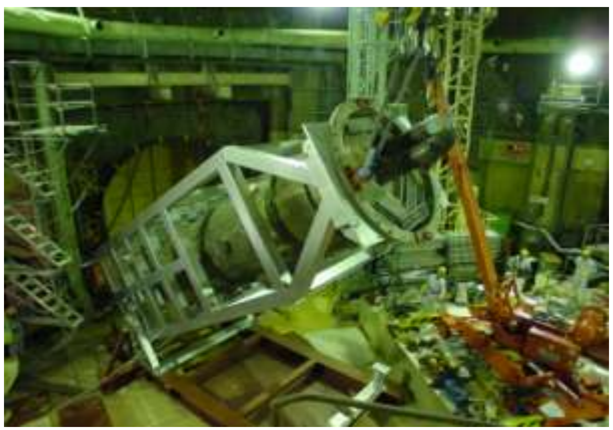
Seal the Section

Dismantle of old IHX

Remove Radiation Shielding Plate

Lift-up

Remove



Transportation Cask laid down

Move from RCV to Maintenance Build.

Lift-down inside Pit of Maintenance Building

Reduction of radiation exposure

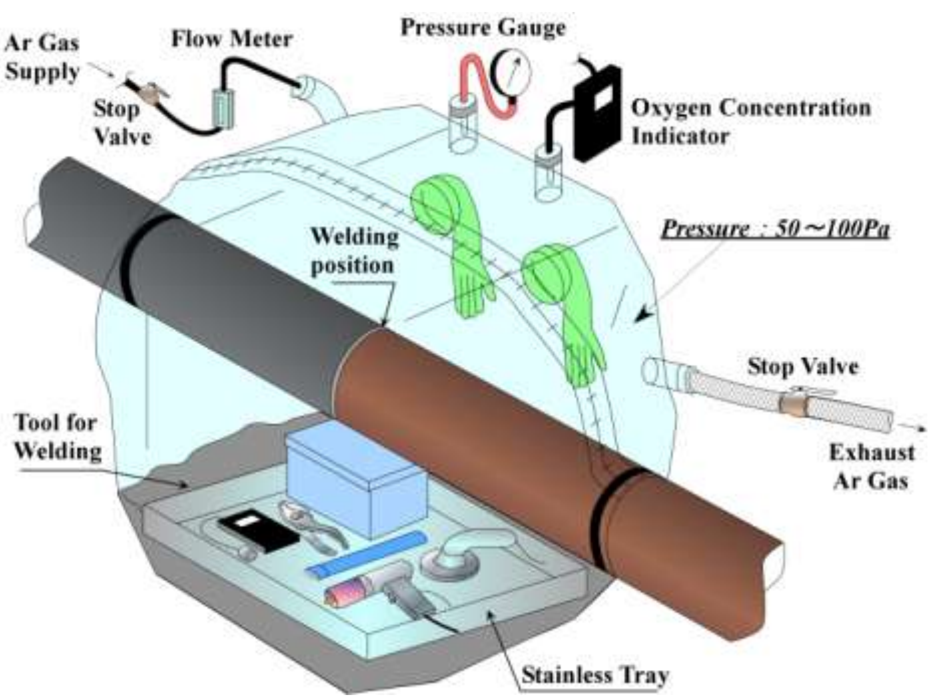
- Mock-up test (piping model and full simulation full scale model)
- Temporary installation of radiation shielding

Temporary radiation shielding

Mock-up training

Total accumulative radiation dose
2235man.mSv

Purity control of sodium



Utilization of seal bag and glove box

Control value of impurity concentration

Argon gas inside seal bag

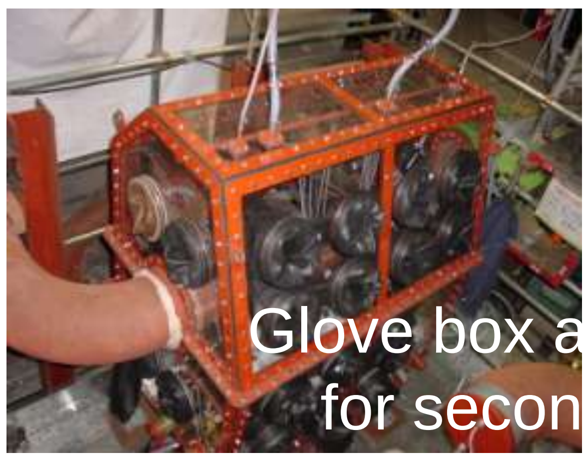
O_2 < 1,000 ppm

Cover gas in cooling system

O_2 < 300 ppm

N_2 < 1,200 ppm

Seal bag for primary pipe



Glove box and seal bag for secondary pipe

Progress History of Monju

Dec. 2016 Japan's gov. released New Policy for FR Development, and decided Monju Decommissioning

Revision of plant maintenance plan

Apr. 2014 Basic Energy Plan (Japan's gov.)

Sep. 2013 Research Plan for MONJU (MEXT)

- May 2013 Ordered to suspend the preparatory activities for SST by NRA
- Nov. 2012 Identified inadequate maintenance management by NRA

