NRC Workshop Advanced Non-Light Water Reactors – Materials and Component Integrity

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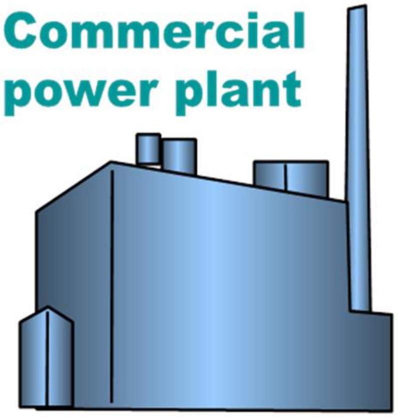
Sodium cooled fast reactors (SFR)
– Japan' s experience and future

1. Fast reactor development in Japan
(Experimental reactor Joyo, Prototype FBR NP Monju, Future)
2. Design future of Sodium cooled Fast Reactor(SFR) and deterioration of sodium retaining components.
3. Surveillance and monitoring experience of Joyo and Monju
4. Activities towards future
5. closing

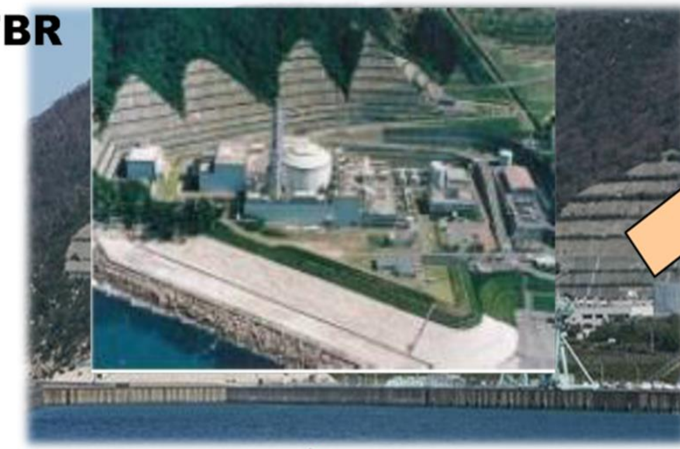
Joyo and Monju toward Commercial Reactors

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

Commercial power plant



To confirm the performance of sodium cooled FBR power plant

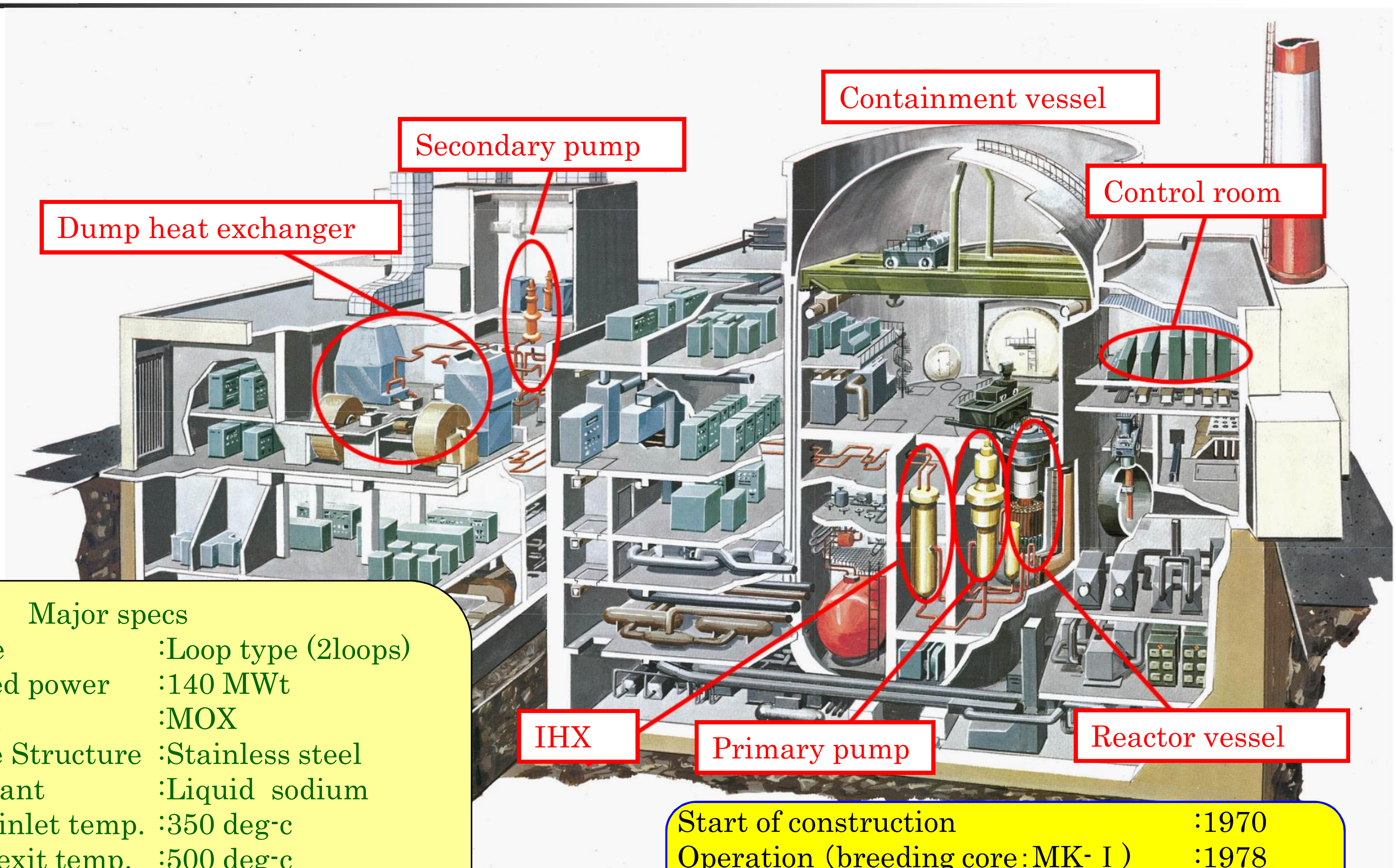


Experimental fast reactor JOYO

140MWt

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

Outline of Joyo



Major specs

Type	:Loop type (2loops)
Rated power	:140 MWt
Fuel	:MOX
Core Structure	:Stainless steel
Coolant	:Liquid sodium
R/V inlet temp.	:350 deg-c
exit temp.	:500 deg-c
Core dia.	:80cm
Core height	:50cm

