IGORR 22nd & IAEA TM
June 18 (Wed), 2025
@ MITO City Civic Center 16:00 – 16:20

東京都市大学
TOKYO CITY UNIVERSITY

Conceptual Design of High-Flux Fast Spectrum Irradiation Reactor with Metallic Slim Fuel Pin

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Introduction

1. Background
2. Objectives

1.1 Background

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- In the development of fast reactors, **fast neutron irradiation fields** are essential for clarifying the degradation behavior of reactor structural materials.

- Fast neutron irradiation field in western countries

➡ **Only "Joyo"**

Experimental Fast Reactor "Joyo"[1]

Japan should continue to lead by
- Restarting Joyo
- Installing new irradiation fields

This study proposes a new core concept ----

By boosting fast neutron flux, I aim to:
- ➢ **Shorten** irradiation times
- ➢ **Accelerate** R&D cycle of fast reactors
- ➢ **Increase** production of medical isotopes

[1] https://mainichi.jp/articles/20230524/k00/00m/020/248000c

1.2 Objectives

The objectives of my study are to:

1. Evaluate the effectiveness of following design approaches to boost fast neutron flux
   - ➢ Applying metal fuel pin with reduced diameter
   - ➢ Enriching inner core with more Pu

2. Perform the core design and assess its irradiation performance

Approaches for Achieving High Neutron Flux

2. Approaches for Achieving High Neutron Flux

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$$Q = E_R V_{core} \sum_g \Sigma_{fg} \phi_g$$

Q	Reactor thermal power (W)
E_R	Energy per fission (J/fission)
V_{core}	Core Volume (cm^3)
Σ_{fg}	Macroscopic fission cross-section of energy group g (cm^{-1})
ϕ_g	Neutron flux of energy group g ($n/cm^2/s$)

- Q → fixed at **100 MWt**
  - To ensure sufficient flux in relatively small EPZ* in Japan
- **Core volume reduction** affects on:
  - Criticality — applied **high-density metal fuel** to ensure enough fissile material
  - Power density — applied **narrow fuel pins** to avoid Fuel-cladding chemical interactions
- To **boost** neutron flux in irradiation area
  - Divided core into two regions
  - Enriched inner core with more Pu

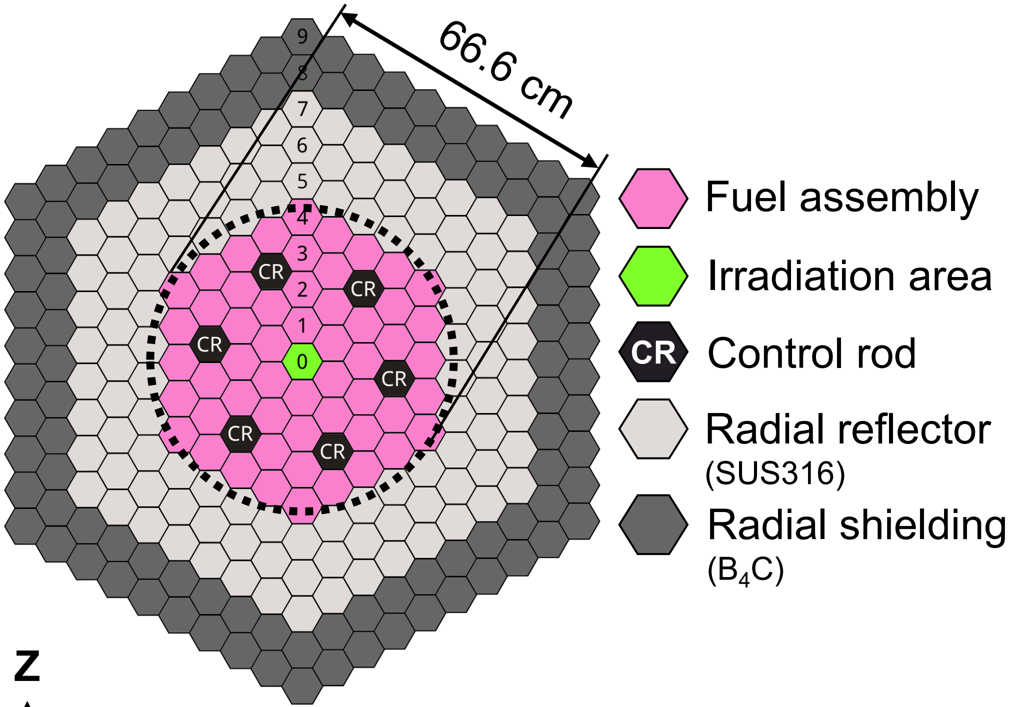
→ **Fission reactions ↑**
Local neutron flux ↑
Around irradiation area

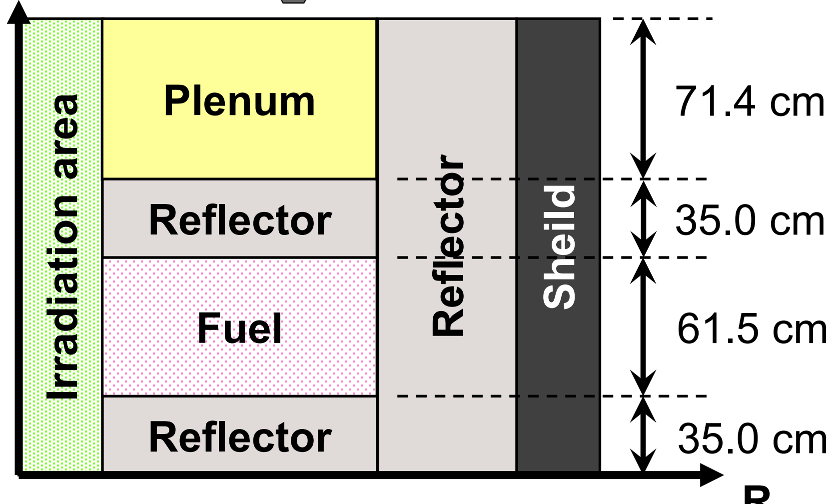
* EPZ : Emergency Planning Zone

Analysis Methods

1. Core Specifications
2. Design Conditions
3. Calculation Code
4. Evaluation Items

3.1 Core Specifications





Core cross-section

Specifications of HFSR

Items	Values
Thermal Power (MWt)	100
Operating cycle length	Parameter
Number of Pu enrichment regions (-)	1 or 2
Number of FAs (-)	54
Number of CRs (-)	6
Fuel [1)]	U-Pu-6Zr
Smear density (%TD)	75
Pu enrichment (wt%)	≤ 30
U enrichment (wt%)	0.2
Coolant	Sodium
H/D (-)	0.924

1) Pu Isotopic ratio (wt%)
Pu-238/239/240/241/242/Am-241 = 1/66/24/4/3/2

Reference:

[2] Japan Atomic Energy Agency. *Oarai Research and Development Center (South District) Experimental Fast Reactor Facility ("Joyo"): Article 32 (Core, etc.), Part 1: Paragraphs 1–3.* Japan Atomic Energy Agency, 2022.