Mar. 2011 Fukushima Daiich NPS Accident

· Aug. 2012 Recovery of IVTM trouble
- Aug. 2010 IVTM trouble

July 2010 Completion of SST-1

May 2010 Restart of SST-1

2005-2007 Plant modification to improve sodium safety

Dec.1995 Sodium leak accident

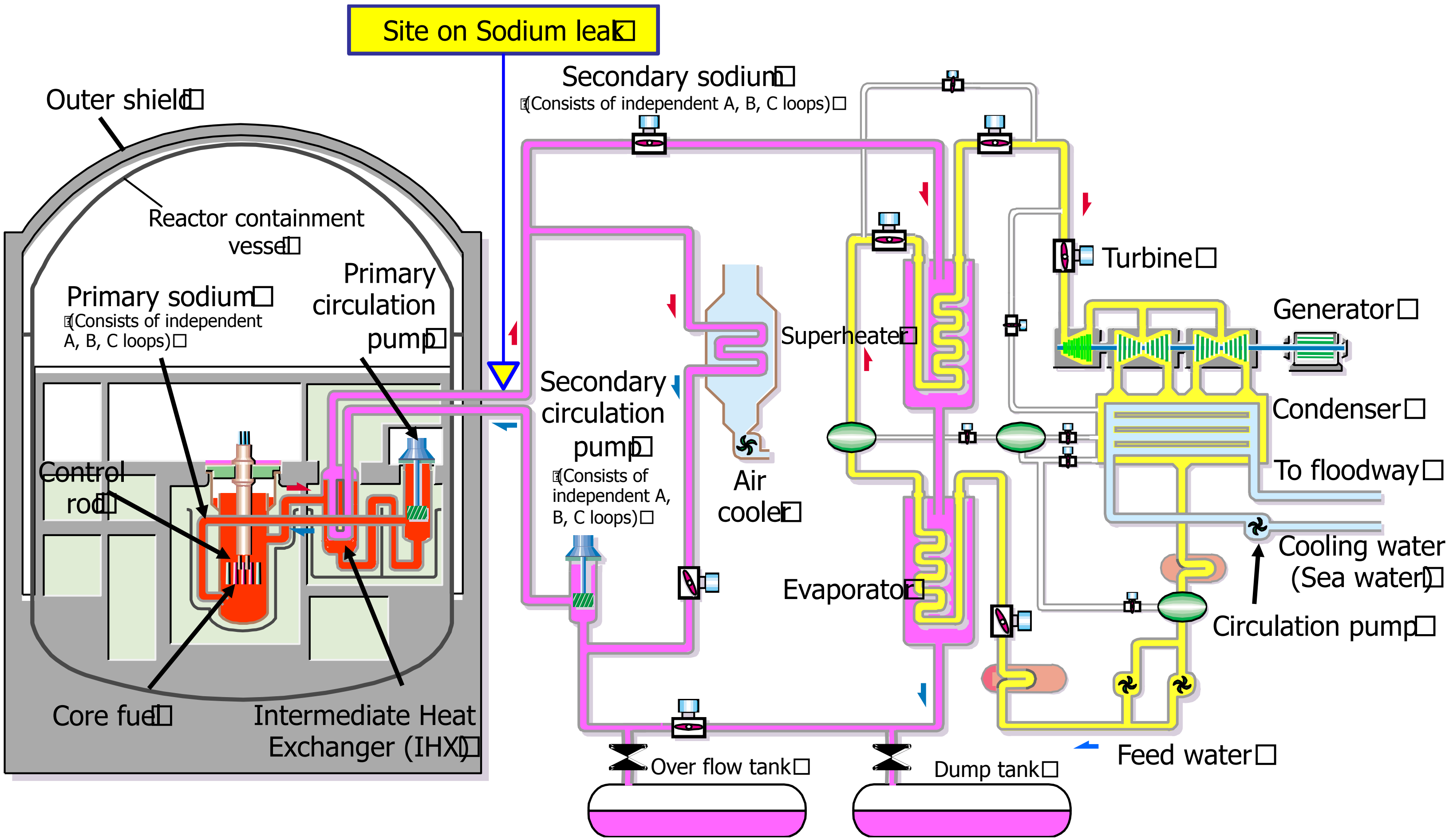
Aug.1995 First grid

Apr. 1994 Criticality

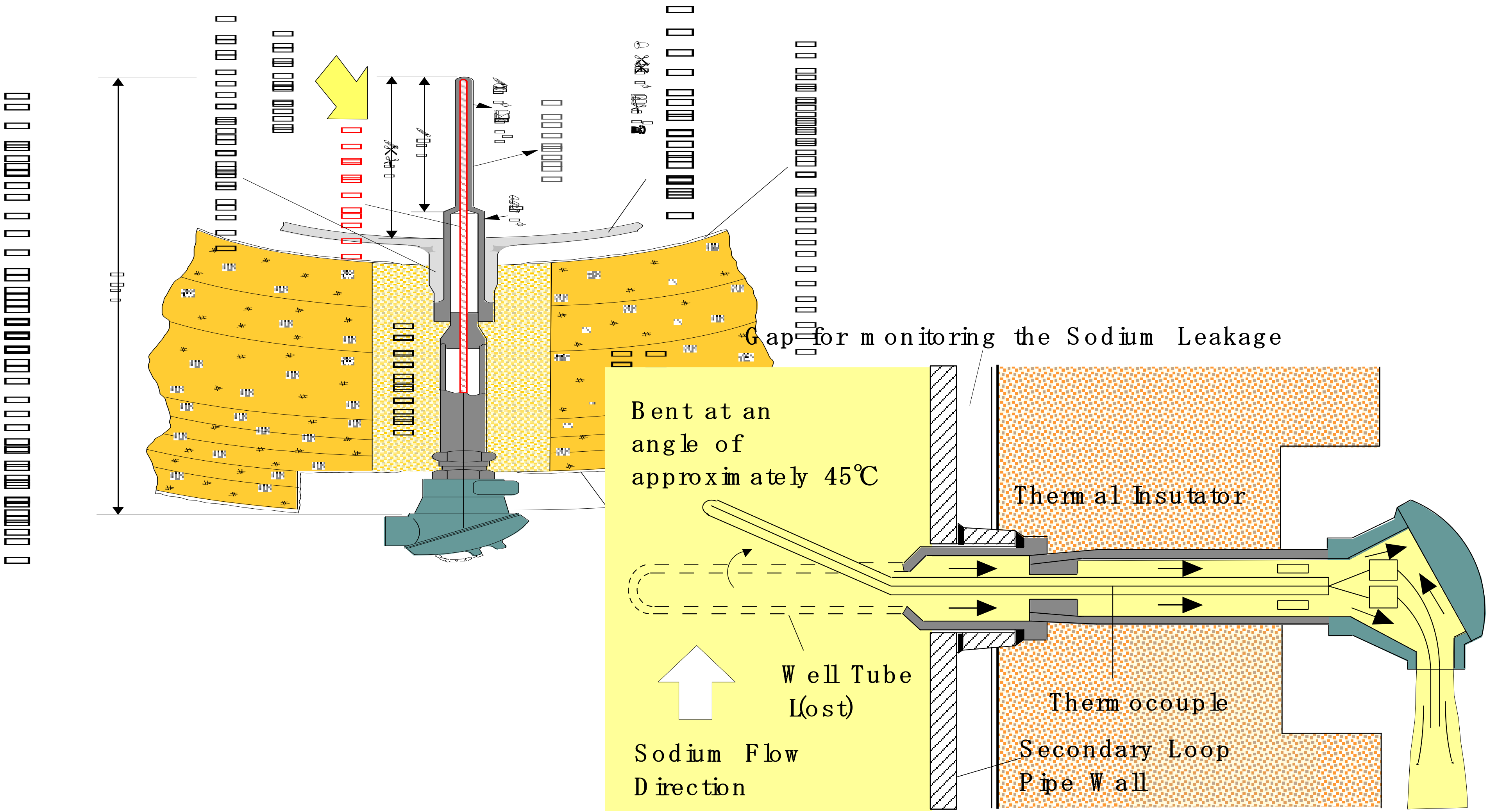
Sodium Leakage in secondary loop

Dec. 8th, 1995

- Sodium Leakage through failure of a thermocouple well at Secondary loop just after IHX