Start of construction	:1970
Operation（breeding core：MK-Ⅰ）	:1978
（irradiation core：MK-II）	:1983
Upgrading core（MK-III）criticality	:July 2003
Start of irradiation with MK-III core	:May 2004

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

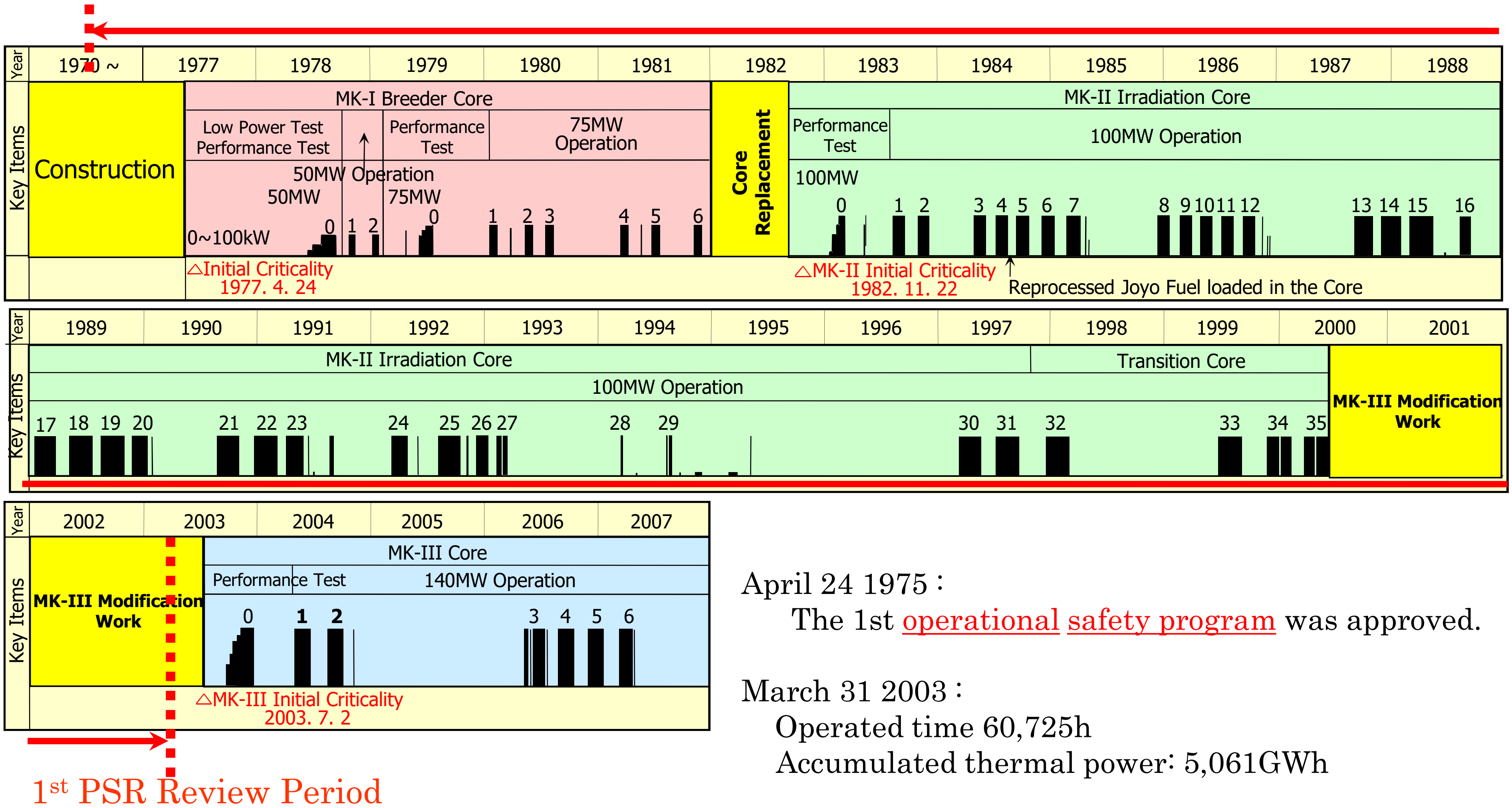
MK-Ⅰ

1981

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

1977.4 Attain First Criticality

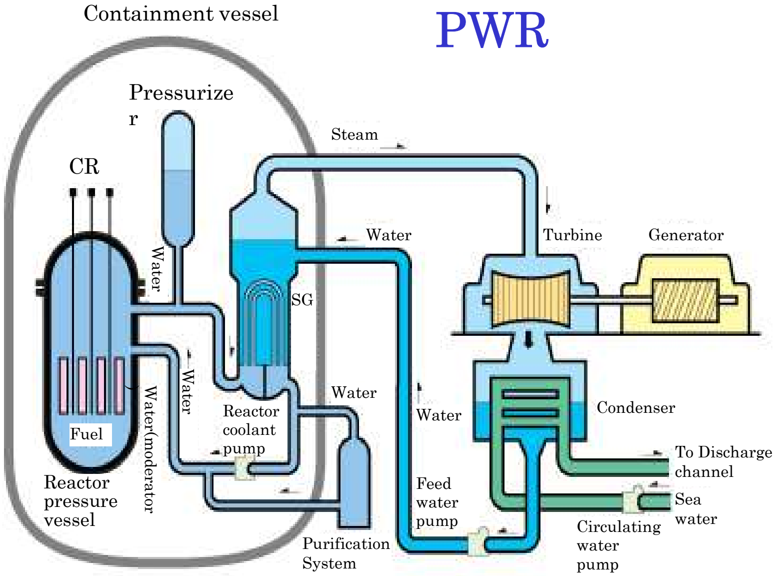
Operating history of Joyo

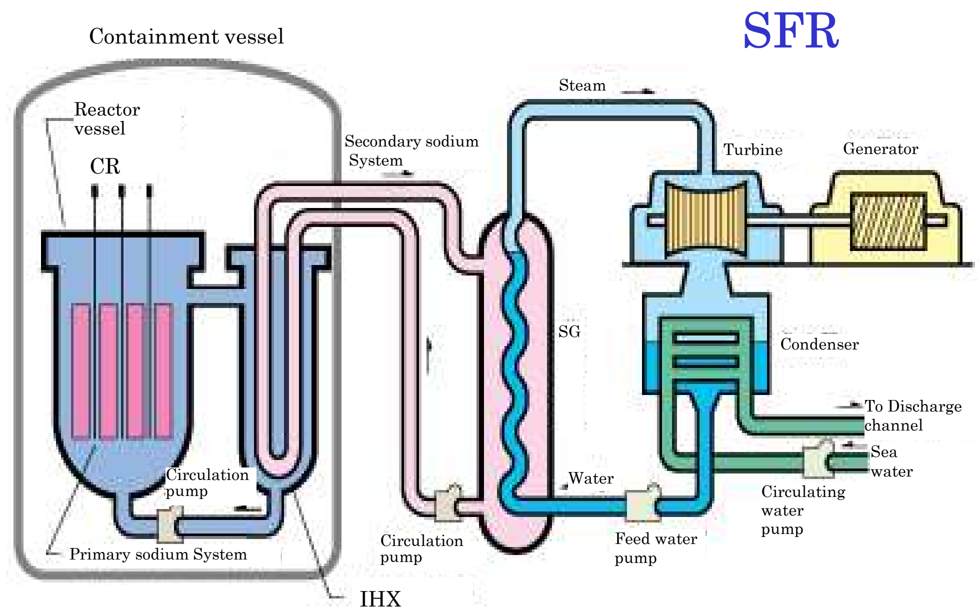


April 24 1975 :
The 1st operational safety program was approved.

