The US NRC Workshop
Advanced Non-Light-Water Reactors –
Materials and Integrity - December 9-11,
2019, Maryland, USA

R&D activities for advanced reactors in JAEA

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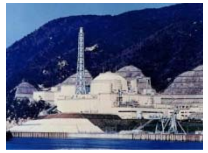
Contents

- Fast reactors
  - Approach
  - Codes and Standards
  - R&Ds on materials and structures
- High Temperature Gas-cooled Reactors
  - History and status of Japan’s HTTR
  - R&D accomplishments and plans

Fast Reactor

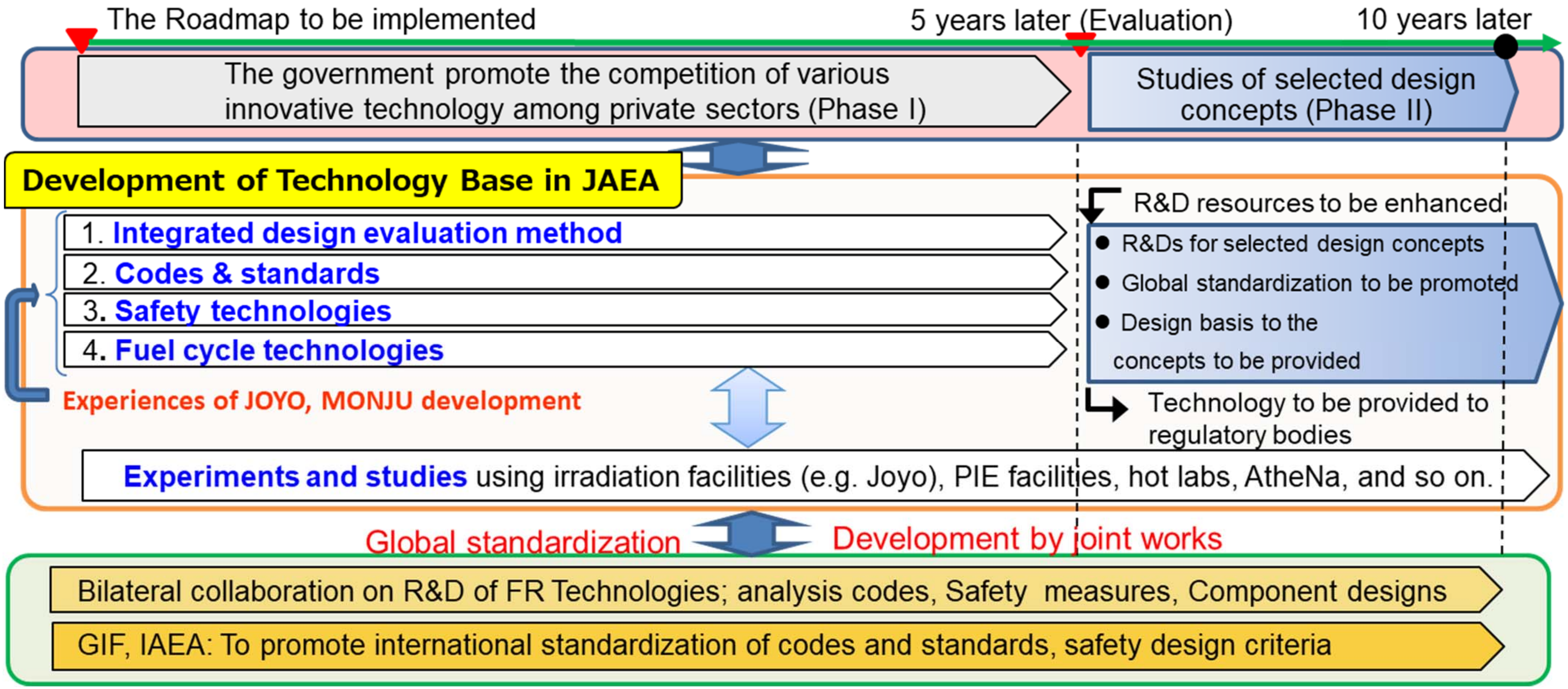
The Roadmap and Development of FR Technology base in JAEA



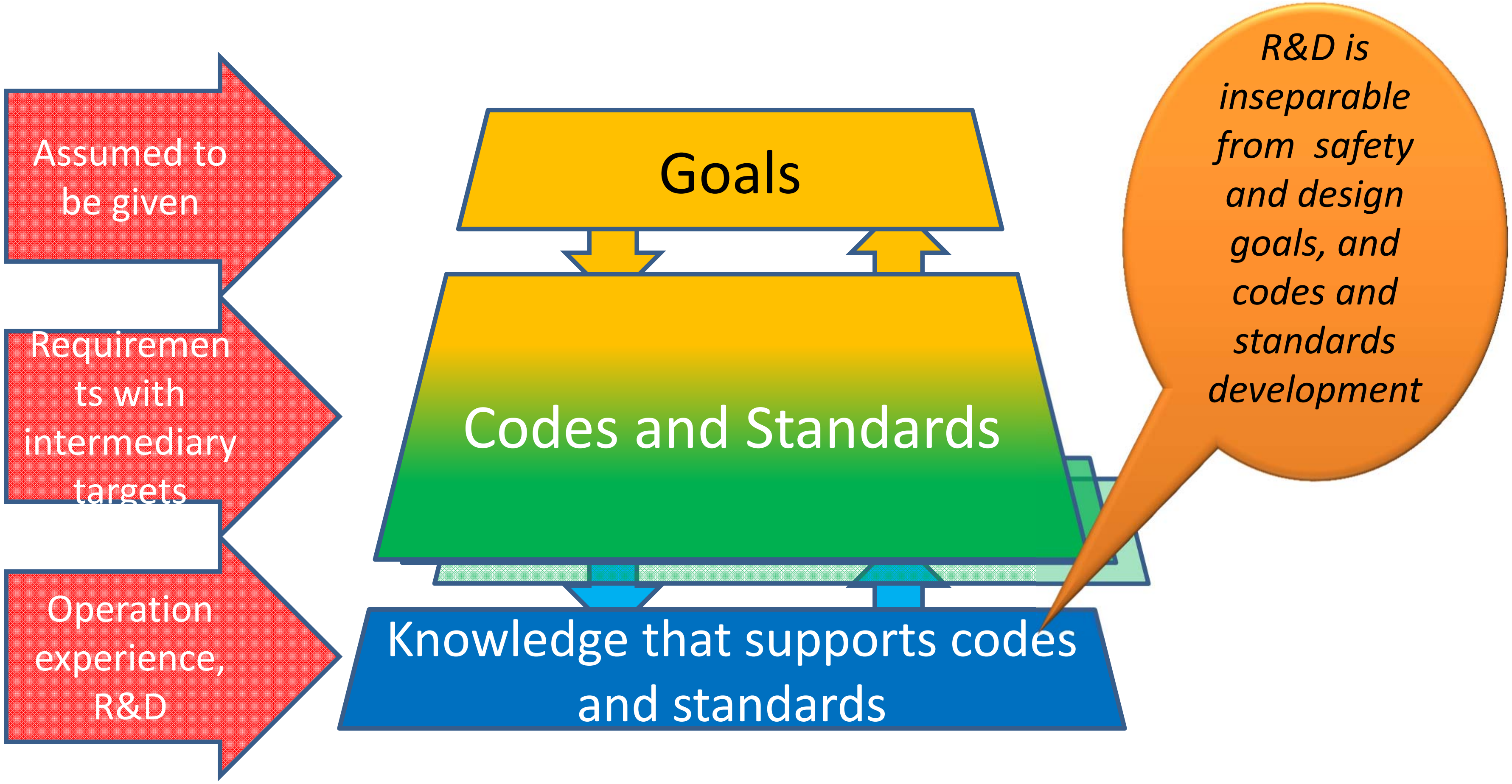
- *The Strategic Roadmap* was decided at the Inter-Ministerial Council for Nuclear Power on Dec. 21, 2018.