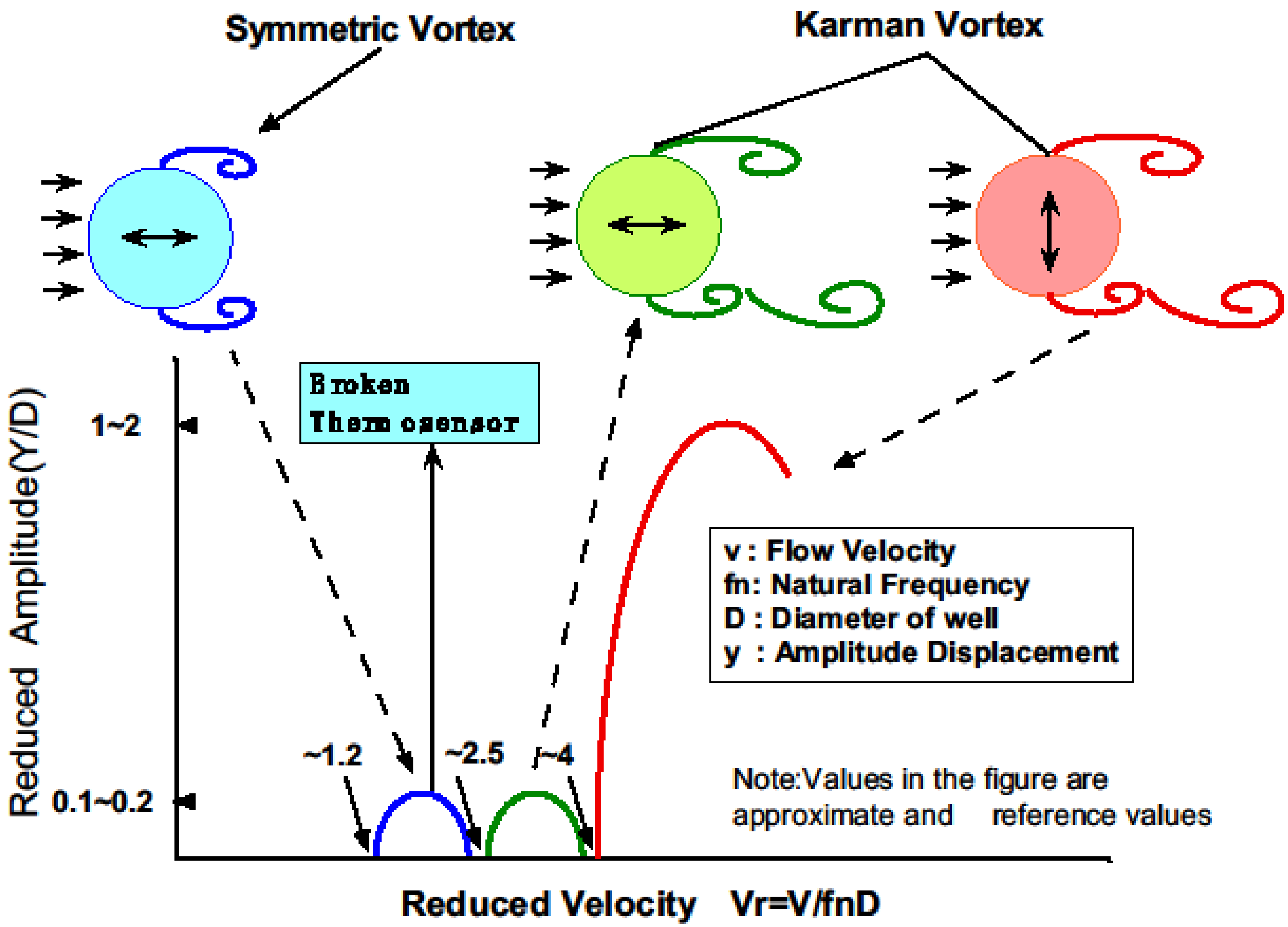


High cycle fatigue and failure due to Flow Induced Vibration in Thermocouple well