March 31 2003 :
Operated time 60,725h
Accumulated thermal power: 5,061GWh

Feature of SFR





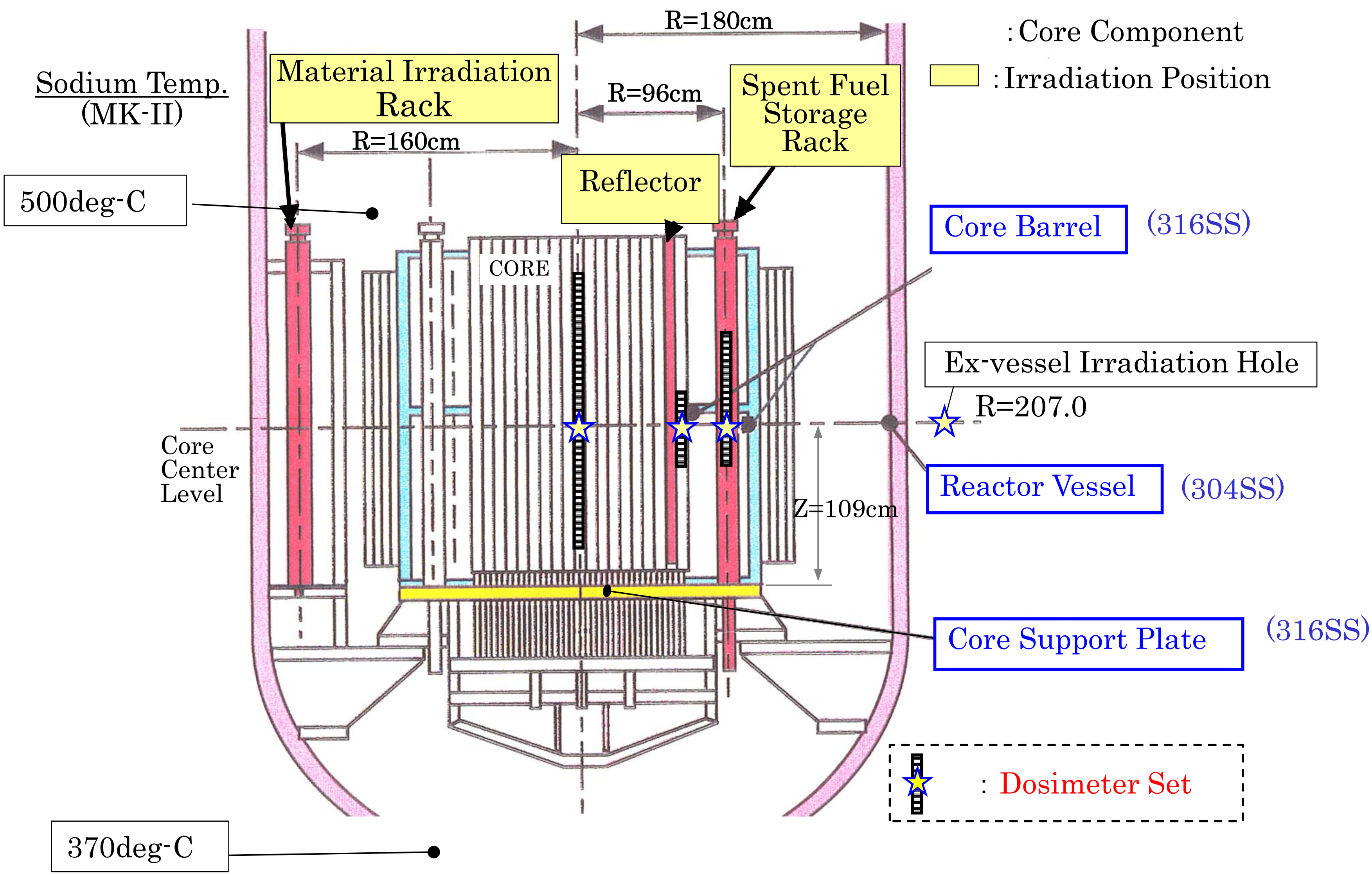
Feature of SFR comparison with LWR

Subject	LWR (PWR)	SFR (Monju)	SFR Feature
Coolant	Water	Sodium	Large thermal stress
RV exit temp.	320 deg-c	529 deg-c	Elevated Temp.
RV inlet and exit Temp. Difference	30 deg-c	132 deg-c	Large temp. difference and thermal stress
Coolant pressure	16 MPa	1 MPa	Low press.
RV innner Dia.	4 m	7 m	Large Dia.
RV wall thickness	200 mm	50 mm	Thin thickness
RV dia. and thickness ratio	20	140	Thin thickness and large dia.

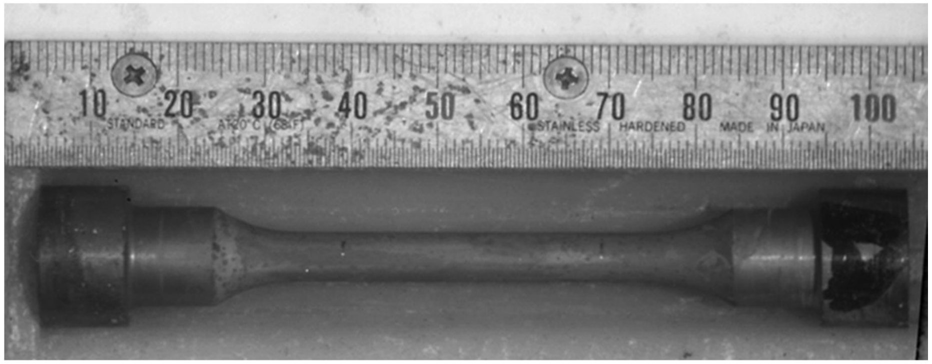
Aging phenomena identified in Sodium

Damage	Aging degradation	Consideration in design
Thickness reduction	Corrosion	Corrosion of steel in sodium at low oxygen concentration is very low and not considered as degradation to care. Significant corrosion of inner surface of sodium retaining pipe of Joyo piping operated over 60,000hours was not observed.