□ The Roadmap defines **the action plans for FR development** as follows,

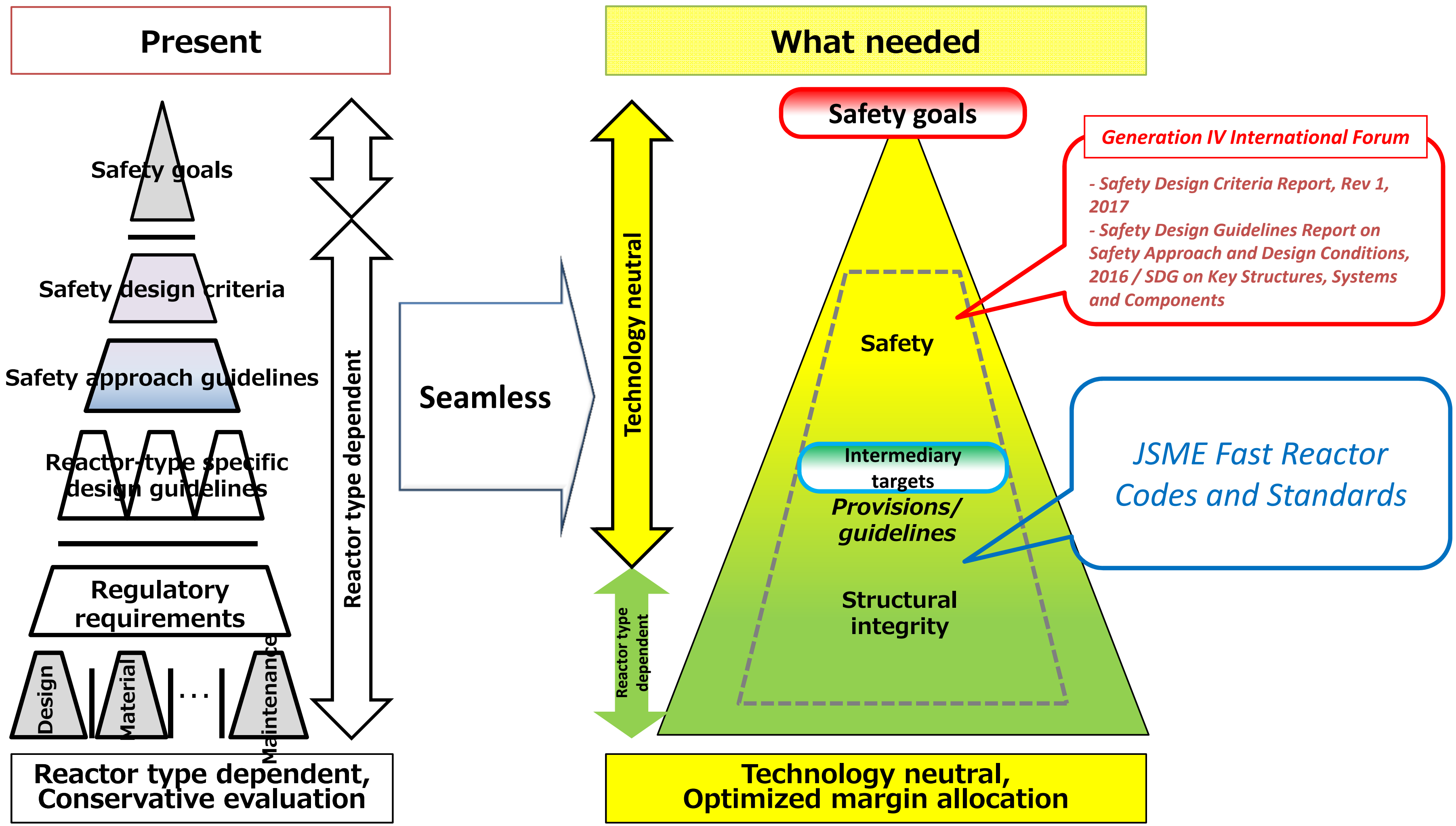
- The development process can be divided into three phases:
  - Phase 1: Promotion of competition (First ~5 years)
  - Phase 2: Narrowing down and focusing on the technology (from ~2024)
  - Phase 3: Examination of critical issues and process of the development



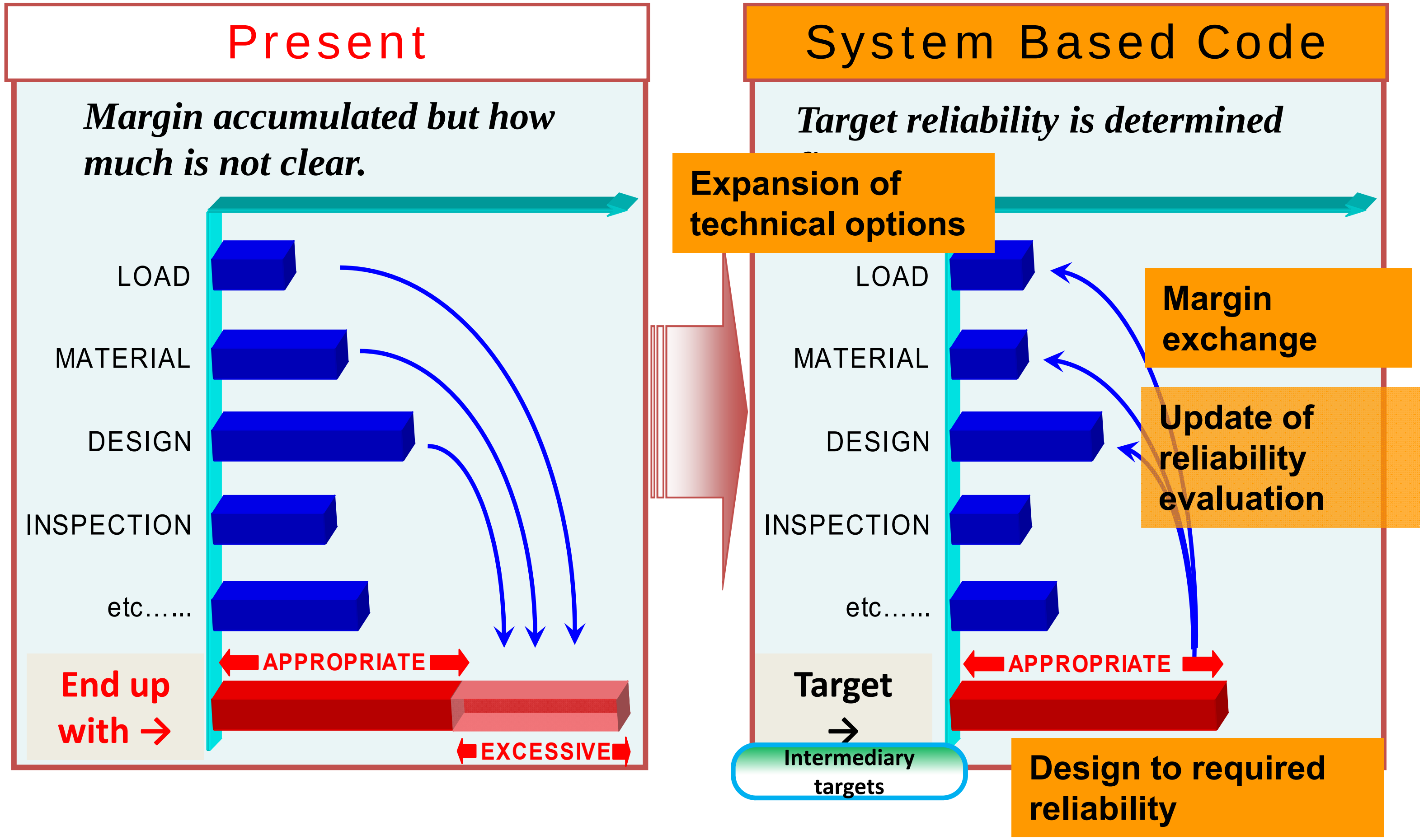
“Wholistic approach”



"Seamlessly structured C&S"