The Sodium Leak Flow Path

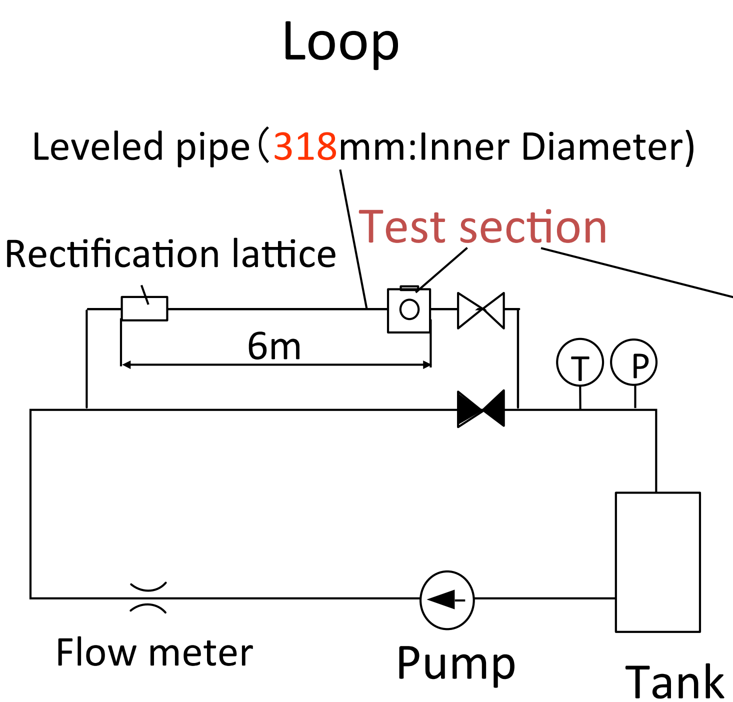
Flow Induced Vibration

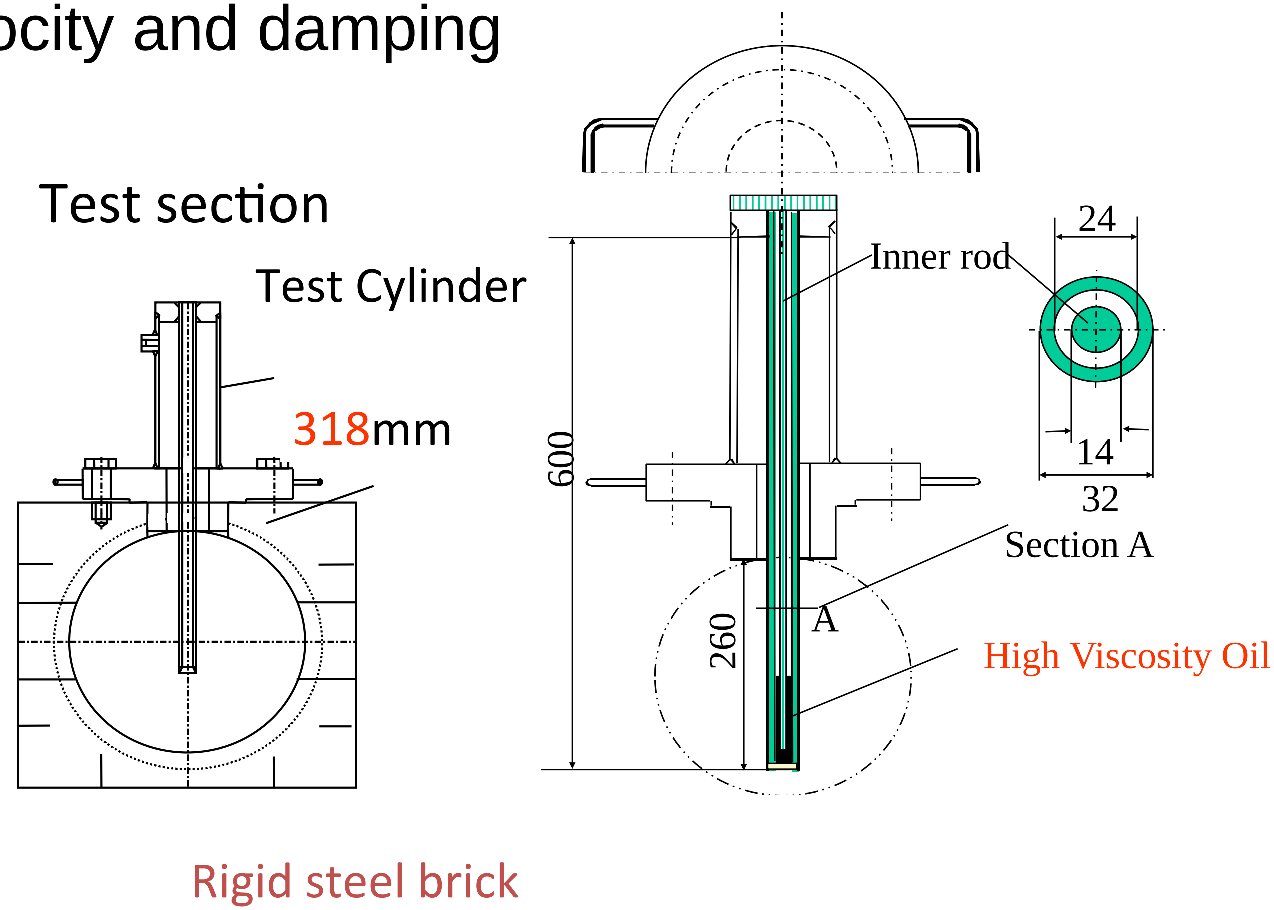


Explanation of Vortex Shedding

Experiments on Symmetric Vortex and Vibration

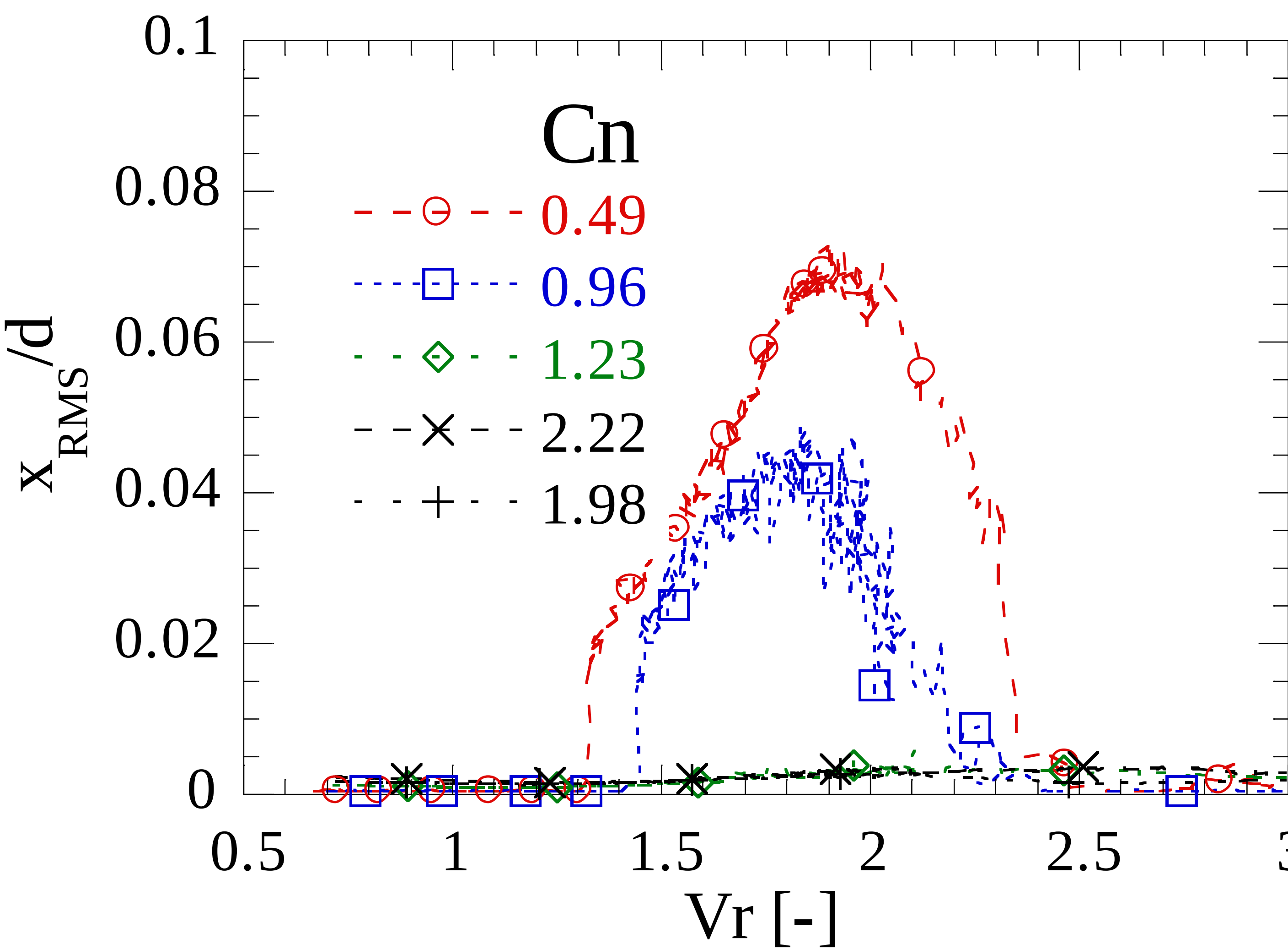
- Several Experiments were done with parameters of flow velocity and damping of vibration