Crack	Fatigue and creep fatigue	Fatigue and creep fatigue are major degradation of SFR, because of SFR future ,i.e. elevated temperature and high temperature difference. In the design, cumulative creep and fatigue damage were evaluated and satisfy with criteria of JSME. On the other hand, cracks are reported many SFRs in the world at earlier stage, fatigue and creep fatigue are to be cared.
	SCC	No environmental factors for SCC in impurity controlled sodium exist and no SCC damage were reported so far. SCC is not degradation to care.
Material property degradation	Neutron irradiation degradation	Damage of reactor vessel and internal structure increase with neutron irradiation. In design, the degree of damage is evaluated using accelerated material irradiation test data by irradiation reactor. To confirm material property change under the actual irradiation condition, the test pieces are loaded. As the property change of SFR irradiation shows ductility reduction, elongation at fracture of unloaded test piece is measured.

Specimen Irradiation Positions

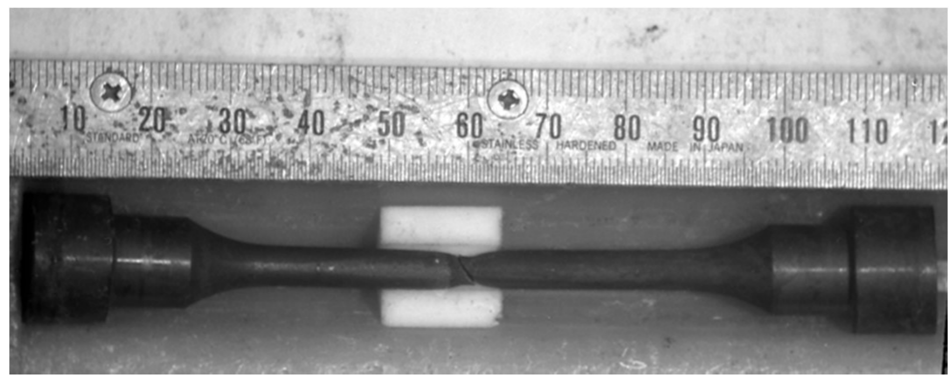


Post Irradiation Examination (Tensile Test)