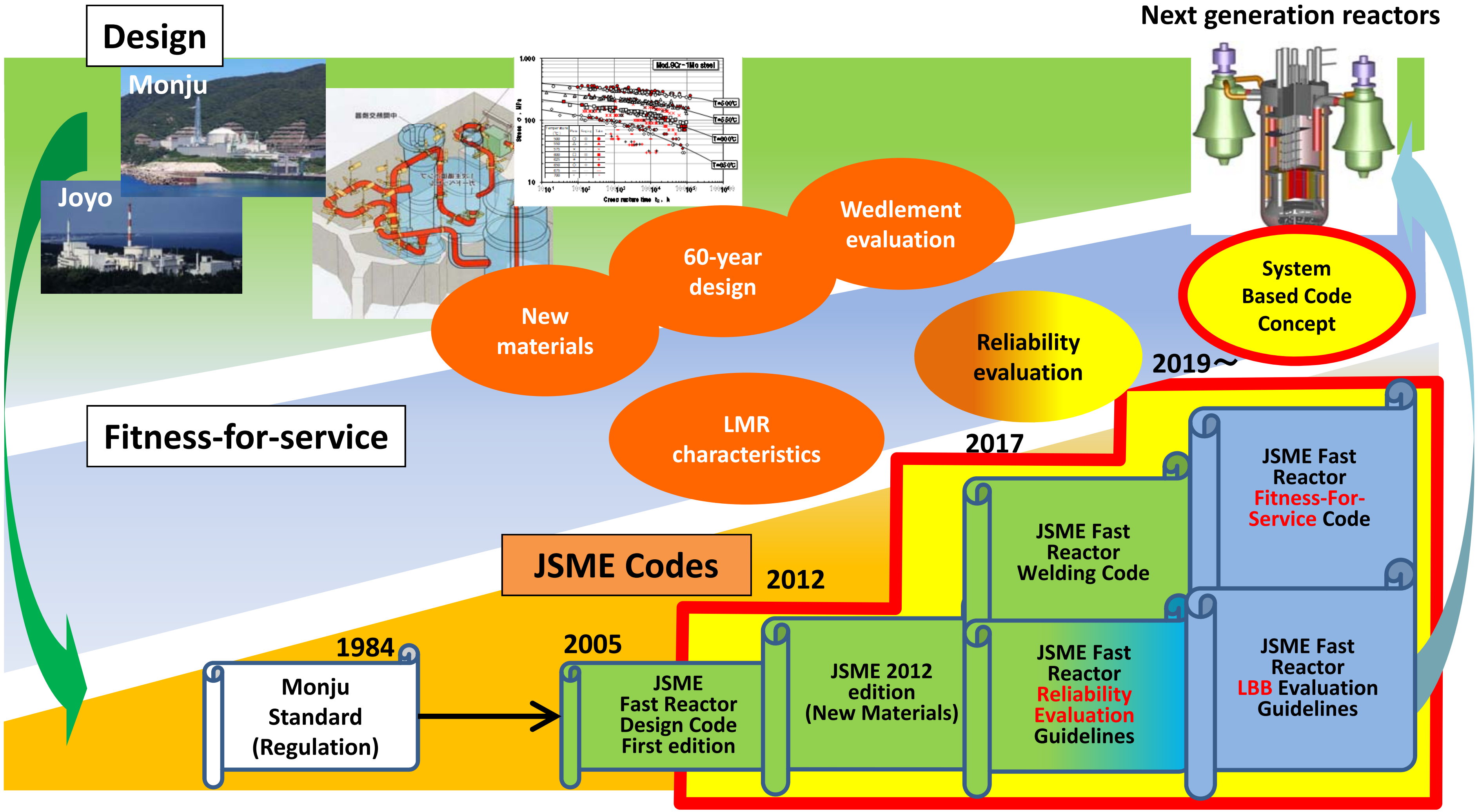


System Based Code Concept



Asada, Y., Japanese Activities Concerning Nuclear Codes and Standards – Part II, Journal of Pressure Vessel Technology, ASME 128 (2006) 64.

Fast Reactor Codes



JSME Fast Reactor Design Code

60-year design: Region splitting analysis method

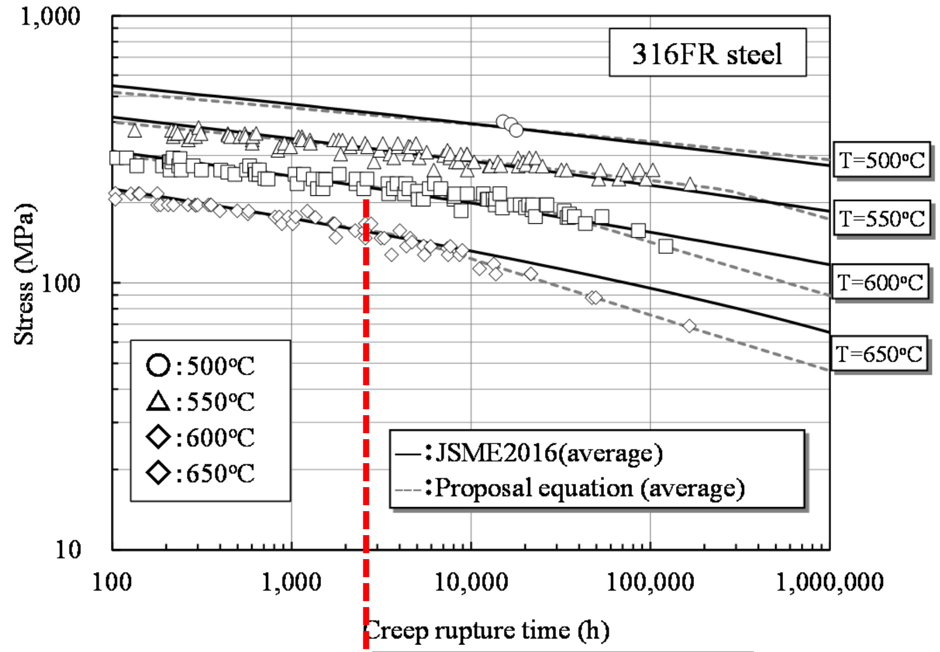
Mechanisms

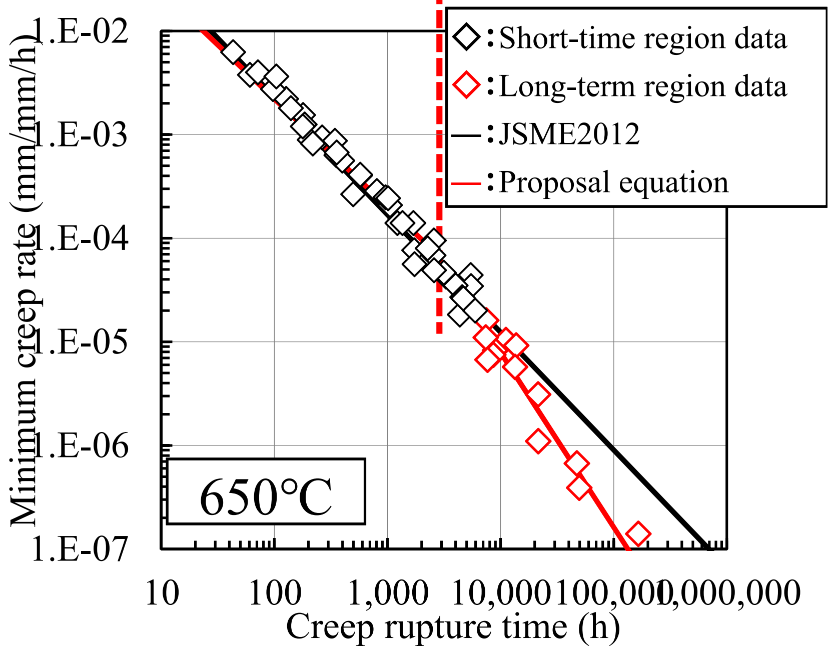
	Short term	Long term
316FR Steel	No coarse precipitates	Coarse precipitates Laves phase
Mod.9Cr-1Mo Steel	Homogeneous recovery	Heterogeneous recovery

Interception (h)

Temperature (℃)	550	600	650
316FR Steel	90,337	14,125	2,700
Mod.9Cr-1Mo Steel	25,338	3,180	500

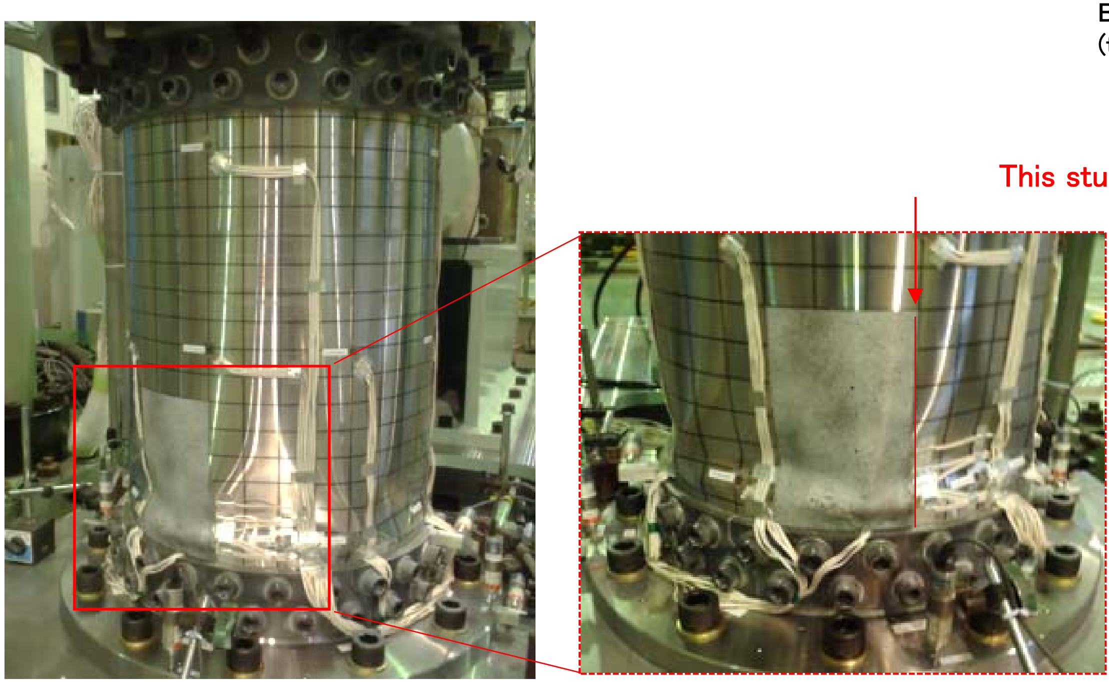
- ✔ The code covers irradiation effects and sodium environmental effects