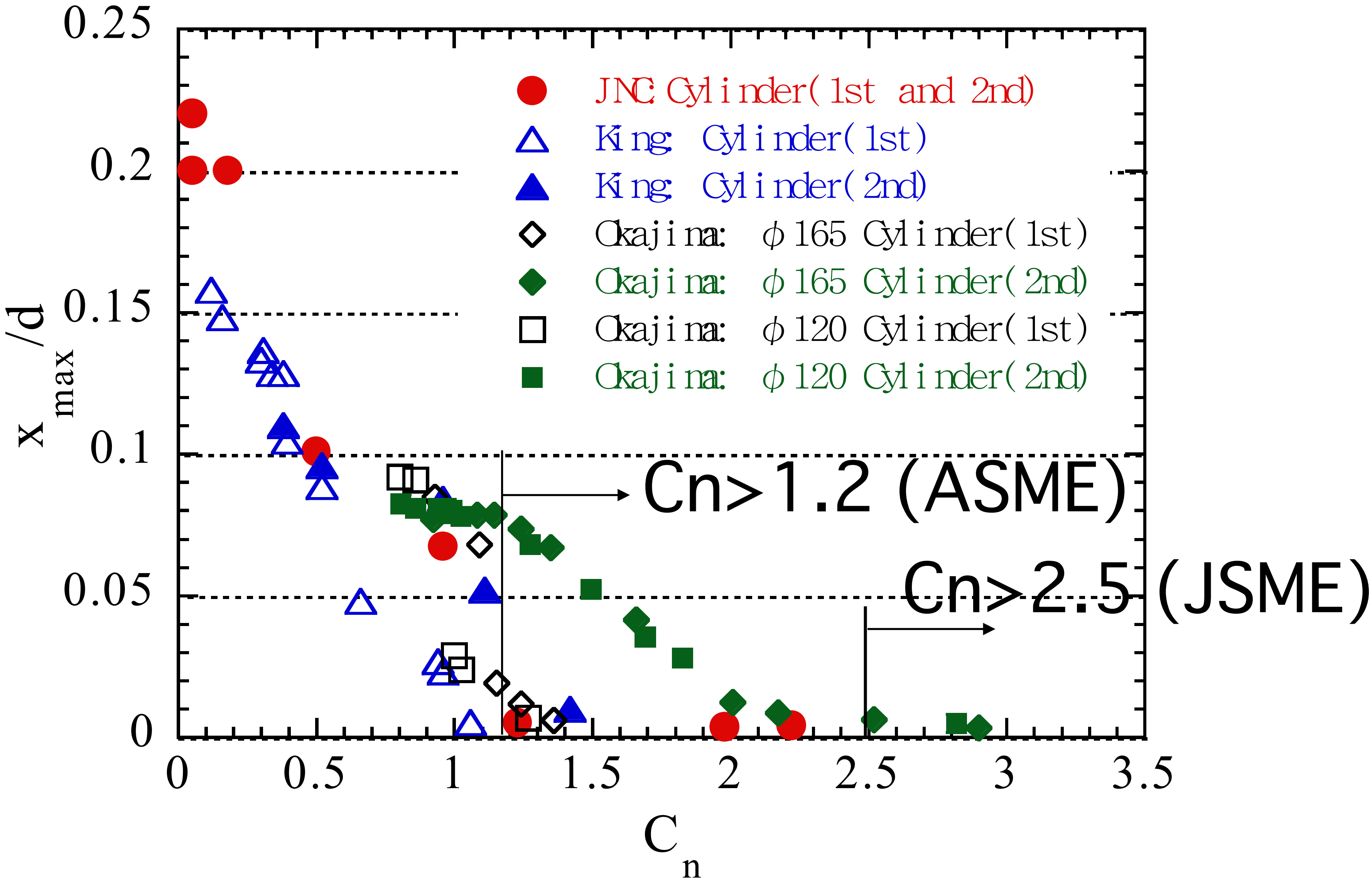


Re: ~ 3 x 10^6

Displacement in the flow direction



Summary of Data and Guide Lines



Guide Lines to prevent Flow Induced Vibration

ASME guideline(’95)	**JSME guideline(’98)**
Vr<3.3 and Cn>1.2	Vr<3.3 and Cn>2.5

Vr<3.3 : **Avoidance Criterion for VIV in cross-flow direction**

Cn>1.2 : **Suppression Criterion for VIV in flow direction**

Reactor Physics: Breeding Ratio Evaluation

Reaction Rate Distribution (Power Distribution)

- Reaction Rate Distribution
  & Reaction Rate Ratio

⇩

- Evaluation of Breeding Ratio
  & Power Distribution

Foils

A total of 2015 foils

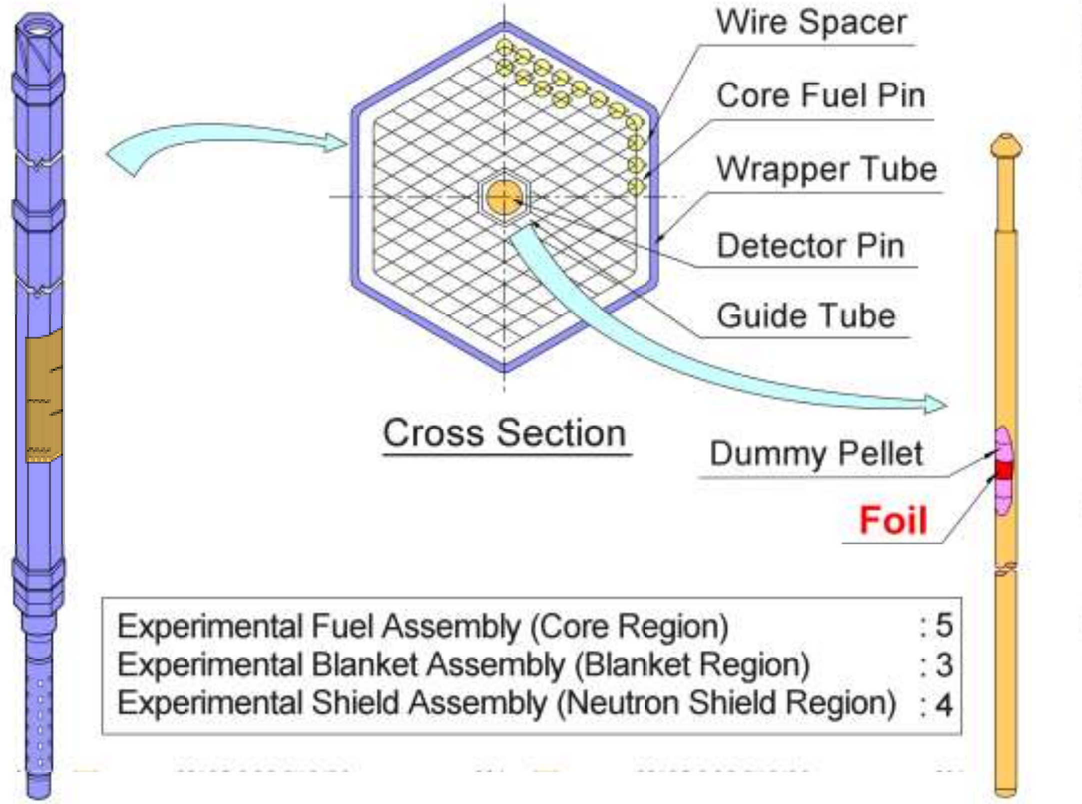
(1) Fission Foils

Pu-239, U-235,
U-238, Np-237

(2) Activation Foils

① (n,p) Reaction :
Higher Energy Neutron
Ni, Ti, Fe(Fe-54)

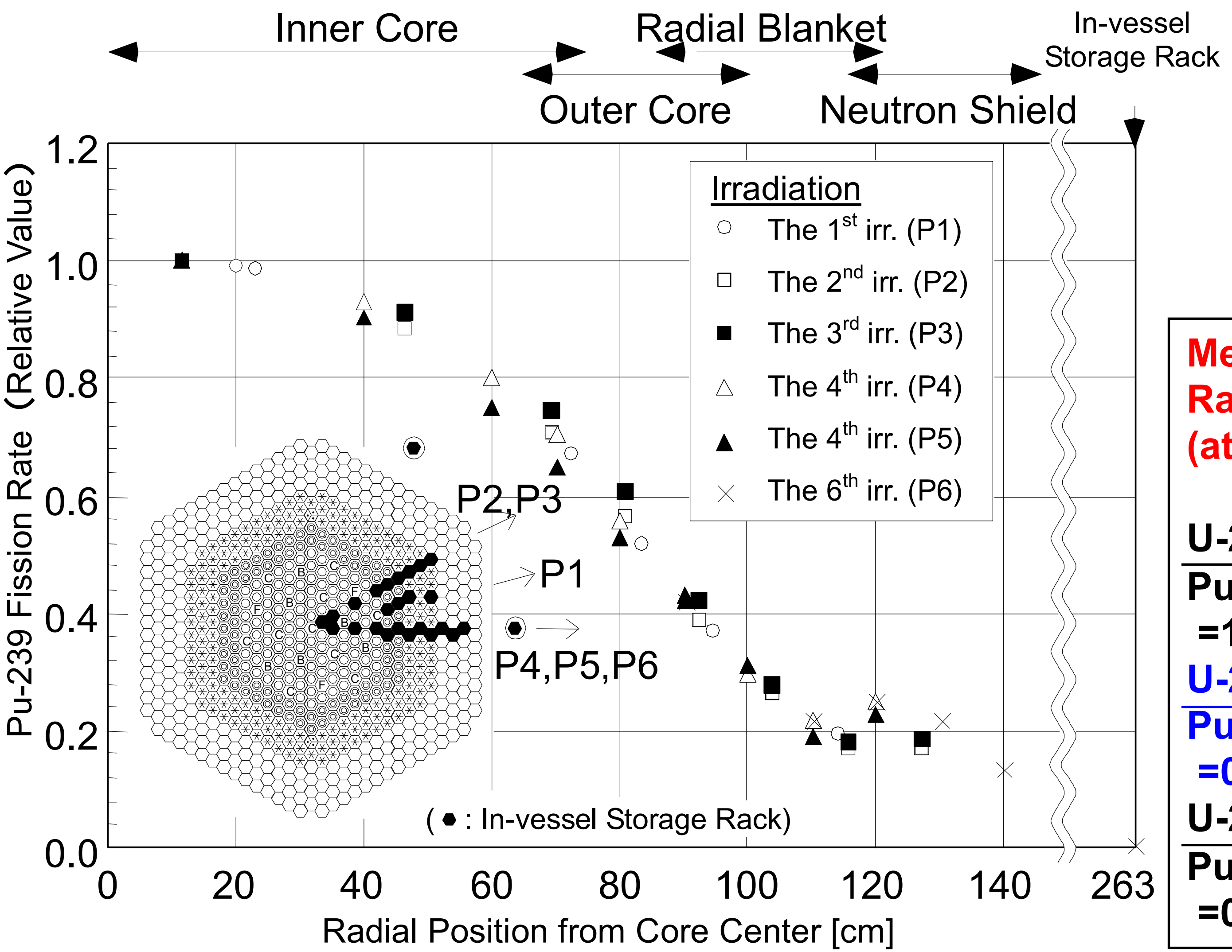
② (n, γ) Reaction :
Lower Energy Reaction
Au, Fe(Fe-58), Na, Sc, Co