3.2 Design Conditions

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Design Conditions

Items	Values
Max. linear heat rate (W/cm)	≤ 300
Excess reactivity (Δk/k)	< 0.035
Sodium void reactivity ($)	< 8
Fast neutron irradiation fluence (n/cm^2)	5×10^{23}
Max. cladding temperature (℃)	≤ 650

3.3 Calculation Codes

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Codes

- Continuous-energy Monte Carlo code MVP
- MVP-BURN, which is a coupling code of MVP and a burn-up calculation module BURN

Calculation conditions

Items	Assembly	Full core
Codes	MVP3.0 MVP-BURN	←
Nuclear data library	JENDL-4.0	←
Number of histories	10,000	←
Number of batches	500	1,000
Skipped batches	20	50

Cells with "←" indicate the same value as in the left column.

3.4 Evaluation Items

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Fast neutron flux $\boldsymbol{\phi_{fast}}$

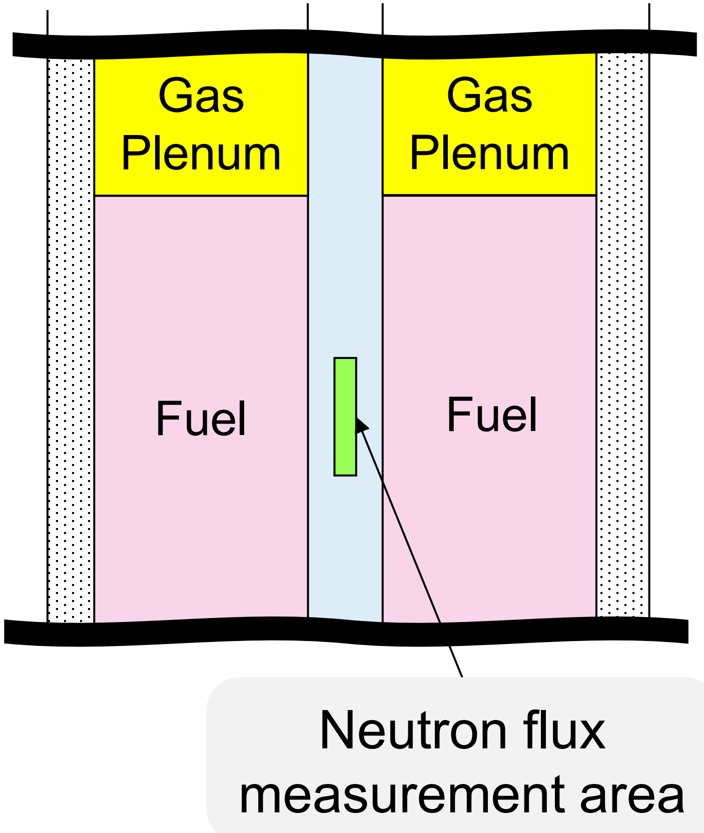
- Values for 1st cycle at BOC

Annual fast neutron fluence $\boldsymbol{F}$

$$F = \phi_{fast} \times CF \times 31{,}536{,}000 \left[\frac{s}{y}\right]$$

$$CF = \frac{Operation[day]}{Operation[day] + outage[day]}$$

F	Annual fast neutron fluence (n/cm^2)
ϕ_{fast}	Fast neutron flux (n/cm^2/s, E≧0.1MeV)
CF	Capacity factor (–)



Axial core cross-section

Neutronic Analyses

1. Effect of Number of Fuel Pins in Assembly
2. Effect of Higher Pu Enrichment in Inner Core

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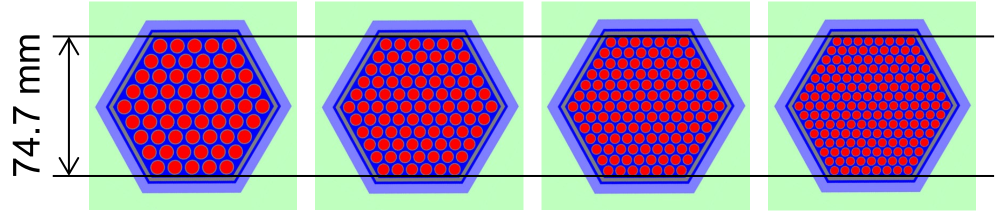
Analysis Conditions



Infinite single assembly model

Fuel assembly specifications for each case

	Items	Case 1	Case 2	Case 3	Case 4	limit
	Number of fuel pins (-)	61	91	127	169	
Fixed	Avg. linear heat rate (W/cm)	300.0	←	←	←	
Fixed	P/D	1.18	←	←	←	
Fixed	Wrapper tube inner flat-to-flat distance (mm)	74.7	←	←	←	
	Fuel pin diameter (mm)	7.86	6.47	5.50	4.78	≥ 4.42
	Fuel slug diameter (mm)	5.94	4.89	4.16	3.61	
	Cladding wall thickness (mm)	0.50	0.41	0.35	0.30	≥ 0.30
	Wire diameter (mm)	0.69	0.57	0.48	0.42	≥ 0.35

↑ Same specifications as Joyo MK-IV (Case 3)