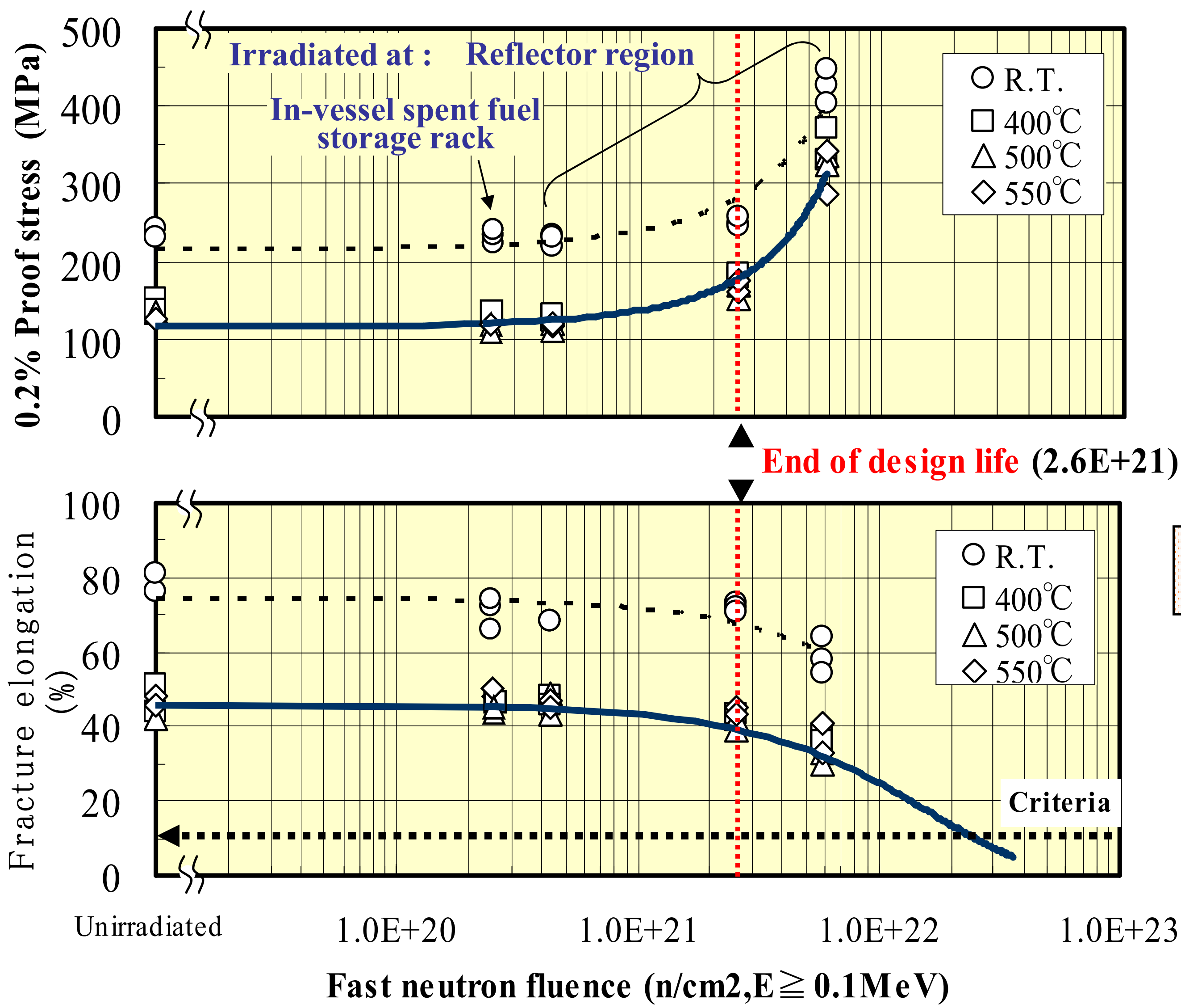
Before

After



Material ： 304SS (Reactor vessel material)
Irradiation subassembly No.： II-04 (Irradiated at in-vessel spent fuel storage rack)
Test temp. ： 400 deg-C

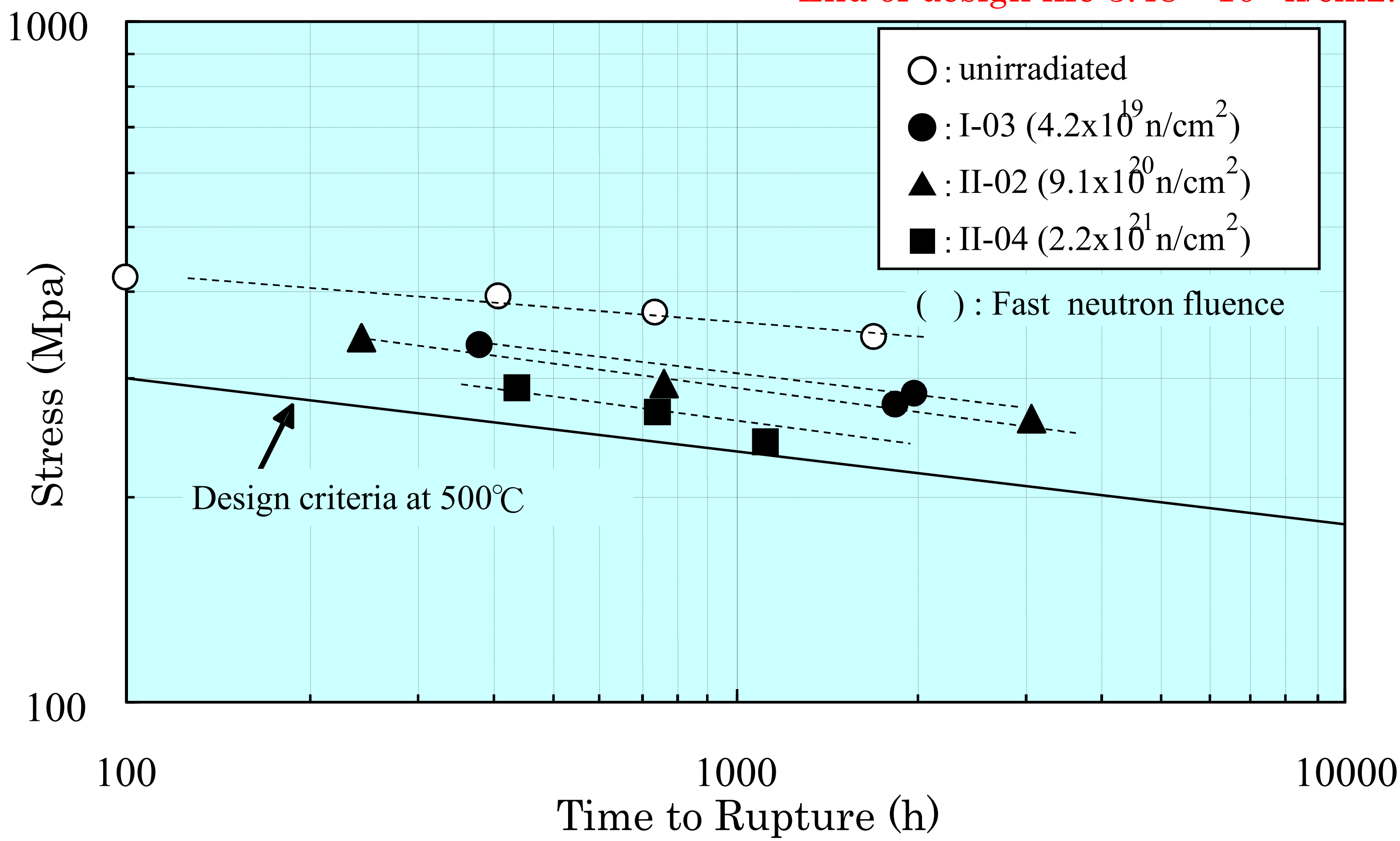
Radiation Deterioration: Neutron
- Core Support Plate (Base metal: 316SS) -