Experimental Fuel Assembly
(Core Region)

Detector Pin

Measurement of Pu-239 fission rate distribution



Uncertainty in the Relative Distribution of Measured Pu-239 Fission Rate : 0.7-2.2%

Measured Reaction Rate Ratio (at the Core Center) :

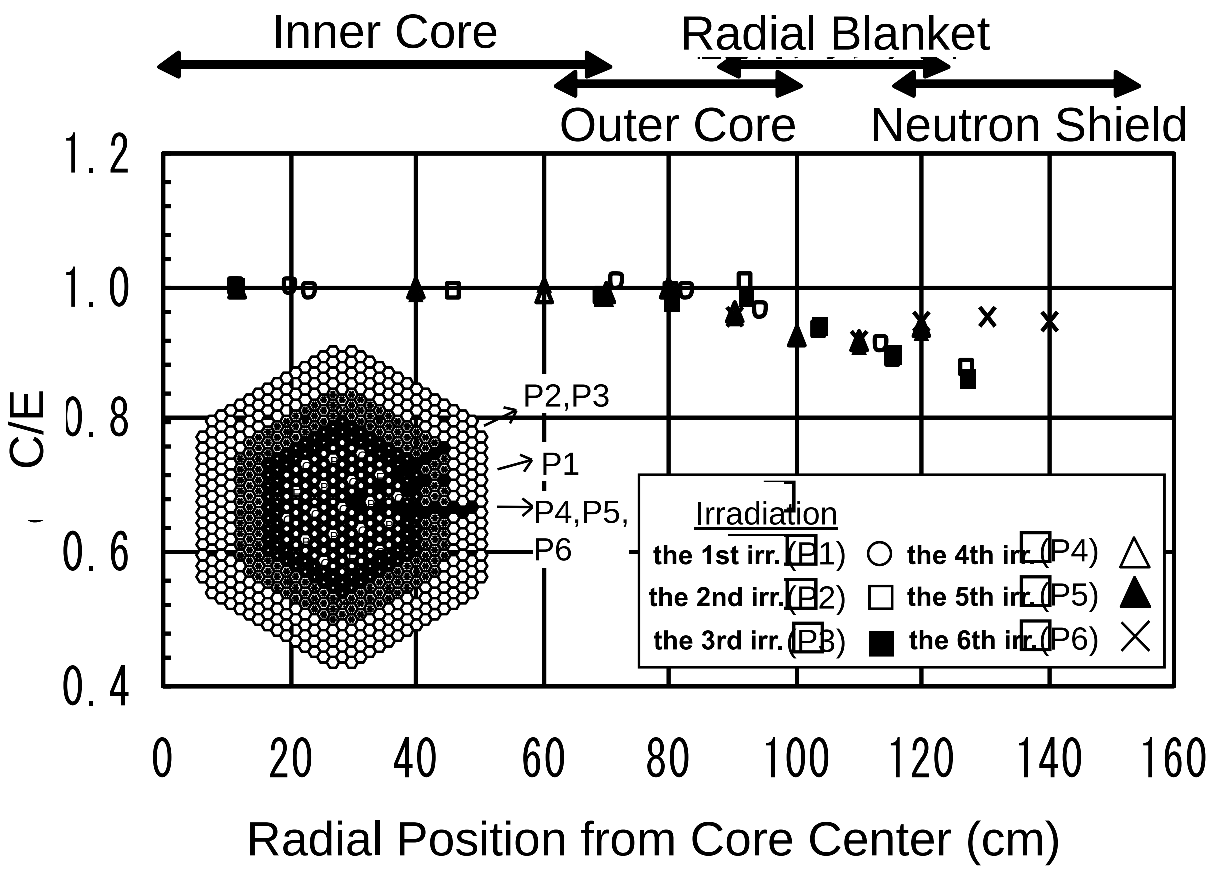
$$\frac{\text{U-235(n,f)}}{\text{Pu-239(n,f)}} = 1.07\text{-}1.08 \pm 3.7\%$$

$$\frac{\text{U-238(n,}\gamma\text{)}}{\text{Pu-239(n,f)}} = 0.164\text{-}0.167 \pm 6.2\%$$

$$\frac{\text{U-238(n,f)}}{\text{Pu-239(n,f)}} = 0.0242\text{-}0.0246 \pm 3.9\%$$

Radial Distribution of Measured Pu-239 Fission Rate on the Core Mid-plane

Comparison of Pu-239 fission rate between Computation and Experiment



C/E Radial Distribution of Normalized Pu-239 Fission Rate on the Core Mid-plane (Tri-Z, 70g Calculation by DIF3D Code)

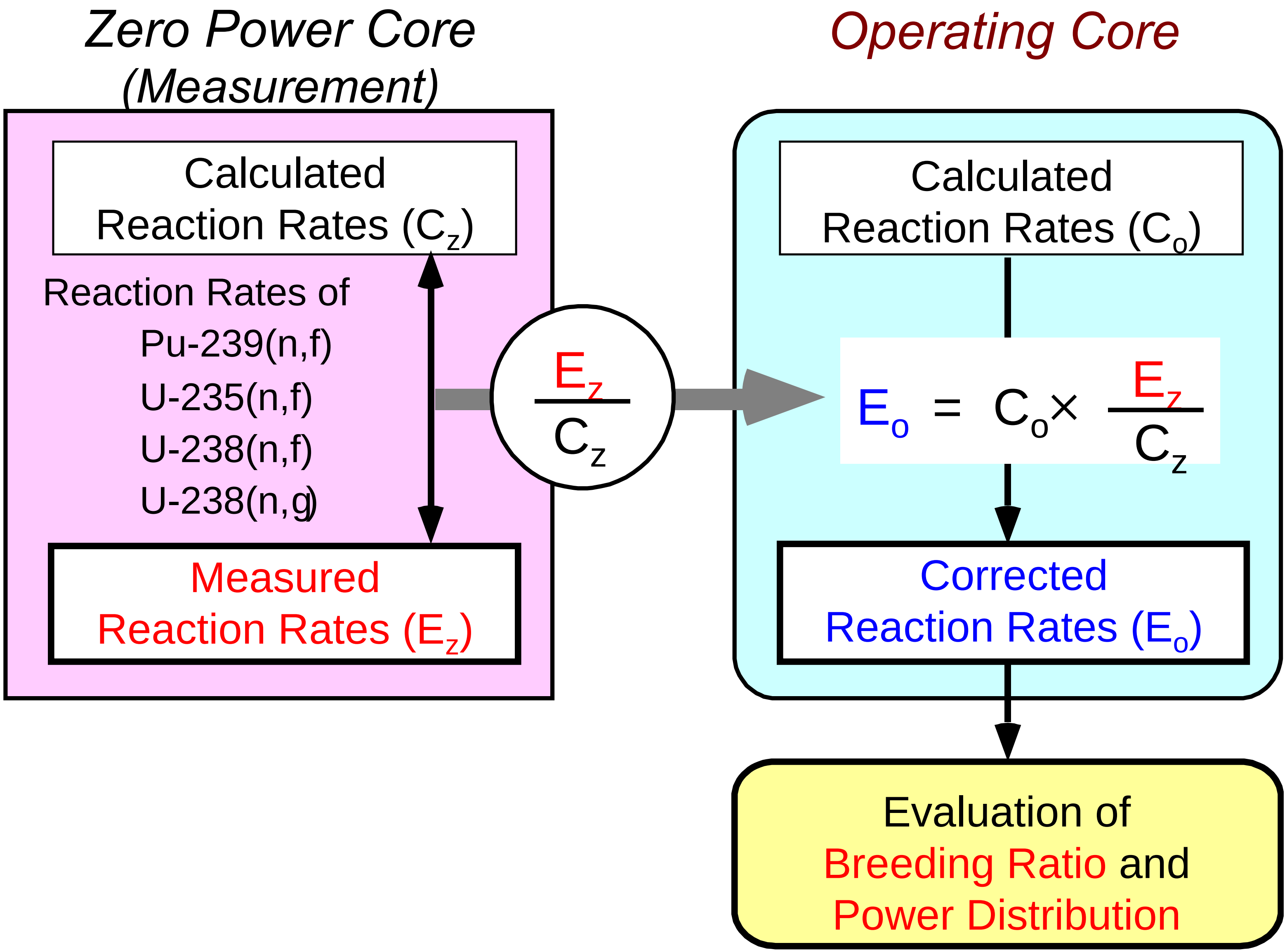
C/E of Reaction Rate Ratio (at the Core Center):