4.1 Effect of Number of Fuel Pins in Assembly

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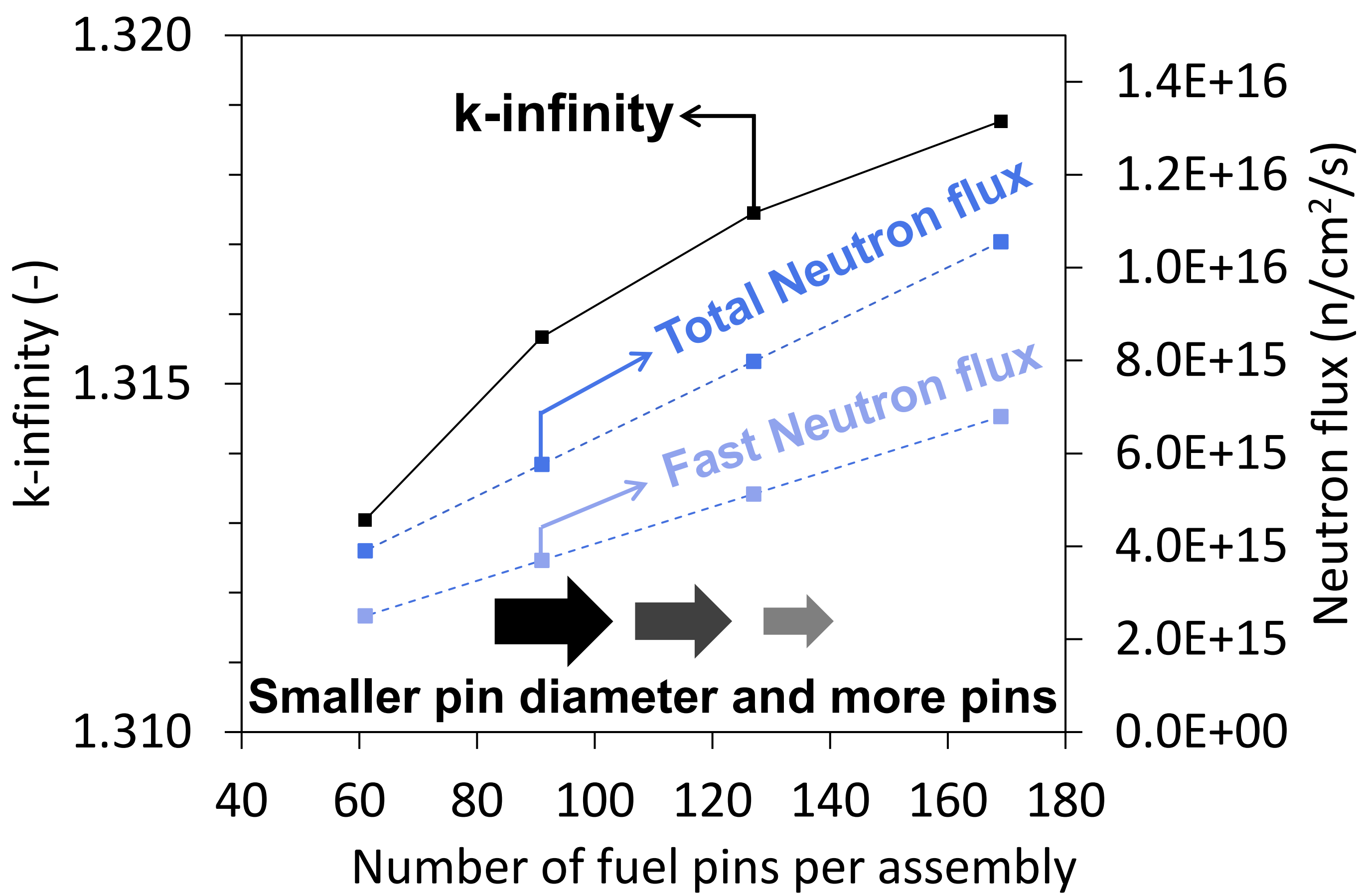
Results



- Increasing the number of fuel pins
  → An increase in fuel volume and power per fuel assembly

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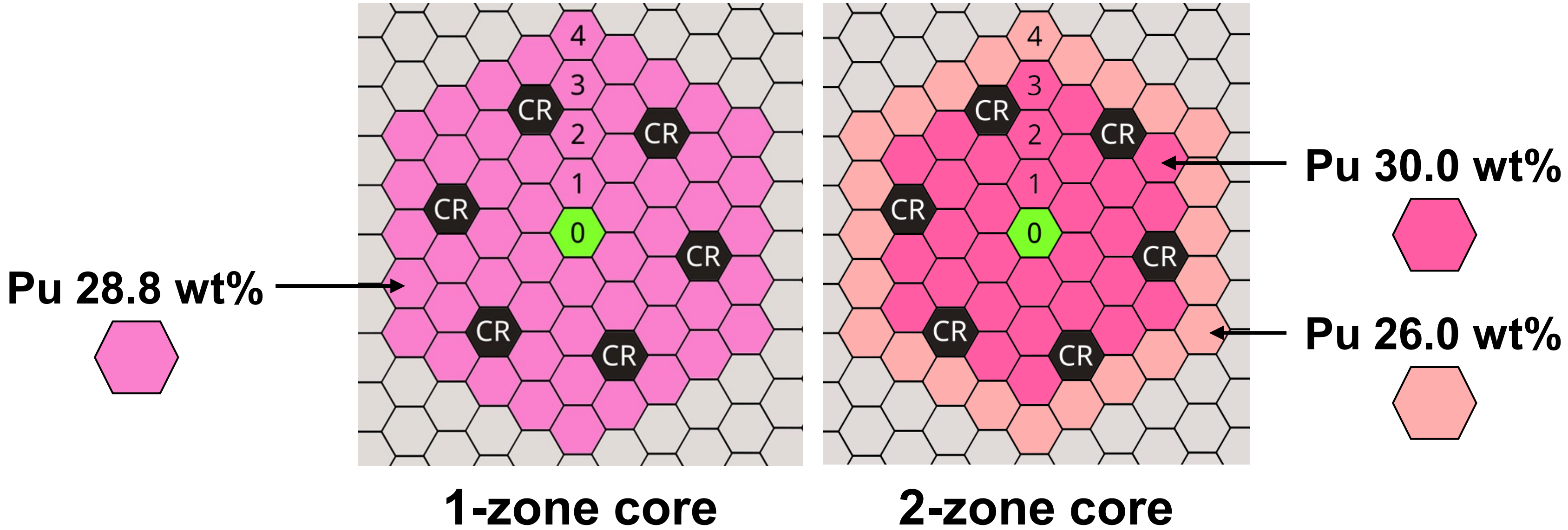
Analysis Conditions

Items	1-zone core	2-zone core	Design target
Thermal Power (MWt)	100	←	
Number of FAs (-)	54	←	
Number of fuel pins per FAs (-)	169	←	
Pu enrichment (I/O, wt%)	28.8	30.0 / 26.0	≦ 30.0