Integrity of core support plate material was confirmed

Integrity of Core Structure Materials (Irradiation effects) — Reactor Vessel Specimen (304SS) —

End of design life:3.48×10^{19}n/cm2.



Measured results satisfied the design criteria

Design criteria: JSME Codes for Nuclear Power Generation Facilities
— Rules on Design and Construction for Nuclear Power Plants Part 2: Fast Reactor

Erosion for the removed Secondary Main pipe (by MK-III modification)

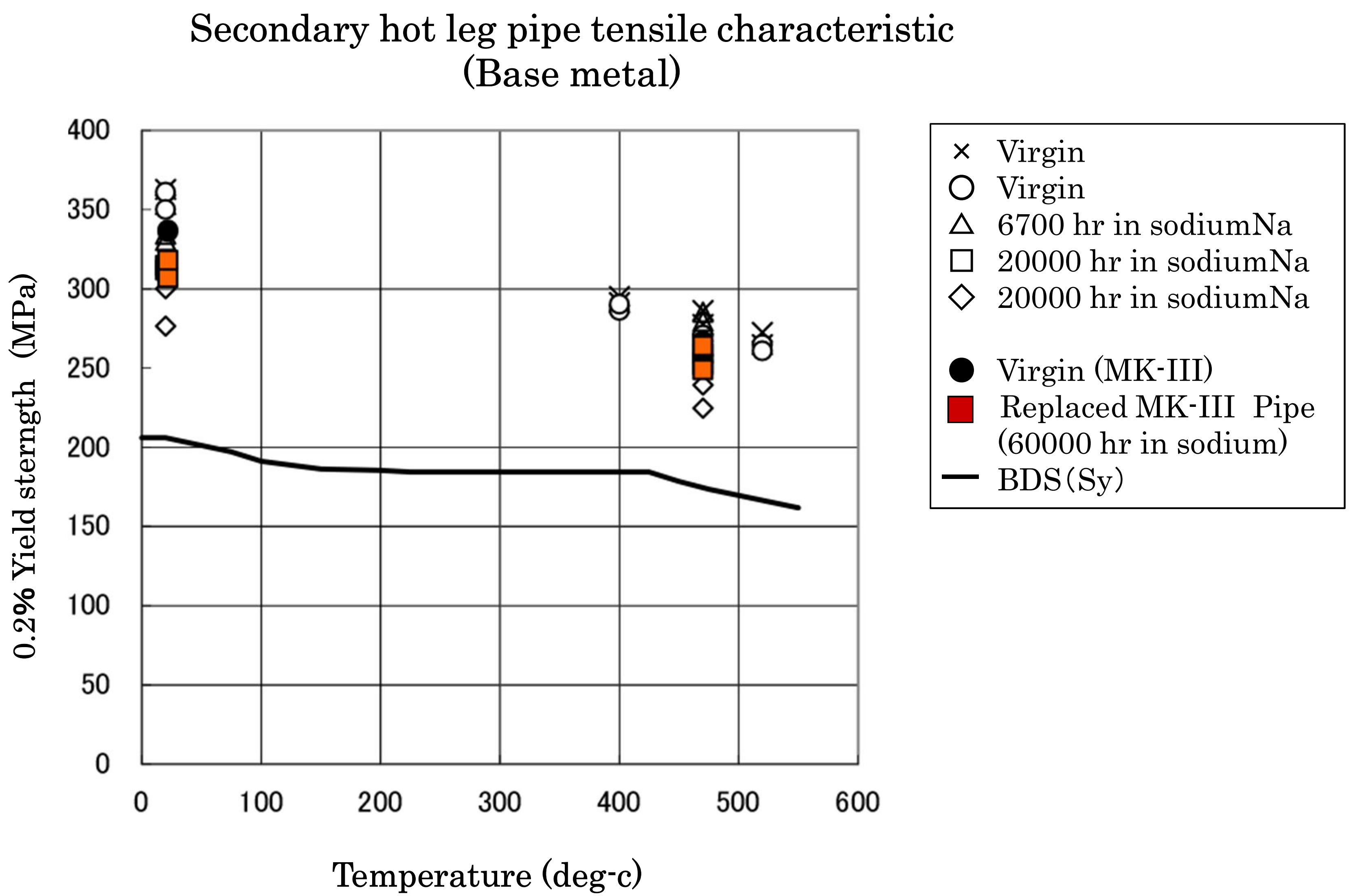
Results:

- ✓ No significant reduction in the wall thickness of straight pipe and elbow part of the secondary main cooling system.
- ✓ No corrosion/erosion in the inner surface.