$\frac{\text{U-235(n,f)}}{\text{Pu-239(n,f)}}$ =0.98~0.99

$\frac{\text{U-238(n,}\gamma\text{)}}{\text{Pu-239(n,f)}}$ =1.00~1.02

$\frac{\text{U-238(n,f)}}{\text{Pu-239(n,f)}}$ =0.97~0.98

Evaluation Procedure of Breeding Ratio and Power Distribution



Evaluation of Breeding Ratio

Evaluation Results of Breeding Ratio

	Evaluated Breeding Ratio from the Measurement *			
	Inner Core	Outer Core	Axial Blanket	Radial Blanket
JENDL-3.2 Library	0.399	0.208	0.217	0.361
	0.607		0.578	
	1.185			
JENDL-2 Library	0.397	0.207	0.218	0.361
	0.604		0.579	
	1.183			

* : at the Beginning of Cycle of Full Power Operation in the Initial Core

Reactor Physics on a Core with Am-241

- Pu-241 (half-life 14 years) decayed during 14.4 year suspension
- Am-241 accumulated.
- The excess reactivity decrease approximately 4%⊿k/k.
- The refueling in June to July, 2009.
- The following three-types of core fuel sub-assemblies in the core.
  - ✓ Type I: used in the previous SST more than 14 years ago. (114)
  - ✓ Type II: stored outside the core for more than 14 years. (78)
  - ✓ Type III: newly fabricated. (6)
- Pu contents : Type I < Type II < Type III
- Approximately 1.5 wt% of Am-241 in the Monju restart core.

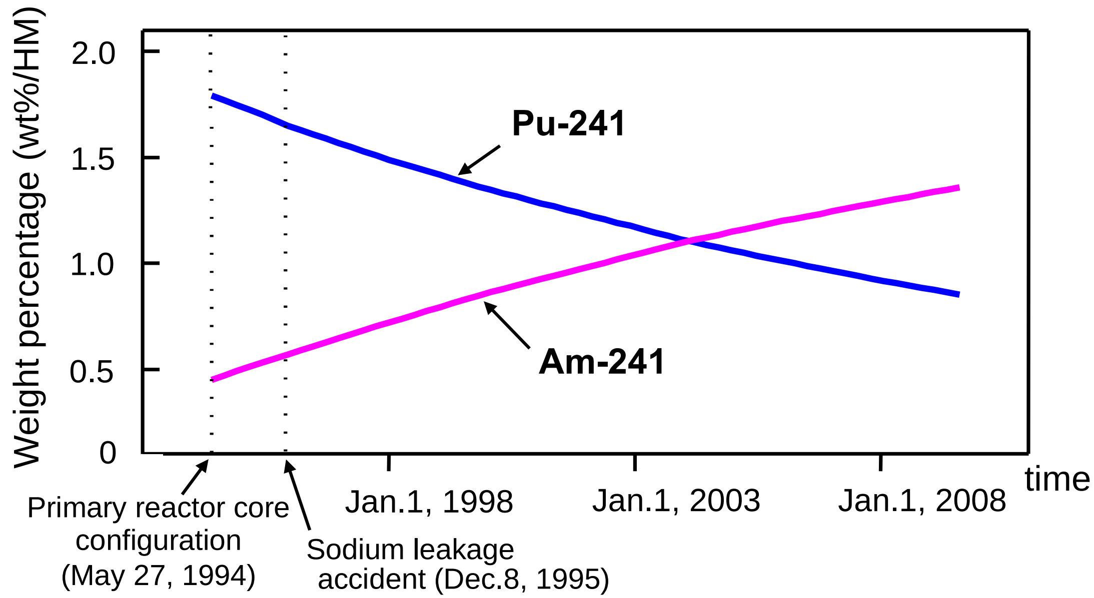
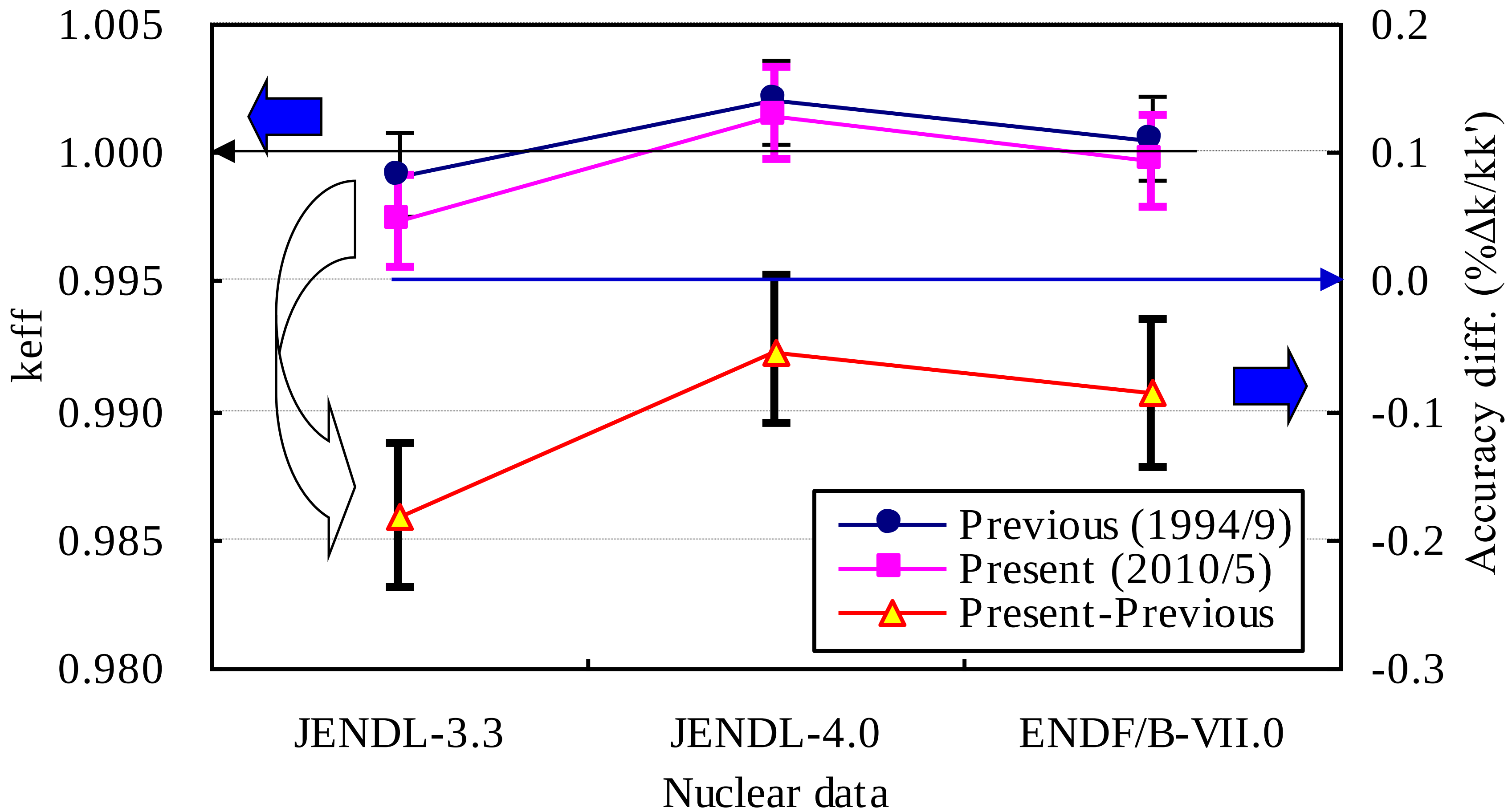