4.2 Effect of Higher Pu Enrichment in Inner Core

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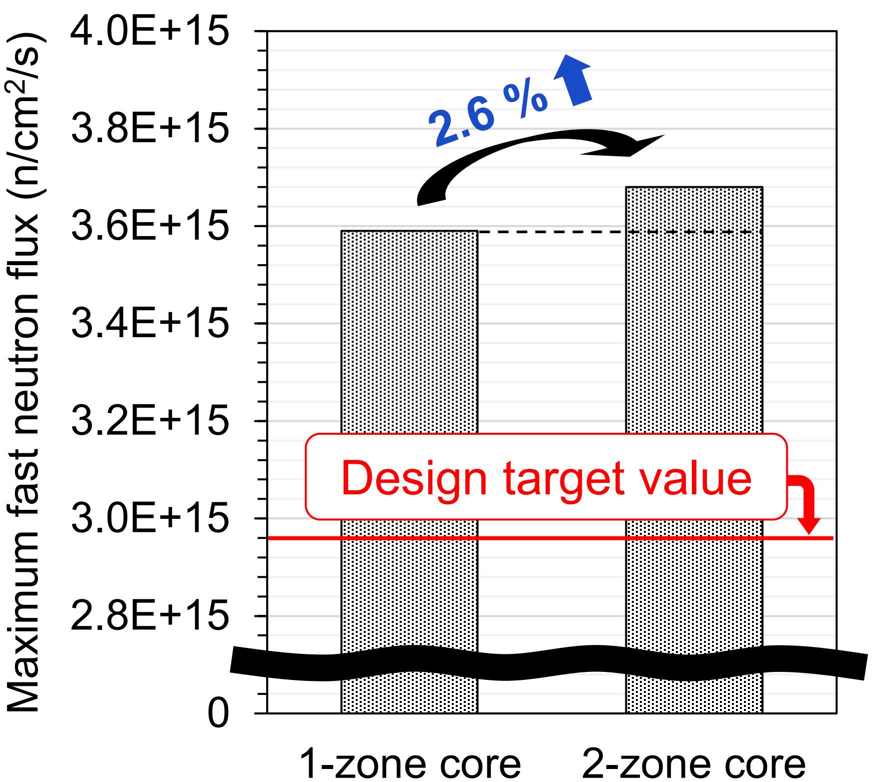
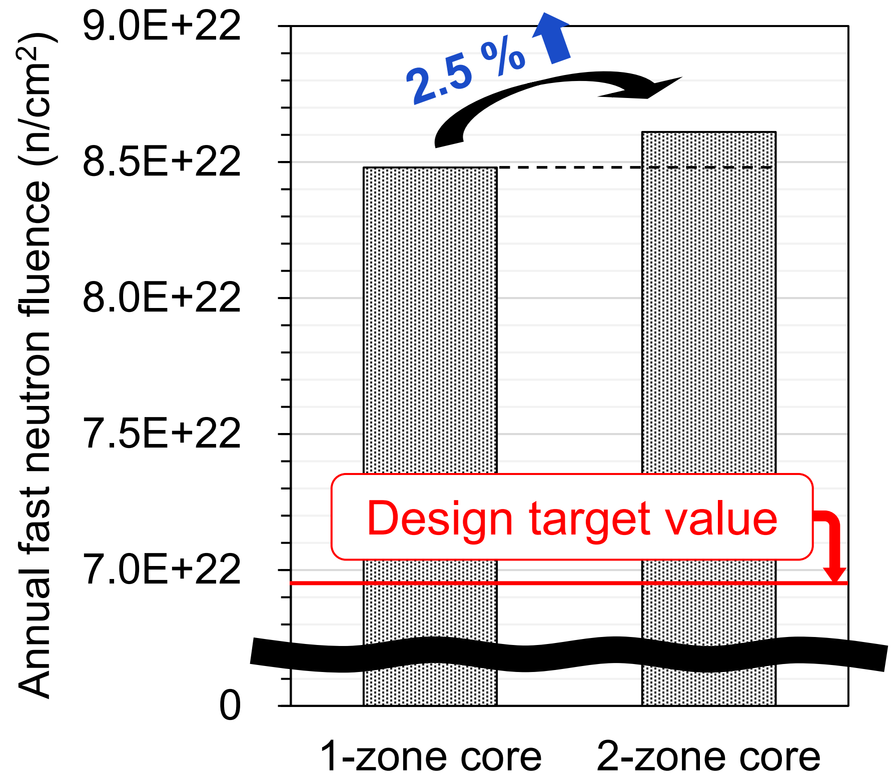


Maximum fast neutron flux
(1st cycle at BOC)



Annual fast neutron fluence
(1st cycle at BOC)

- Both cases meet design targets
- 2-zone core → higher irradiation performance

4.2 Effect of Higher Pu Enrichment in Inner Core

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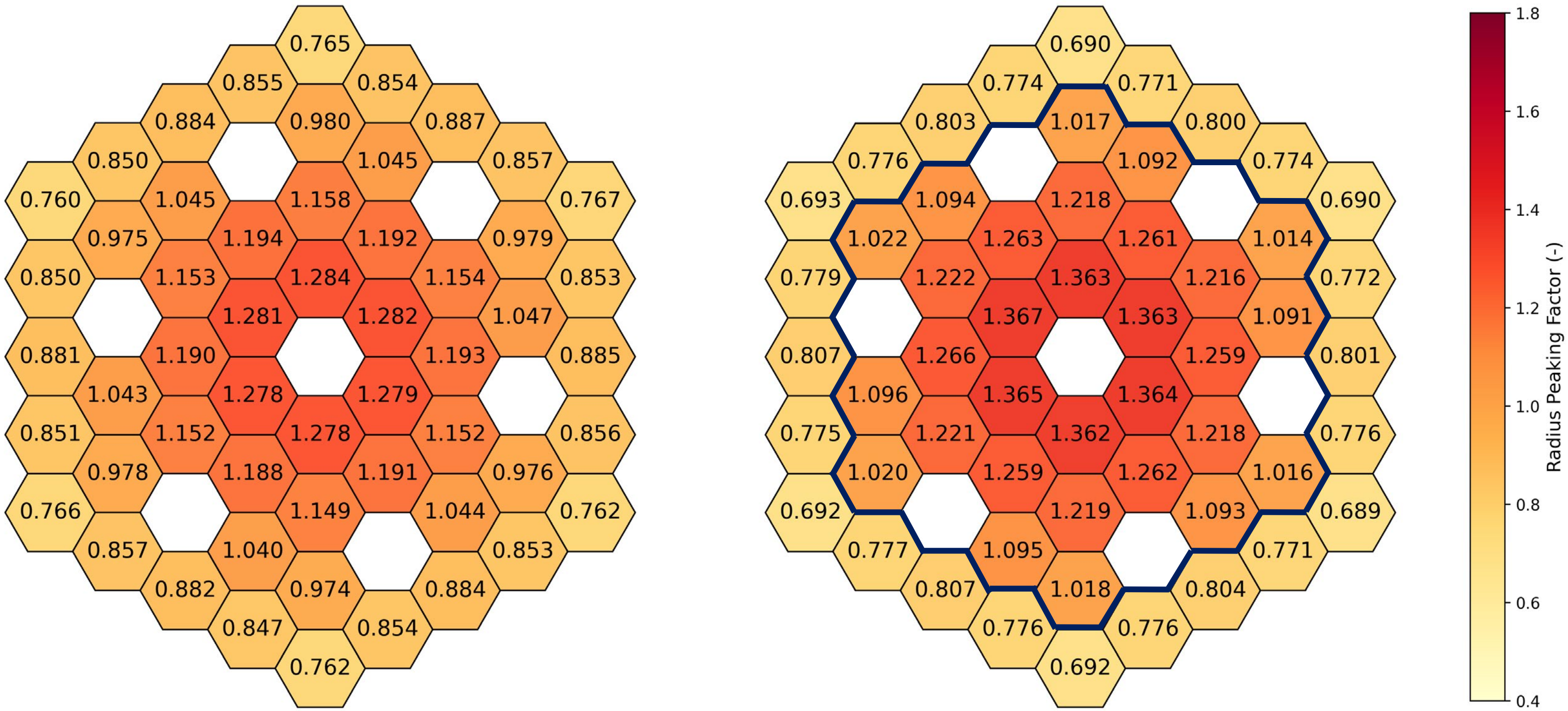
Core Radial Power Distribution (1st cycle at BOC)