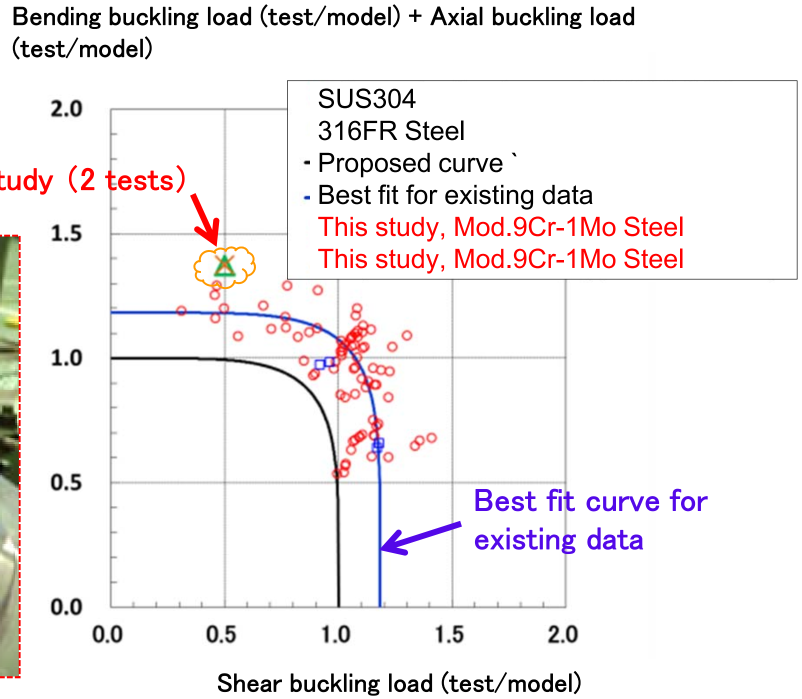


Buckling evaluation methods

- Apply to large diameter thin walled vessels:
  - <u>elastic buckling</u>
  - <u>shear buckling</u>
  - Mod.9Cr-1Mo steel (high yield stress)



Test piece after unloading



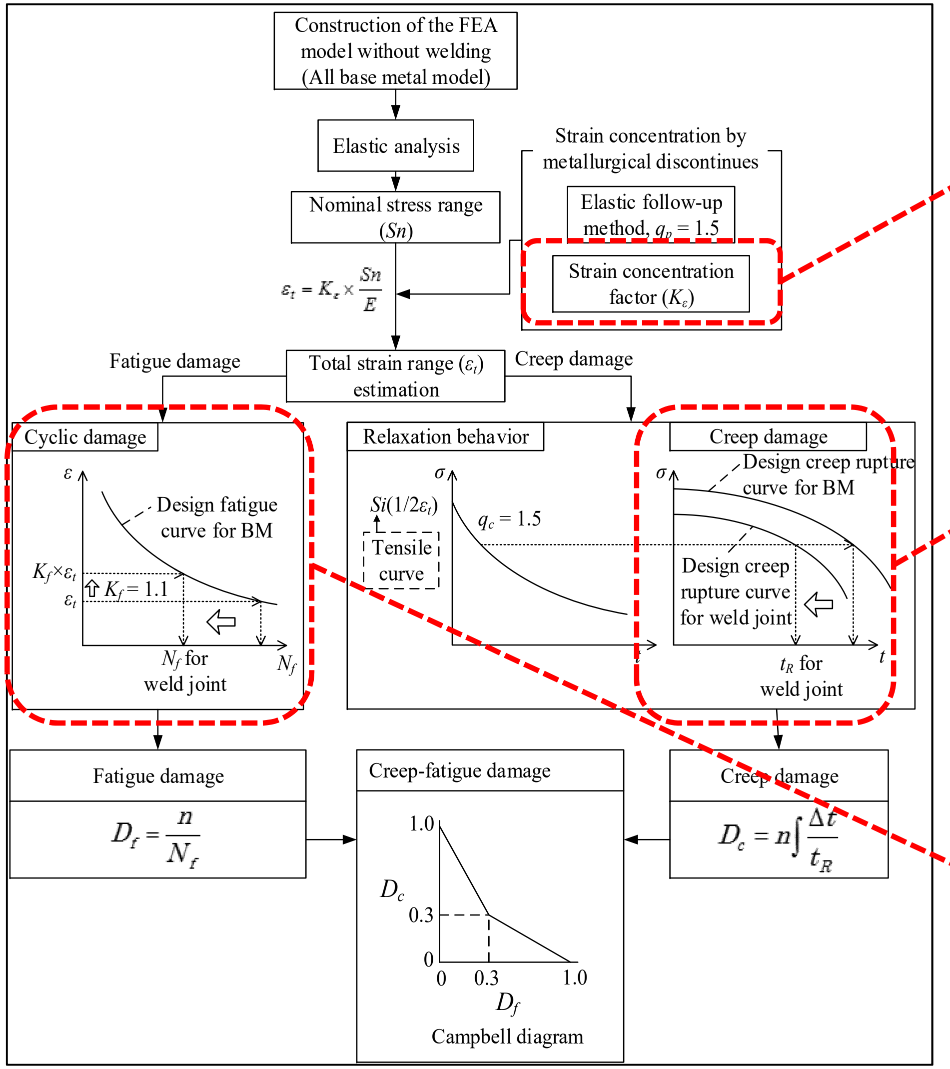
Test results and proposed curve*

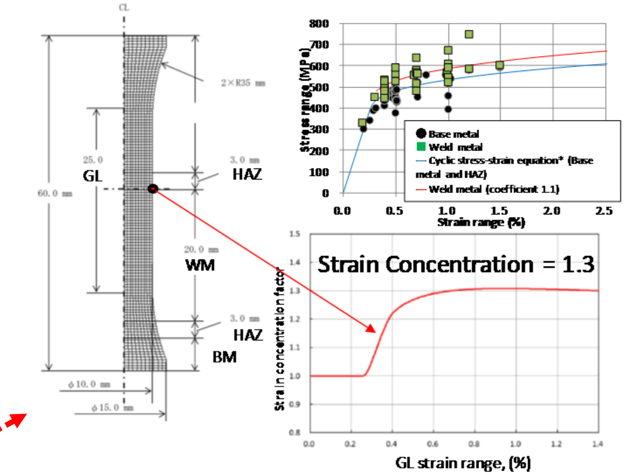
* Basic idea is based on "Akiyama, H. et al, Transactions of the JSME (in Japanese), A, Vol. 60, No. 575（1994）