IHX outlet elbow inner and outer surface

Material:2 1/4Cr-1Mo steel

Yield strength of removed secondary pipe)



Sodium Leak Continuous Monitoring

Concept of the LBB (Leak before Break)

Leak Before Break (LBB) is satisfied for SFRs.

✔Ductile Structural Material : Austenitic SS
✔Low Pressure Boundary : Sodium Coolant

↓

Continuous Monitoring is adopted as a main measure of the In-Service Inspection.

✔Sodium Leak Monitoring (SoLM)
✔Ar gas Leak Monitoring (ArLM)

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

Sodium Leak Detection Systems (example)

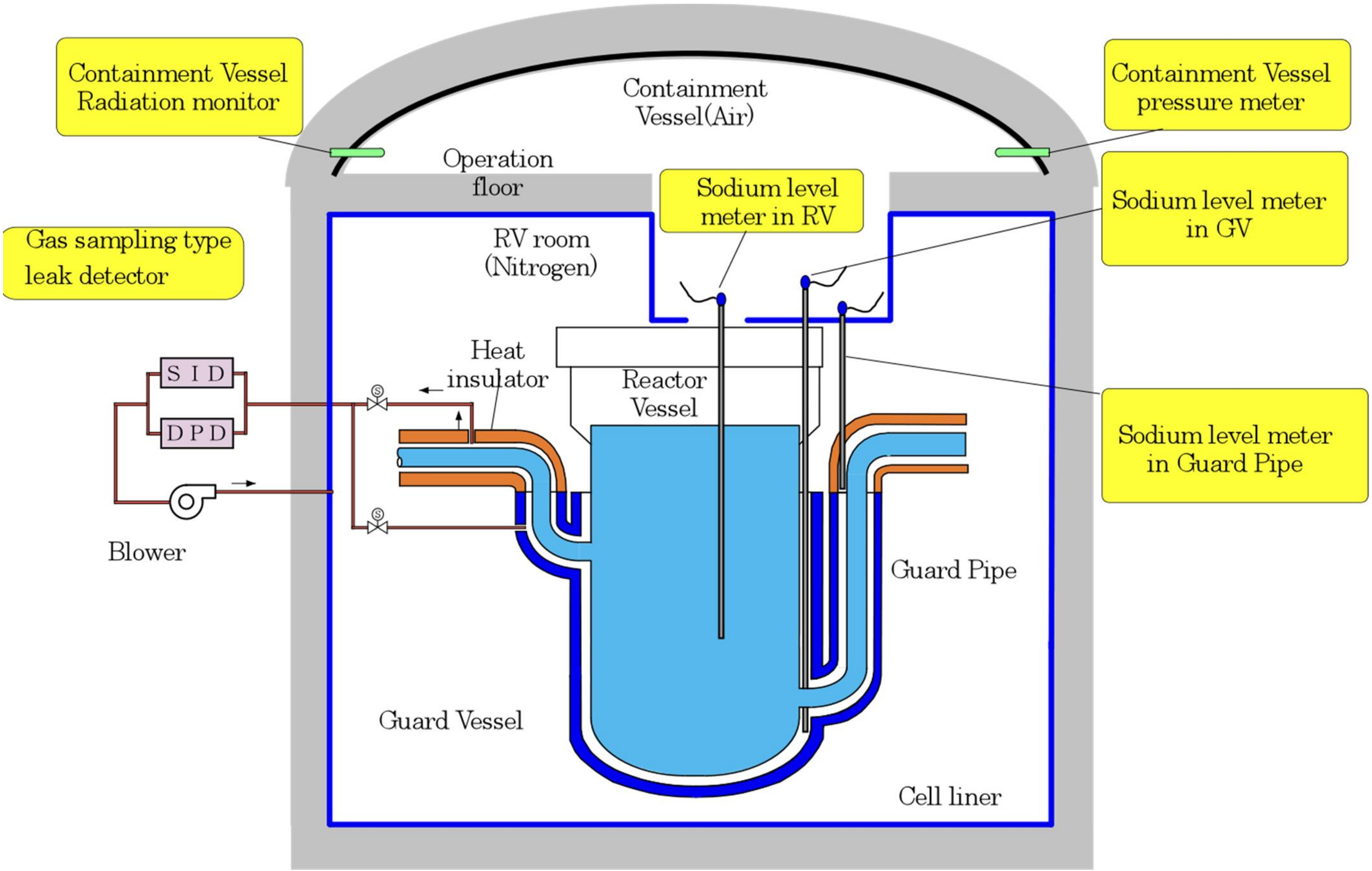
The Various kinds of sodium leak detectors in Monju are classified as shown in the table.

Object	Detector Type	Leak Scale	Reactor Operation
Primary System	SID (Sodium Ionization Detector)	Very Small Scale*1	Manually Scram
	DPD (Differential Pressure Detector)		
	CLD (Contact Type Sodium Leak Detector)		
	Process Data Changing (Process instrumentation : O/T Sodium Level)	Middle Scale*2 to Large Scale*3	
	Process Data Changing (Safety protection system: R/V Sodium Level or G/V Sodium Level or C/V Under Floor Temp. or Pressure or Radiation Dosage of C/V Above Floor)		Automatically Scram
Secondary System	RID (radioactive Ionization Detector), CLD	Very Small Scale to Small Scale	Manually Scram
	Smoke type Detector (Fire Alarm System)		
	Process Data Changing (Sodium Level or Temp.)	Large Scale	Automatically Scram

*1: Sodium leak rate >1 X10^{-3} kg/h.
*2: Sodium leak rate 1 to 50 X10^{3} kg/h.
*3: Sodium leak rate >50 X10^{3} kg/h.