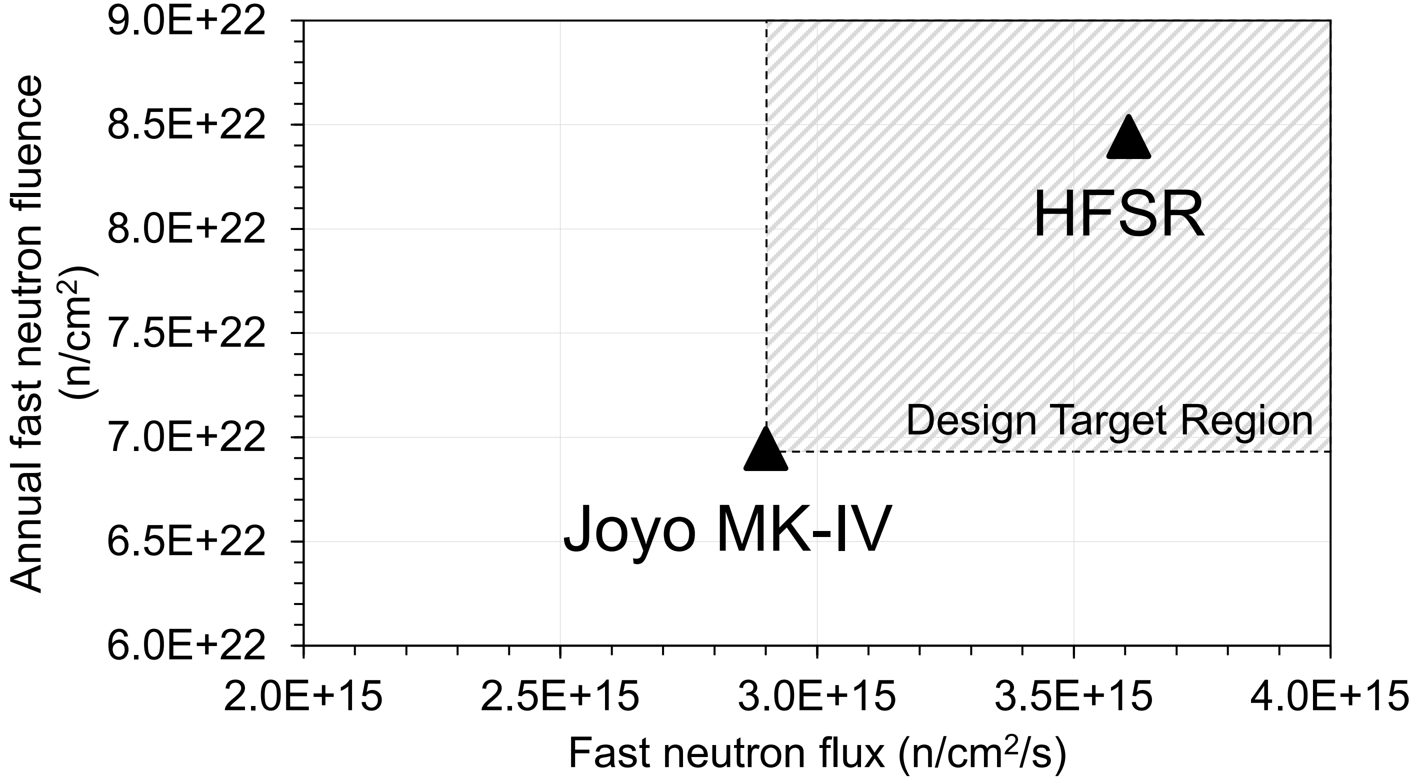
1-zone core **2-zone core**

Items	1-zone core	2-zone core	Design Targets
Radial peaking factor (-)	1.28	1.37	
Max. linear heat rate(W/cm)	279.2	297.4	≦ 300

Performance & Safety Assessment

1. Irradiation Performance
2. Safety Parameters
3. Thermal-hydraulic Summary Evaluation

5.1 Irradiation Performance



- Fast neutron flux: +24%
- Annual fast neutron fluence: +22%
  (both vs. Joyo MK-IV)

5.2 Safety Parameters

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Safety parameters for 1st cycle at BOC

	Joyo MK-IV		HFSR
Sodium void reactivity ($)	-(2.7～6.6)	❯	-2.36
Doppler coefficient (10^{-3} Tdk/dT)	-(1.3～3.3)	≈	-2.88

▌Sodium void reactivity

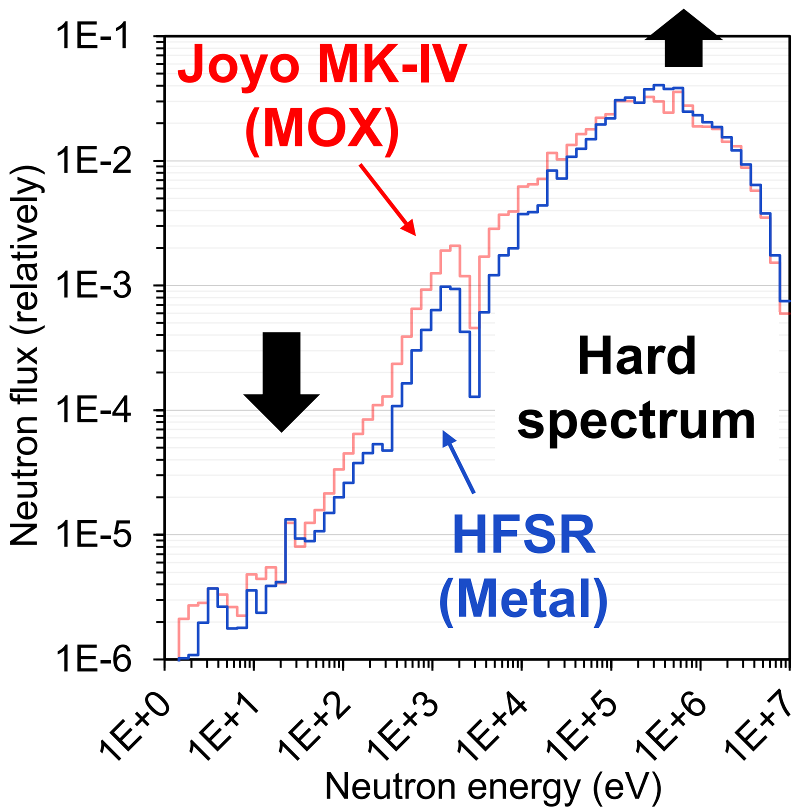
- Hard neutron spectrum
- Weak neutron leakage core shape (H/D=0.924)