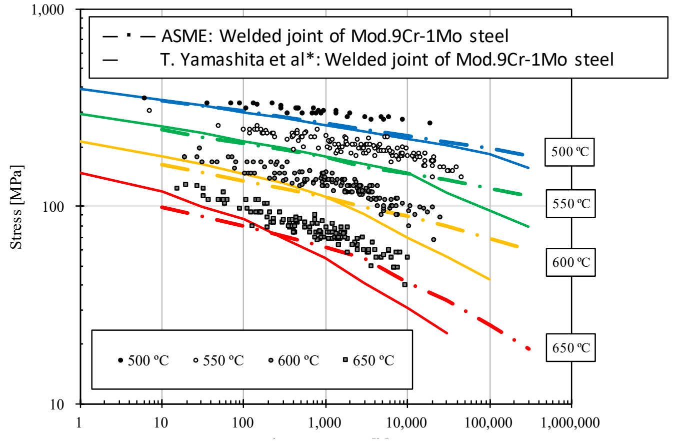
JSME Fast Reactor Design Code

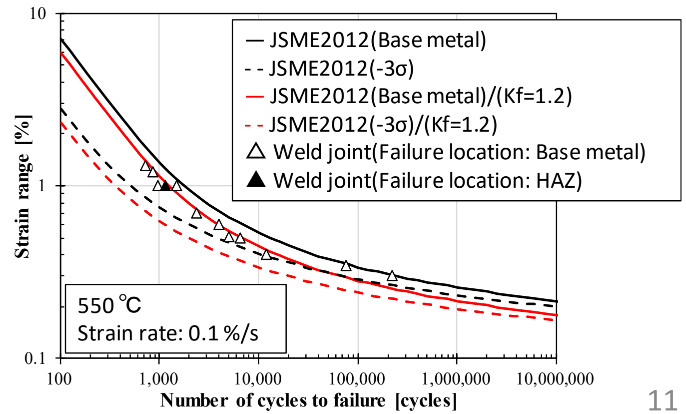
Welded joint evaluation method

Evaluation flow









JSME Fast Reactor
Fitness-For-Service Code

Leak Before Break (LBB) is satisfied for SFRs.
- ✔ Ductile Structural Material : Austenitic SS
- ✔ Low Pressure Boundary : Sodium Coolant

Continuous Monitoring is adopted.
- ✔ Sodium Leak Monitoring (SoLM)
- ✔ Ar gas Leak Monitoring (ArLM)

Periodic tests such as Visual Test (VT) and Material Surveillance are also adopted complementary.

Sodium and radioactive cover gas

	Small	Large
Retaining	**CM-2(CM-3)**	**CM-1（Leak detection sensitivity is required）**
Not retaining	**System leakage test**	**NDI and system leakage test**

Consequence of leak

Parts Examined		Rules on Fitness-for-Service for Fast Reactor	LWR
Primary coolant boundary welds	Sodium retaining parts	Continuous monitoring 1 (CM-1) Small diameter pipe: Continuous monitoring 2 (CM-2)	Volumetric examination, surface examination Small diameter pipe: System leak test and VT-2
	Cover gas retaining parts	Continuous monitoring 3 (CM-3)	—
Welded attachment		VTM-1	Surface examination

Advanced logic flow based on the System Based Code Concept has been developed in collaboration with ASME (ASME Code Case N-875)

ASME Code Case N-875

Code Case N-875, Alternative Inservice Inspection Requirements for Liquid-Metal Reactor Passive Components Section XI, Division 3

Inquiry: Under what conditions may the System Based Code (SBC) be used to determine alternative examinations to Table IMB-2500-1, Examination Categories B-A, B-B, B-J-1, B-J-2, and B-N, when examining Class 1 liquid-metal-retaining components and their integral attachments in accordance with Section XI, Division 3, IMB-2500?

Reply: It is the opinion of the Committee that the examination methods shown in Tables 2A through 2E of this Case may be used as an alternative to the methods shown in Table IMB-2500-1, Examination Categories B-A, B-B, BJ-1, B-J-2, and B-N, provided the following requirements are met.

Table 1 Alternative requirements to TABLE IMB-2500-1 EXAMINATION CATEGORIES, B-A

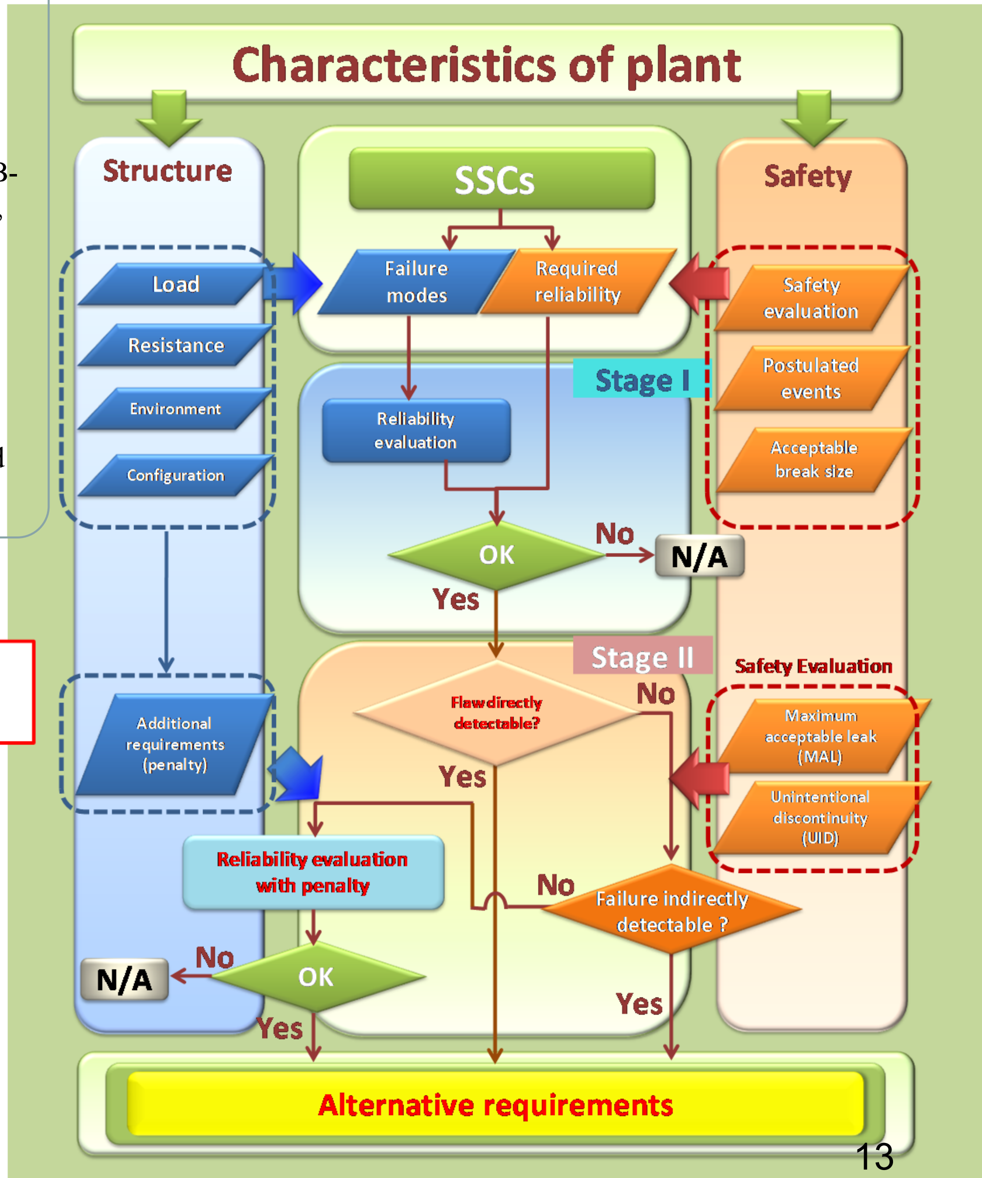
Table 2A
Alternative Requirements to Table IMB-2500-1 – Examination Category B-A

EXAMINATION CATEGORY B-A, LIQUID-METAL-RETAINING WELDS IN VESSELS PROTECTED BY GUARD

Item No.	Parts Examined	Examination Method [Note (1)]	Acceptance Standard	
B1.10	– Reactor Vessel – Longitudinal and Circumferential Shell Welds – Meridonal and Circumferential Head Welds – Shell-to-Flange Welds – Shell-to-Head Welds – Nozzle-to-Vessel Welds – Nozzle-to-Pipe Welds	Continuous monitoring [Note (3)] [Note (4)]	[Note (5)]	
B1.20	– Primary Pump Tanks [Note (2)] – All Liquid-Metal-Retaining Welds (Including Nozzle to Tank and Nozzle-to-Pipe)	Continuous monitoring [Note (3)] [Note (4)]	[Note (5)]	All Welds
B1.30	– Intermediate Heat Exchanger [Note (2)] – All Liquid-Metal-Retaining Welds in Shell – Nozzle-to-Shell Welds – Tubesheet-to-Shell Welds	Continuous monitoring [Note (3)] [Note (4)]	[Note (5)]	All Welds