Fig.1 Weight % of Pu-241 and Am-241

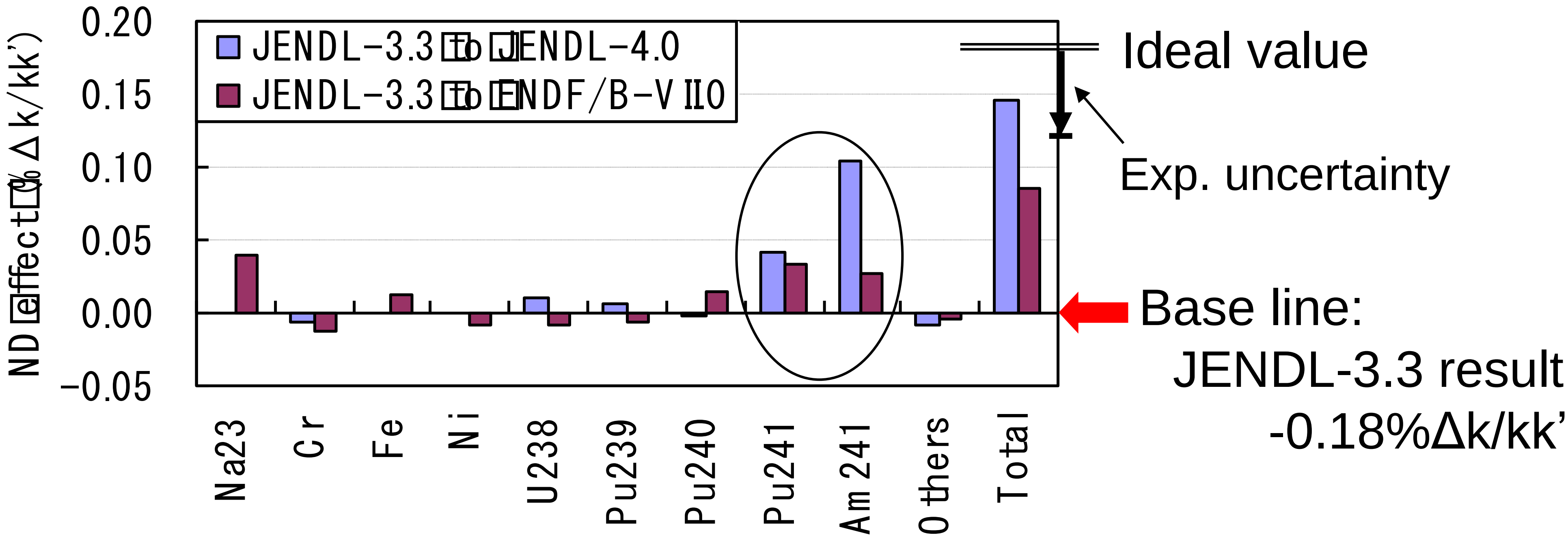
Accuracy difference between the cores

Accuracy difference (unit in reactivity). Ideally the difference should be zero.



- Dominant uncertainty (fuel composition) almost cancelled.
- Clear underestimation in JENDL-3.3 result.

Nuclear Data (ND) Effect on the accuracy of criticality with Am-241



Present - Previous

- Contributions of Am-241 and Pu-241 data clearly appear.
- Change in Am-241 capture and Pu-241 fission (JENDL-3.3→JENDL-4.0) is reasonable.

Ideas on Knowledge Management

- Archive the plant designs and data
  - The final designs and reasons of real plants
  - Reactor experiments
  - Experiences of Maintenance and also Maintenance Program
- Build a Numerical Plant based on simulation models and the archived database
  - Data of wide range of experiments
  - New experiments for simulation models
- Design and R&D work for next reactor with international collaboration

Conclusions

- Data and findings from Joyo
  - Change of reactor components and core
  - Reactor Experiments on
    - Irradiation of fuel, core material,
    - Power to Melt,
    - Natural Circulation for Decay heat removals
- Data and findings from Monju
  - Designs and fabrication techniques
  - Maintenance Program and experiences
  - Reactor experiments: Core characteristics
- Knowledge Managements
  - Archives of existing data and findings
  - Numerical Plant for collections of knowledge
  - Designs and R&Ds with International collaborations

Major Specifications of Cores in Joyo

Items		MK-I	MK-II	MK-III
Reactor Thermal Power	(MWt)	50/75	100	140
Max. Number of Driver Fuel S/A		82	67	85
Max. Number of Test Fuel S/A		0	9	21
Core Diameter	(cm)	80	73	80
Core Height	(cm)	60	55	50
^{235}U Enrichment	(wt%)	~23	~18	~18
Pu Content Total	(wt%)	~18	<30	<30
Fissile	(wt%)	–	~20	~16/21*
Max. Linear Heat Rate	(W/cm)	320	400	420
Max. Neutron Flux Total	(n/cm^2s)	3.2×10^{15}	4.5×10^{15}	5.7×10^{15}
Fast (>0.1MeV)	(n/cm^2s)	2.2×10^{15}	3.2×10^{15}	4.0×10^{15}
Max. Burn-up (Pin Average)	(GWd/t)	42	75	90
Primary Coolant System Flow Rate	(t/h)	2,200	2,200	2,700
Temp. (Inlet)	(℃)	370	370	350
Temp. (Outlet)	(℃)	435/470	500	500
Blanket/Reflector/Shielding		Blanket/SUS	SUS / SUS	SUS / B_4C

*) Inner/Outer Core