➡ **HFSR > Joyo MK-IV**

▌Doppler Coefficient

- Hard neutron spectrum
  → Low absorption in U-238 resonance range
- Use of depleted U

➡ **HFSR ≈ Joyo MK-IV**



Neutron spectrum (Core center)

5.3 Thermal-hydraulic Summary Evaluation

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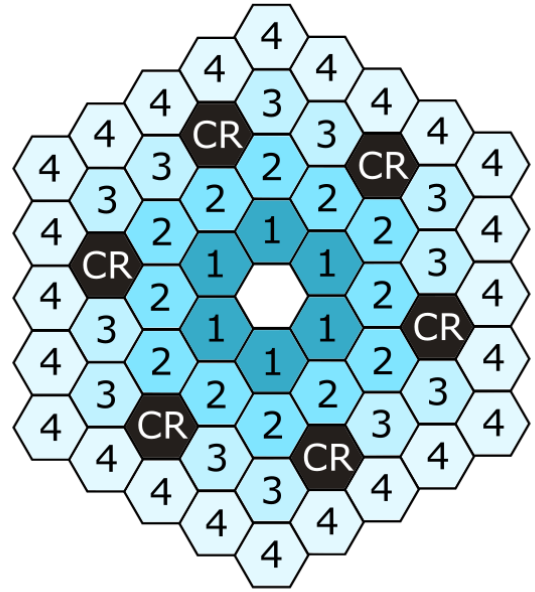
❙ Coolant Flow Distribution

Calculated necessary flow rate of assembly to ensure that maximum cladding temperature of assembly is 650℃ or less[3]

❙ Pressure Drop in Fuel Pin Bundle

Calculated using Novendstern Model[4]

$$\Delta p = M f_{smooth} \frac{L}{D_{e1}} \frac{\rho V_1^2}{2}$$



N : Flow Region Number

Classification of Flow Region

Regions	Flow region	Max. assembly power (MW)	Assembly flow rate (kg/s)	Max. pressure drop in bundle section (MPa)
Inner core	**1**	**2.53**	**10.41**	**0.45**
	2	2.34	9.64	0.40
	3	2.03	8.35	0.31
Outer core	4	1.49	6.15	0.18

References: [3] N.Takaki, N.Uto, T.Mizuno, "Design Studies on Small Fast Reactor Cores(II)," JNC TN9400 2003-066, 2003.
[4] Alan E.Waltar, Donald R.Todd, Pavel V.Tsvetkov, FAST BREEDER REACTORS, ERC Publishing Company, 2016.

5.3 Thermal-hydraulic Summary Evaluation

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Regions	Flow region	Max. assembly power (MW)	Assembly flow rate (kg/s)	Max. pressure drop in bundle section (MPa)
Inner core	**1**	**2.53**	**10.41**	**0.45**
	2	2.34	9.64	0.40
	3	2.03	8.35	0.31
Outer core	4	1.49	6.15	0.18

- Total assembly flow rate : **426 kg/s** (limit : 750 kg/s)
  - Distributable flow rate
- Max. pressure drop in bundle section : **0.45 MPa**

Joyo MK-IV : 0.2 MPa
Monju : 0.3 ~ 0.4 MPa

Pressure drop, which is 12.5% higher than Monju, can be covered with slight increase in pump power.

Reference:
[5] Y.Okano, N.Kobayashi, T.Ogawa, S.Ohki, M.Naganuma, T.Okubo, T.Mizuno, T.Ogata, N.Ueda, S.Nishimura, “Fuel and Core Design Studies on Metal Fuel Sodium-cooled Fast Reactor (3) Joint Research Report for JFY2007&2008”, JAEA-Research 2009-025, 2009.

Conclusion

6. Conclusion

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- Performed conceptual design and irradiation performance evaluation of a high-flux fast neutron spectrum irradiation reactor (HFSR) using metallic slim fuel pin

Approaches
- Metal fuel
- Reducing fuel pin diameter
- higher Pu enrichment in inner core

- High power density
- Intentional increase in radial power peaking factor

High fast neutron flux and fluence in irradiation area (core center)

- Max. pressure drop is higher than Joyo and Monju.
  - ➢ **Thermal feasibility can be achieved with a slight increase in pump power.**

Appendix

Actinium-225

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- Ac-225 is a medical radioisotope
- **Treatments using Ac-225:**
  - ➢ involve irradiation by alpha particle from inside a tumor.
    → highly effective
  - ➢ Doesn't need shielding
    → allowing relaxation of restrictions on entry into hospital room
- **Ac-225 is in short supply globally**
  - It has a half-time of only ten days
  - US, Germany and Russia are able to supply it at present
  - There are currently no high-energy neutron irradiation fields.
    → global trend is to produce neutrons using accelerators.

Reference:
https://www.jaif.or.jp/en/news/6915

Why is 100 MWt?