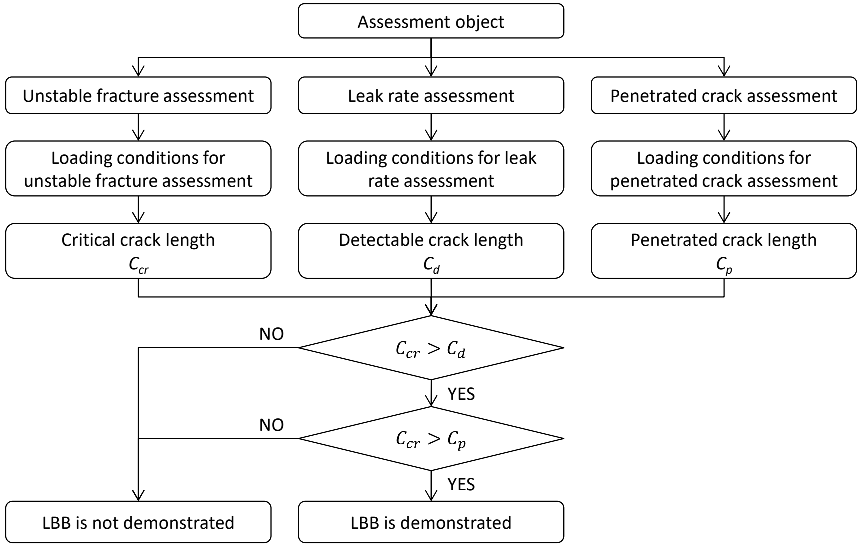
VTM-2 removed

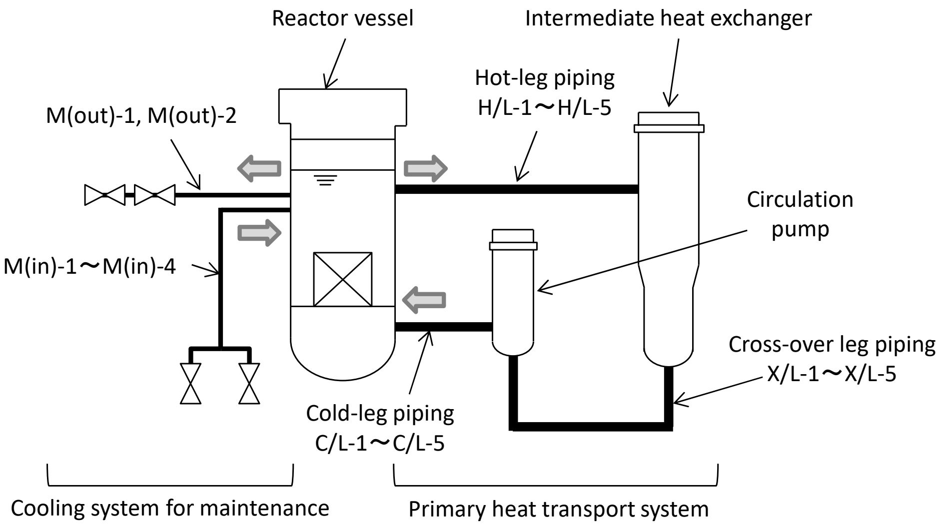
NOTES:
(1) The system design shall consider the access requirements for performance of alternative or additional examinations to those specified herein if structural defects or indications are revealed that might require such examinations.
(2) Pertains to components in loop-type primary systems, not to immersed components in pool-type primary systems.
(3) Shall be provided by installed leak detection systems capable of monitoring the exterior of a liquid-metal-containing systems and providing visual and audible alarms when leakage of liquid metal occurs.
(4) It is not the intent that all leak detectors be in service 100% of the time. The maximum percentage of leak detectors that may be out of service at any one time shall be specified in the Technical Specifications.
(5) Leakage indications shall be evaluated as confirmed or unconfirmed in accordance with the procedure determined by the Owner in advance, including the time for confirmation of leakage. Leakage indications shall be evaluated as confirmed if confirmation takes longer time than the determined time. Confirmed liquid metal leaks shall be cause for immediate shutdown of the system. The system shall be unacceptable for service until the source of the indicated leak has been identified and isolated, or repaired; the component containing the indicated leak shall be unacceptable for service until the leak has been repaired. Unconfirmed indications shall be considered faults of the monitoring system, and the leak detectors shall be repaired to meet the minimum percentage of working leak detectors required in [Note (4)].



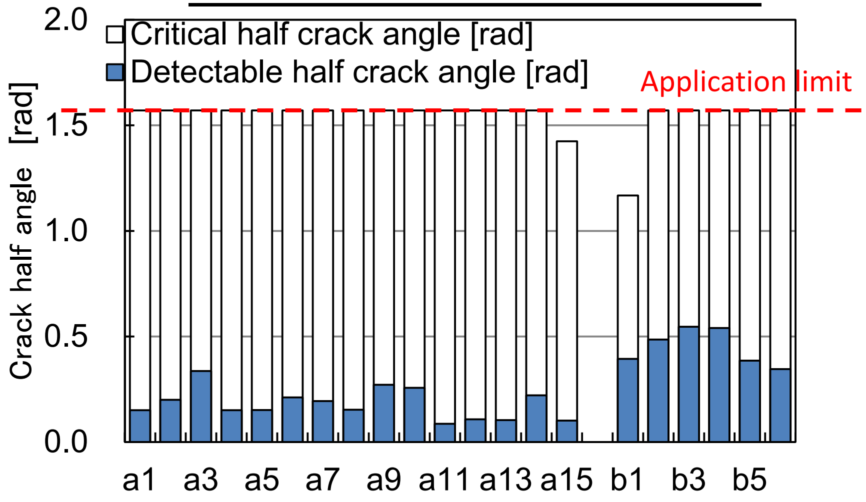
JSME Fast Reactor LBB Evaluation Guidelines