- **NRA Guide for Emergency Preparedness and Response**
  - Range of nuclear emergency planning zone (EPZ)
    - Power Reactor : **~ 30 km**
    - Test Reactor (10~100 MWt) : **~ 5 km**
      (2~10 MWt) : **~ 0.5 km**

- **Increasing range of EPZ affects:**
  - Larger area and more affected population
  - More complex evacuation planning

  and so on…

Sodium Void Reactivity

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- **Analyzed using direct method**

1. K-effective of the standard core
2. K-effective in sodium void composition

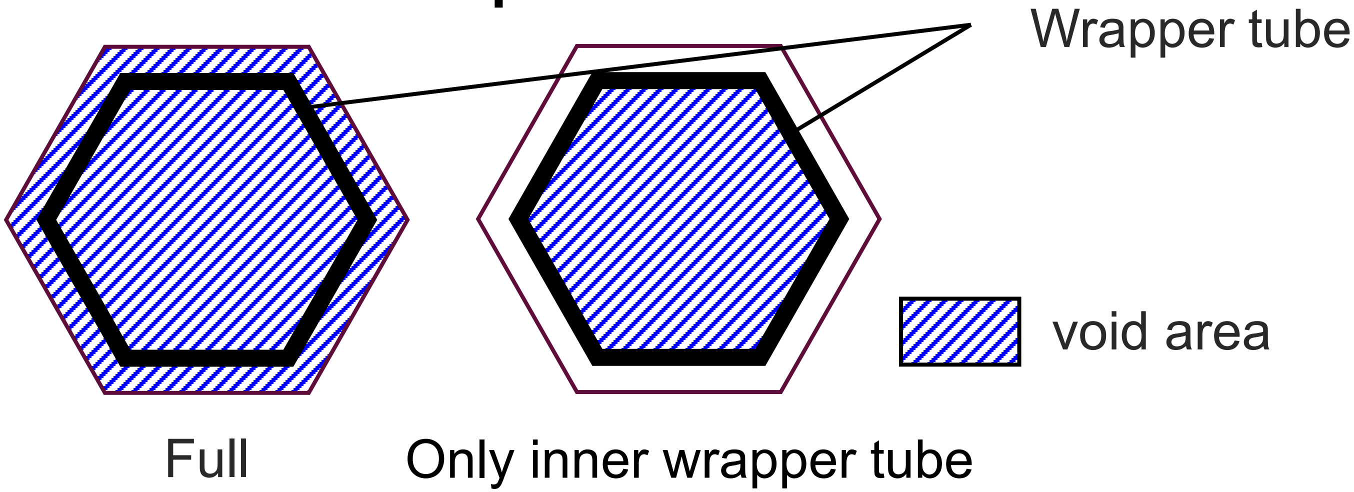
➡ Calculate sodium void reactivity from two k-effective

$$\rho[\Delta k/kk] = \frac{k' - k}{kk'}$$

k : K-effective of the standard core

k': K-effective in sodium void composition

- **2 patterns of sodium void composition**



Reference:

H.Takashita, M.Higuchi, N.Togashi, T.Hayashi, “Report on Neutronic Design Calculational Methods (Technical Report)”, JNC TN8410 2000-011, 2000.

Sodium Void Reactivity

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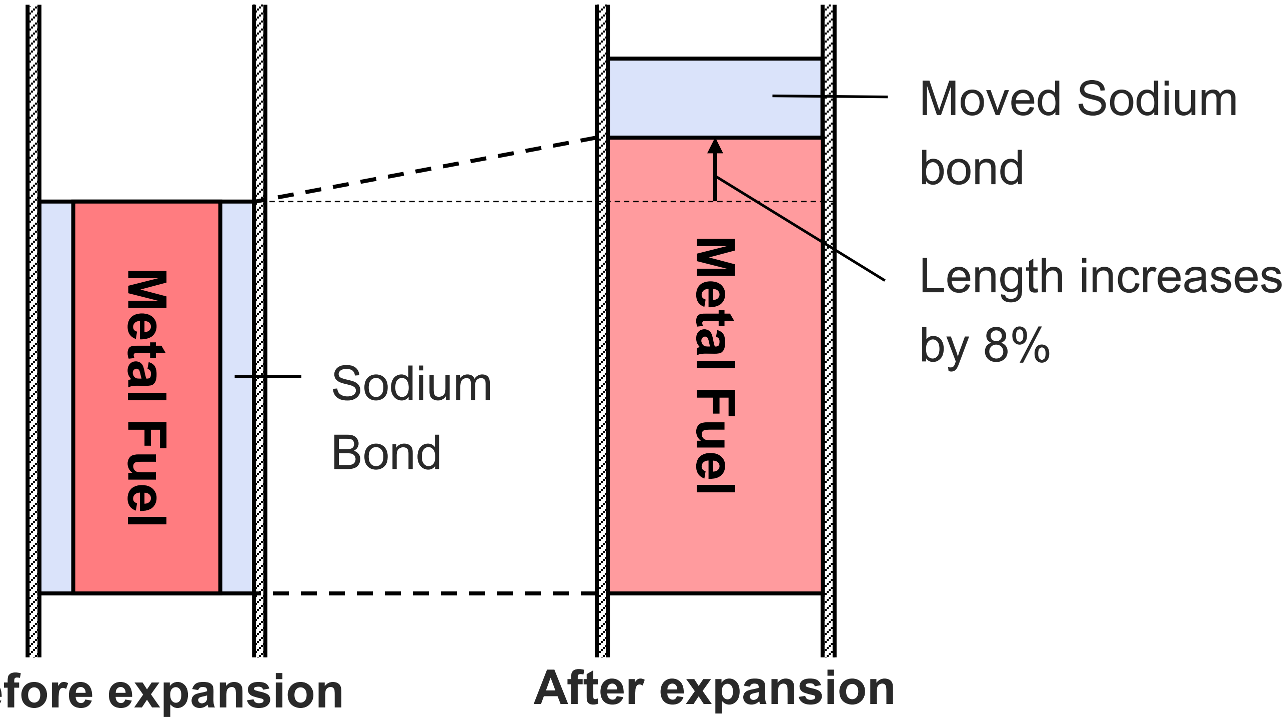
- **Considered deformation of fuel slug**
  1. Radial deformation : Expands until it encounters the cladding tube.
  2. Axial deformation : Fuel slag length increases by 8%.
  3. Sodium Bond : 80% moves to the upper of the fuel, and the remainder leaches into the fuel.



Reference:
T. Ogata et al., Nucl. Technol. 128, 113 (1999).
G.L. Hofman et al., Metallic Fast Reactor Fuels, Nucl. Mater. Part 1, 10A (1994)