Logic flow and application

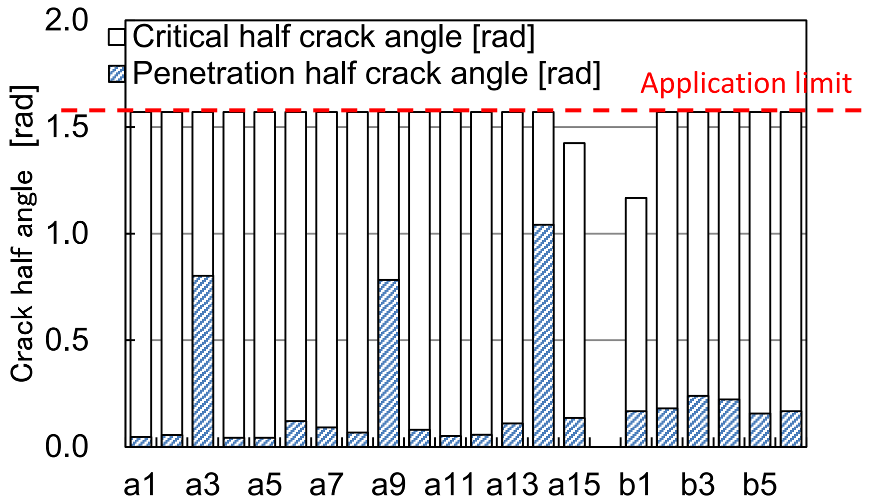




- **Circumferential cracks**



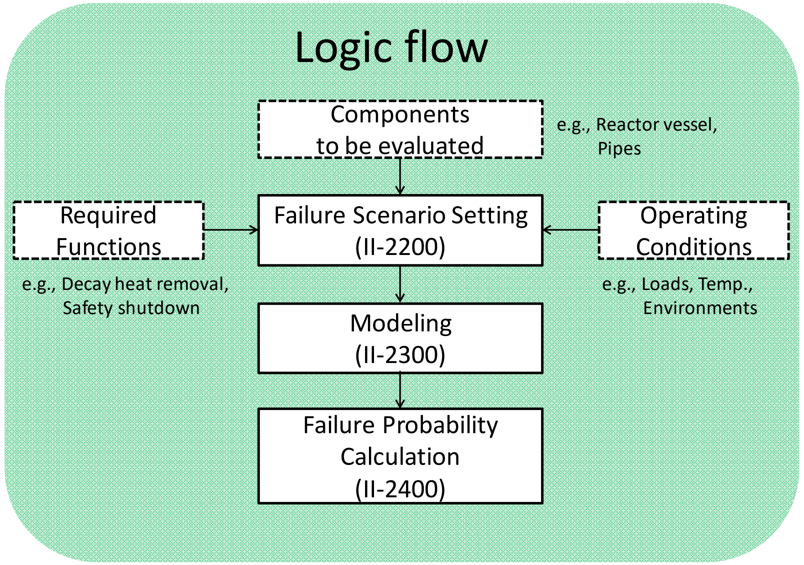
Detectable crack length



Penetration crack length

JSME Fast Reactor
Reliability Evaluation Guidelines

Logic flow and application



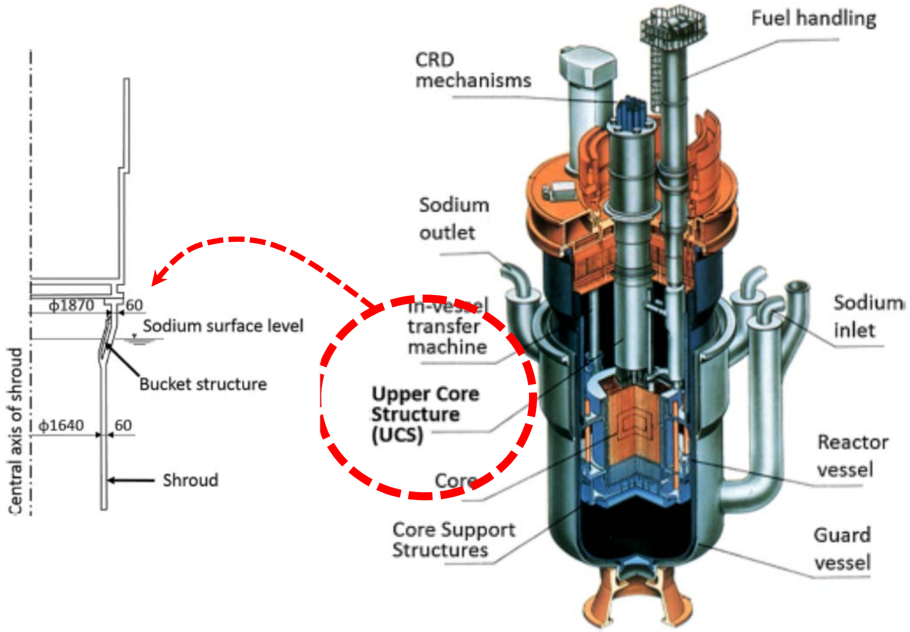


Table 2 Evaluation parameters

Parameter	Value
Material	304SS
Outer diameter (mm)	1875
Wall thickness (mm)	60
Operating temperature (°C)	530
Temperature in cold shutdown (°C)	180
Steady primary stress (MPa)	1
Cyclic stress range (MPa)	350
Axial direction stress range (MPa)	
Primary membrane	0
Primary bending	0
Secondary membrane	0
Secondary bending	175
Shakedown range (MPa)	214
Elastic follow-up factor	2.0
Strain rate (mm/mm/s)	1.0×10^{-8}
Number of cycles per year	15.3

Table 3 Random variables

Random variable	Distribution type	Median	Logarithmic standard deviation, σ_L
Thermal stress factor, χ	Log-normal	1	0.078
Creep rupture time factor, α_R	Log-normal	1	0.678
Creep strain factor, α_c	Log-normal	1	0.824
Fatigue life factor, α_f	Log-normal	1	0.471
Coefficient for creep crack growth, C_c	Log-normal	1.59×10^{-2}	0.422
Coefficient for fatigue crack growth, C_f	Log-normal	6.35×10^{-5}	0.422

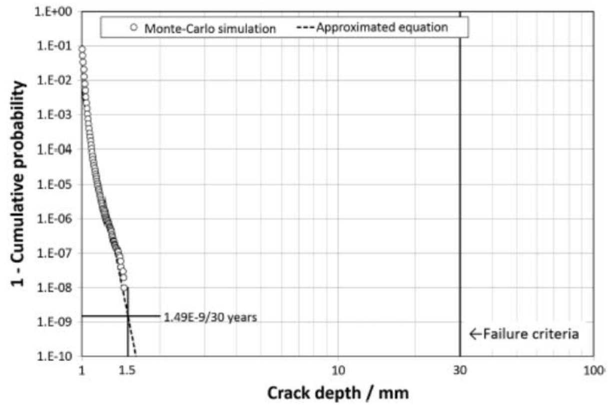


Fig. 5 Probability distribution of crack depth

*Numbers in the charts indicate the chapters of the Appendix II of ASME Code Case N-875

*JSME Guidelines provide information on probabilistic density functions for user's convenience

A scheme in development

ARKADIA

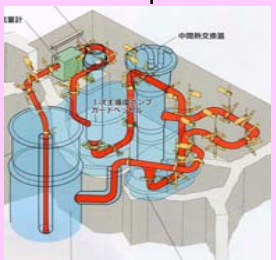
Advanced **R**eactor **K**nowledge- and **A**I-aided **D**esign **I**ntegration **A**pproach through the whole plant lifecycle

Objectives

- Design efficiency improvement for plant lifecycle optimization
- Knowledge management to support design innovation and technology transfer

INPUT

- Plant designs
  - Initial plant designs
  - Advanced plant



- Evaluation Indexes
  - Development goals
  - Design requirements

ARKADIA

Platform (Utilization of AI technology)

Analysis module (Virtual Plant)

- Plant dynamics
- CFD
- Neutronics
- Structural integrity
- Chemical issue,,,

Design optimization module

- Appropriate Design parameters
- Design evaluations for safety,,,
- Optimize by AI

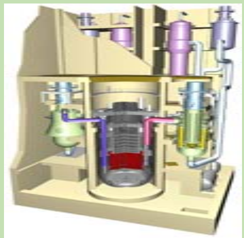
Knowledge Base module

- Plant data, Operating experience
- Design measures
- Codes and standards, database
- Evaluation methods, validation data,,,

OUTPUT

(Function 1)

- Optimized design specifications
  - Core
  - Reactor structure
  - Maintenance and monitoring
- Evaluation of design indexes
  - Safety
  - Reliability
  - Economy,,,



(Function 2) Extraction of appropriate knowledge and guidance

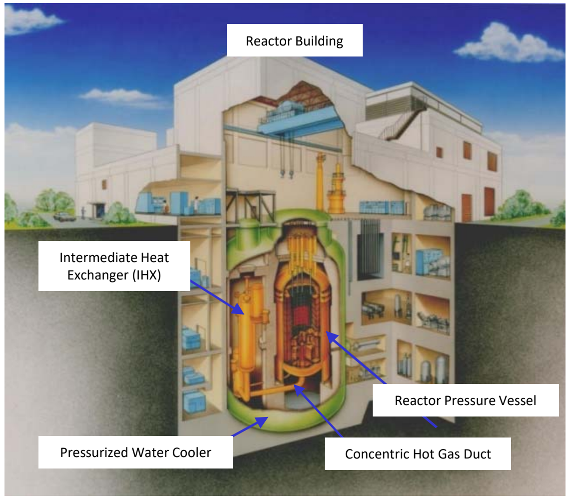
High temperature gas-cooled reactor

History and status of Japan's HTTR

Japan's HTTR

Purpose

- ***Establishment of HTGR technologies***
- ***Establishment of heat application technologies***



Specifications

Reactor thermal power	***30MW***
Coolant	***Helium gas***
Reactor inlet temperature	***395°C***
Reactor outlet temperature	***850°C, 950°C***
Core material	***Graphite***
Fuel	***UO_2 coated particle fuel***
Uranium enrichment	***3% - 10% (Av. 6%)***

Proposal for prototype commercial system

Establishment of fundamental technologies

First in the world

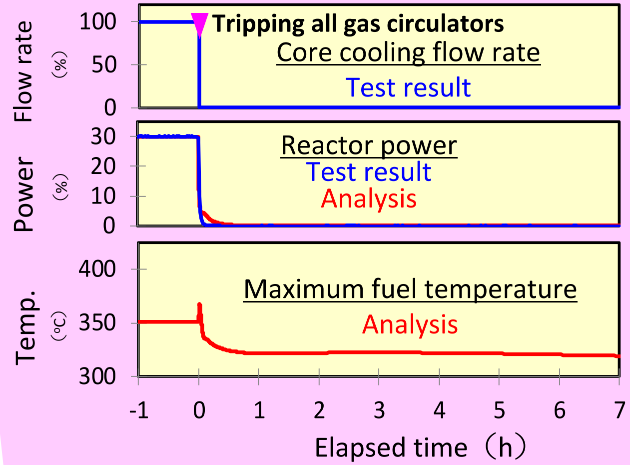
Construction of reactor

Research and development and design

2014	*Conformity review on new regulatory requirements by NRA*
2010	***Loss of forced cooling test*** *(from reactor power 30%)*
	950 °C / 50 d operation
2007	*850 °C / 30 d operation*
2004	***Attainment of reactor outlet coolant temperature 950°C***
2002	*Safety demonstration test (Control rod withdrawal test)*
2001	*Attainment of reactor outlet coolant temperature 850°C*
1998	***First criticality***
1997 ~ 1991	*Construction*
1990 ~ 1989	*Application and permission of construction*
1988 ~ 1985	*Detail design*
	HTTR
1984 ~ 1981	*Basic design*
1980 ~ 1974	*System integrity design*
1973 ~ 1969	*Conceptual design*
	Experimental multi-purpose Very High Temperature Reactor (VHTR)

Results of loss of forced cooling test

Reactor is naturally shut down as soon as the core cooling flow rate become zero. Reactor is kept stable long after the loss of core cooling.



Research and development

Fuel & Material

In-pile helium loop (OGL-1) installed at JMTR

Reactor physics

Very High Temperature Reactor Critical assembly (VHTRC)

Thermal hydraulics

Helium Engineering Demonstration Loop (HENDEL)

HTGR: Accomplishment from HTTR R&D

Experiences of design, construction and operation
(MHI, Toshiba/IHI, Hitachi, Fuji Electric, KHI, etc.)

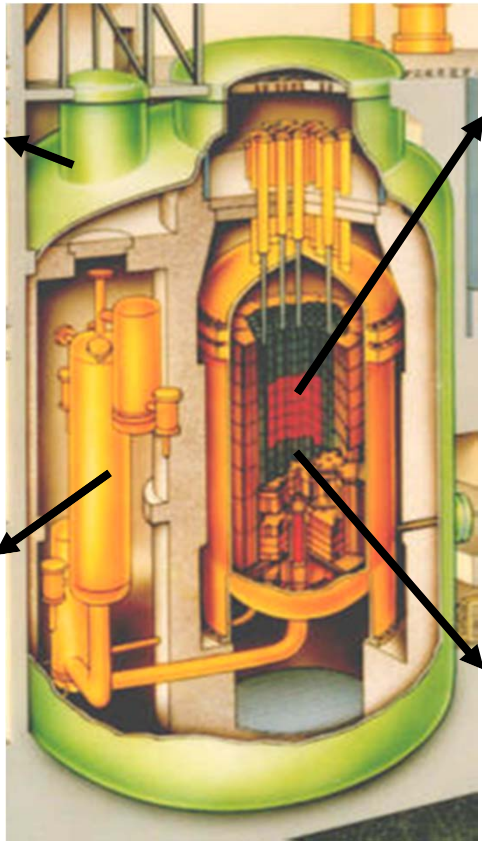
Numerous technical data accumulation of HGTR

Optimum design for commercial HTGR is possible.

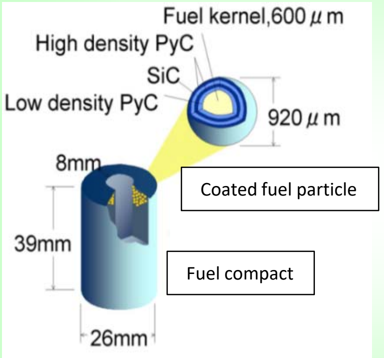
High temperature resistant metal, Hastelloy XR (collaborated with Mitsubishi Material)

Hastelloy XR is applicable at 950°C ,the world's highest temperature for nuclear structural material.

IHX can deliver hot helium gas at 950°C to outside the reactor pressure vessel.



Fuel (Nuclear Fuel Industry)

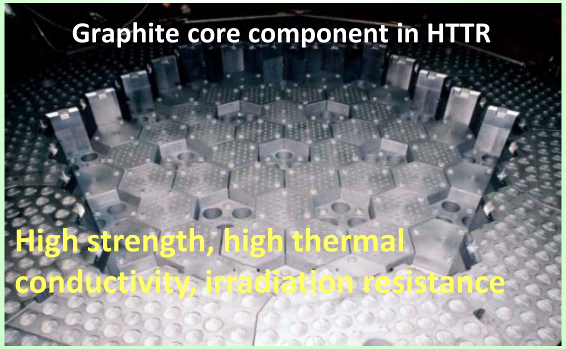


Ceramics coating layer retains fission products inside the coated fuel particle at extreme low leak level.

Ceramics coating is stable for long-term. (3 times higher burnup than LWR)

Graphite, IG-110
(collaborated with Toyo tanso)

World's highest quality graphite (isotropic, high density)



Future R&D Plan on the High Burnup Fuel (1/2)

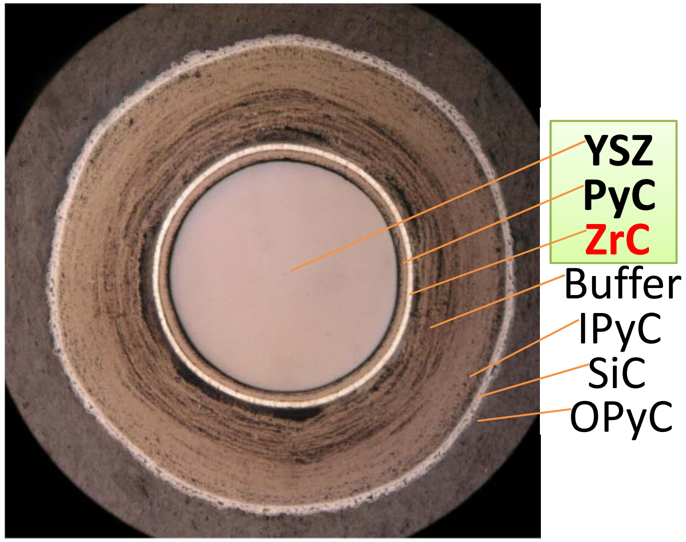
- JAEA has progressed to design 4 practical HTGRs in the basis of the developed HTTR technologies;
  - **Small-type HTGRs** ; **HTR50S, High Performance Commercial (HPC) HTGR**
  - **VHTR proposed in GIF** ; **GTHTR300**
  - Plutonium (Pu)-burner HTGR ; a HTGR concept to reduce Pu inventory by combustion using the inherently safe HTGR
- Burnup of each practical HTGR is 3 to 4 times higher than that of HTTR

Items / Reactor	HTTR (Saito 1994)[1]	HTR50S (Ohashi 2011)[2]	HPC HTGR (Fukaya 2018)[3]	GTHTR 300 (Kunitomi 2004)[4]	Pu-burner HTGR (Okamoto 2018)[5]
Thermal power	30 MW	50 MW	165 MW	600 MW	600 MW
Outlet / inlet Temp.	850 - 950 °C / 395 °C	750 °C / 325°C	750 °C / 325 °C	850 °C / 587 °C	850 °C / 587 °C
Max. Burnup / periods	33 GWd/t / 660 days	100 GWd/t / 730 days	100 GWd/t / 1200days	155 GWd/t / 730 days	625 GWd/t / 430 days
Average power density	2.5 MW/m^3	3.5 MW/m^3	3.8 MW/m^3	5.4 MW/m^3	5 - 6 MW/m^3

(1) S. Saito, et al., Design of High Temperature Engineering Test Reactor (HTTR), JAERI1332 (1994).
(2) H. Ohashi, et al., Conceptual design of small-sized HTGR system for steam supply and electricity generation (HTR50S), Proceedings of SMR2011 (2011).
(3) Y. Fukaya, et al., Conceptual Design Study of a High Performance Commercial HTGR, Proceedings of HTR2018 (2018).
(4) K. Kunitomi, et al., Design study on Gas Turbine High Temperature Reactor (GTHTR300), Transactions of the Atomic Energy Society of Japan, Vol.18[4] (2002) [in Japanese].
(5) K. Okamoto, et al., Study on Pu-burner High Temperature Gas-cooled Reactor in Japan – Concept, Proceedings of HTR2018 (2018).

Future R&D Plan on the High Burnup Fuel (2/2)

- Requirements for the high burnup fuel
  - To extend lifetime of the fuel due to smaller fuel kernel
  - For solutions
    - Technologies to increase particle packing fraction in the fuel compact; ~37 volumetric percent as the current status
    - Fuel technology to adopt larger fuel kernel would be needed
- ZrC-coated-UO_2 TRISO-CFP
  - ZrC eliminates kernel migration and internal gas then increases burnup
    - Larger UO_2 kernel could be applied
  - Both TRISO and ZrC coating technologies were matured in Japan
  - Technologies developed in the future
    - Continuous PyC-ZrC-TRISO coat
    - Irradiation tests and PIEs to demonstrate and model fuel performance



An example of the ZrC-coated TRISO-CFP demonstrated by JAEA (ZrC coating on YSZ surrogating UO_2) and NFI, Ltd. (TRISO coatings).
Ref.: S. Ueta et al., ICONE27-2138

Concluding remarks

- Various R&Ds are going on within JAEA for next generation fast reactors and high temperature gas-cooled reactors.
- Codes and standards are being developed in JSME for fast reactors.
- International collaboration is of great value in these activities.