Fast experimental reactors

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country		Japan		France		United States			Russia		Kazakhstan	-
Reactor name		Joyo MK-IV	Monju	Rapsodie Fortissimo	Phenix	EBR-II	FFTF	CRBRP	BOR60	BN600	BN350	HFSR
First Criticality	[-]	1977	1994	1967	1973	1961	1980	-	1968	1980	1972	-
Thermal Power	[MWt]	100	714	40	350-563	62.5	400	975	55	1470	750	100
Fuel	[-]	MOX	MOX	MOX	MOX	U-Zr	MOX	MOX	MOX	UO2	UO2	U-Pu-Zr
Number of fuel pin per FAs	[本]	127	169	61	217	91	217	217	37	127	127	169
Fuel pin outer diameter	**[mm]**	**5.50**	**6.5**	**5.1**	**6.6**	**4.42**	**5.84**	**5.84**	**6.00**	**6.9**	**6.9**	**4.78**
Cladding wall thickness	**[mm]**	**0.35**	**0.47**	**0.37**	**0.45**	**0.305**	**0.38**	**0.38**	**0.30**	**0.40**	**0.40**	**0.30**
Wrapper tube inner flat-to-flat distance	[mm]	78.5	105	49.8	124	58.2	116	116	44	96	96	78.5
Fuel pin pitch	[mm]	6.47	7.9	5.9	7.7	5.70	7.3	7.3	6.70	8	7.9	
P/D	[-]	1.18	1.22	1.16	1.18	1.29	1.24	1.26	1.12	1.17	1.15	1.18
Wire diameter	**[mm]**	**0.48**	**0.70**	**0.40**	**0.55**	**0.64**	**0.73**	**0.73**	**0.35**	**0.55**	**0.52**	**0.42**

minimum value of each parameter

References: [4] Alan E.Waltar, Donald R.Todd, Pavel V.Tsvetkov, FAST BREEDER REACTORS, ERC Publishing Company, 2016.

Thermal-hydraulic Summary Evaluation

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The following conditions were used to calculate the necessary flow rate of each assembly so that the max. cladding temperature of each assembly would be 650℃ or less[1].

- Coolant outlet/inlet temperature: 456℃ / 300℃
- Maximum temperature inside the cladding tube: 650℃
- Total flow rate of primary coolant: 750 kg/s

References: [3] N.Takaki, N.Uto, T.Mizuno, "Design Studies on Small Fast Reactor Cores(II)," JNC TN9400 2003-066, 2003.

▍Coolant Flow Distribution

Calculated necessary flow rate of assembly to ensure that maximum cladding temperature of assembly is 650℃ or less[3]

$$T_{Clad} = (\Delta T_{Na} \times F_b \times RPF + \Delta T_0) \times HSF + T_{in}$$

$$W_{req} = \frac{P_{SA}}{C_p \Delta T_{Na}}$$

T_{Clad}	[℃]	Max. cladding temperature
ΔT_{Na}	[℃]	Average coolant temperature rise at assembly outlet (mixed)
F_b		Bundle diameter temperature peaking coefficient equivalent value (1.11)
RPF		Assembly inner diameter output peaking coefficient (1.04)
ΔT_0	[℃]	Coolant-cladding temperature rise equivalent value (30)
HSF		Cladding engineering safety factor equivalent value (1.19)
T_{in}	[℃]	Coolant inlet temperature (350° C)
W_{req}	[kg/s]	Required flow rate per assembly
P_{SA}	[W]	Assembly power output
C_p	[J/Kg/K]	constant pressure specific heat（ =1263 J/kg/K ）

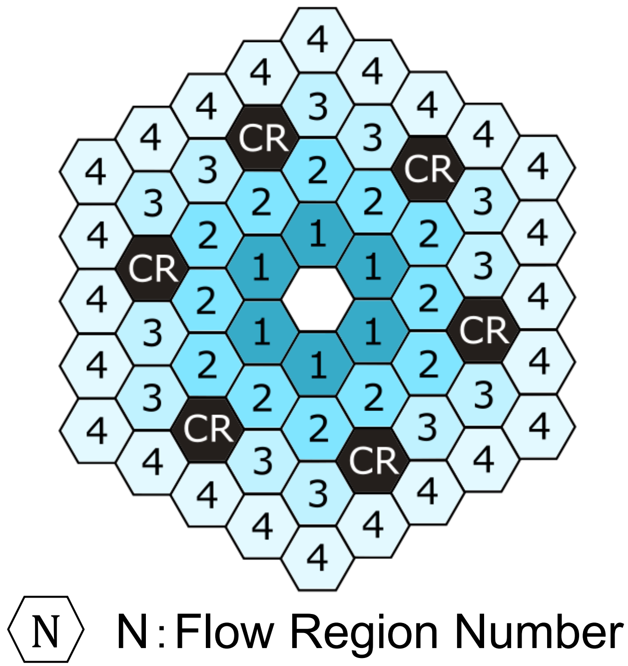
References: [3] N.Takaki, N.Uto, T.Mizuno, “Design Studies on Small Fast Reactor Cores(II),” JNC TN9400 2003-066, 2003.

Pressure Drop in Fuel Bundle Section

Calculated using Novendstern Model[4]

$$\Delta p = M f_{smooth} \frac{L}{D_{e1}} \frac{\rho V_1^2}{2}$$

Δp	[Pa]	Pressure drop
M	[-]	Multiplier
f_1	[-]	Effective friction factor in channel 1
L	[m]	Length of channel
D_{e1}	[m]	Effective hydraulic diameter
ρ	[kg/m^3]	Coolant density
V_1	[m/s]	Velocity in channel 1



Classification of Flow Region

Heavy Metal Inventory of HFSR

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		Joyo MK-4			HFSR		
		Inner core	Outer core	Inner+Outer	Inner core	Outer core	Inner+Outer
Number of fuel assemblies (-)		18	57	75	30	24	54
Number of fuel pins per FAs (-)		127			169		
Cladding internal diameter (cm)		0.480			0.417		
Core height(cm)		50			61.5		
Fuel volume of pin (cm^3)		9.05			8.41		
Fuel volume of assembly (cm^3)		1149.1			1422.1		
Density (g/cm3)	U235	1.177	1.091		0.015	0.016	
	U238	5.362	4.971		7.714	8.167	
	Pu238	0.020	0.025		0.034	0.030	
	Pu239	1.289	1.618		2.251	1.950	
	Pu240	0.469	0.588		0.818	0.709	
	Pu241	0.078	0.098		0.136	0.118	
	Pu242	0.059	0.074		0.102	0.089	
	Am241	0.039	0.049		0.068	0.059	
Inventory (kg)	U235	1.352	1.254	95.81	0.022	0.023	1.22
	U238	6.161	5.712	436.47	10.970	11.615	607.86
	Pu238	0.022	0.028	2.01	0.048	0.042	2.46
	Pu239	1.481	1.860	132.66	3.201	2.774	162.59
	Pu240	0.539	0.676	48.24	1.164	1.009	59.12
	Pu241	0.090	0.113	8.04	0.194	0.168	9.85
	Pu242	0.067	0.085	6.03	0.145	0.126	7.39
	Am241	0.045	0.056	4.02	0.097	0.084	4.93
Heavy Metal Inventory (kg)		9.758	9.783	733.279	15.841	15.841	855.418
		Fissile($^{235}U+^{239}Pu+^{241}Pu$)		236.5	Fissile($^{235}U+^{239}Pu+^{241}Pu$)		173.7
		Fertile		492.8	Fertile		676.8