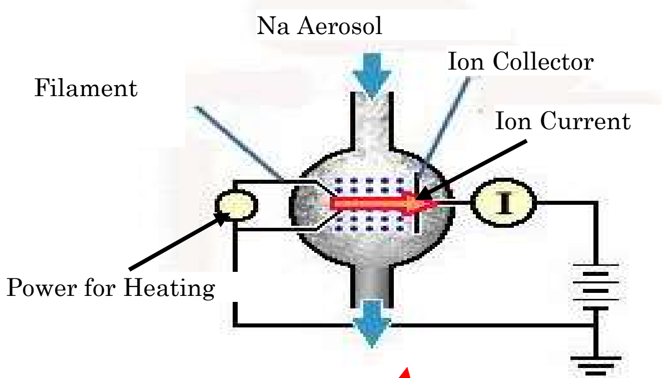
R/V: Reactor Vessel O/T: Overflow Tank
G/V: Guard Vessel C/V: Containment Vessel

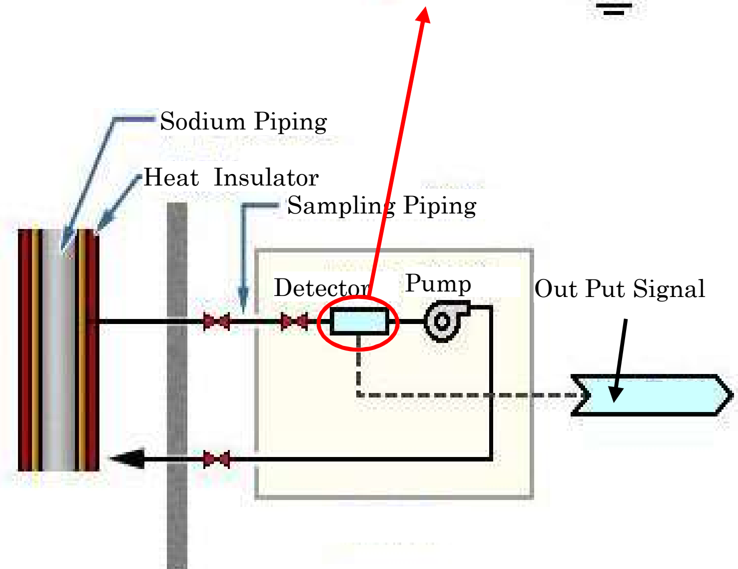
Installation of Sodium Leak Detectors (R/V)



Gas Sampling Type Sodium Leak Detectors (Example)

➢SID (Sodium Ionization Detector)





<Gas Sampling Method>

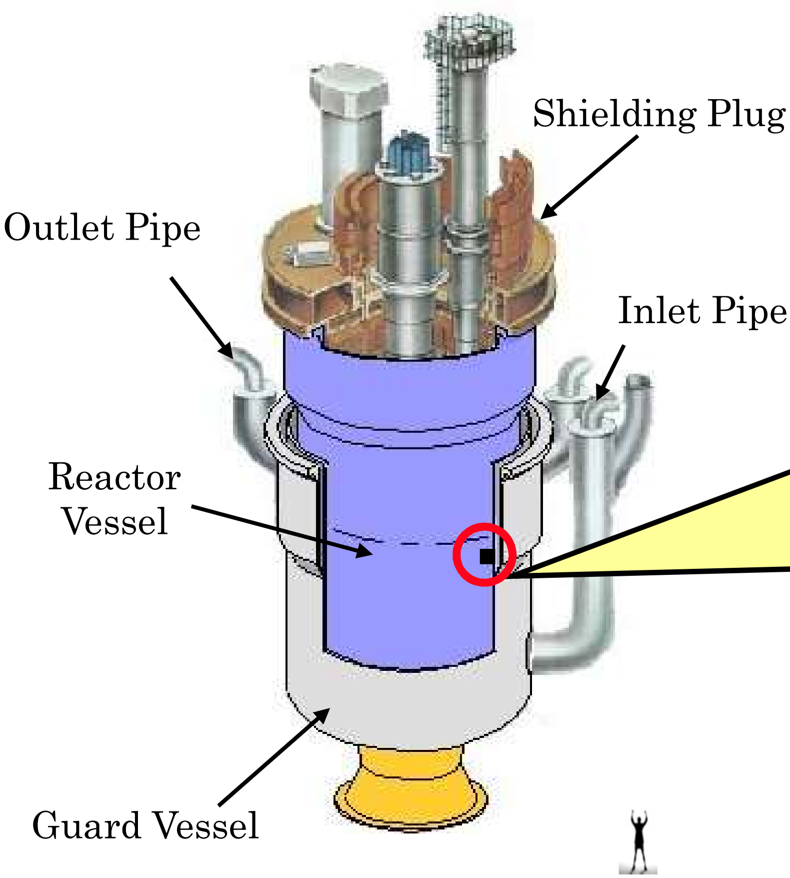
◆SID is best suitable to detect sodium leak in the inert gas atmosphere (detector would burn if used in air).
(Sensitivity: $\geq 1 \times 10^{-10}$ g Na/cc (minimum))

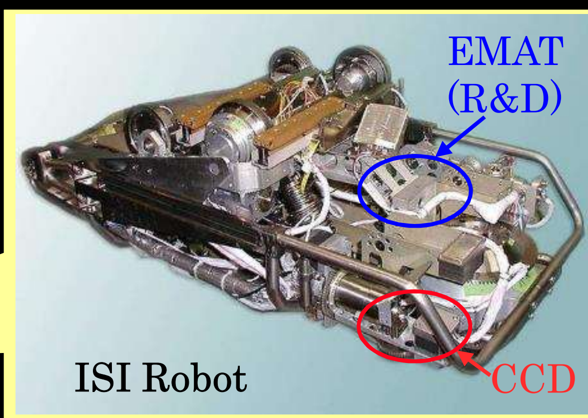
Source: M. Sawada & G. Rodoriguez, pocketbook on Sodium Technology by JAEA & CEA, P36, July 2006

ISI Technology Development

ISI device for Reactor Vessel

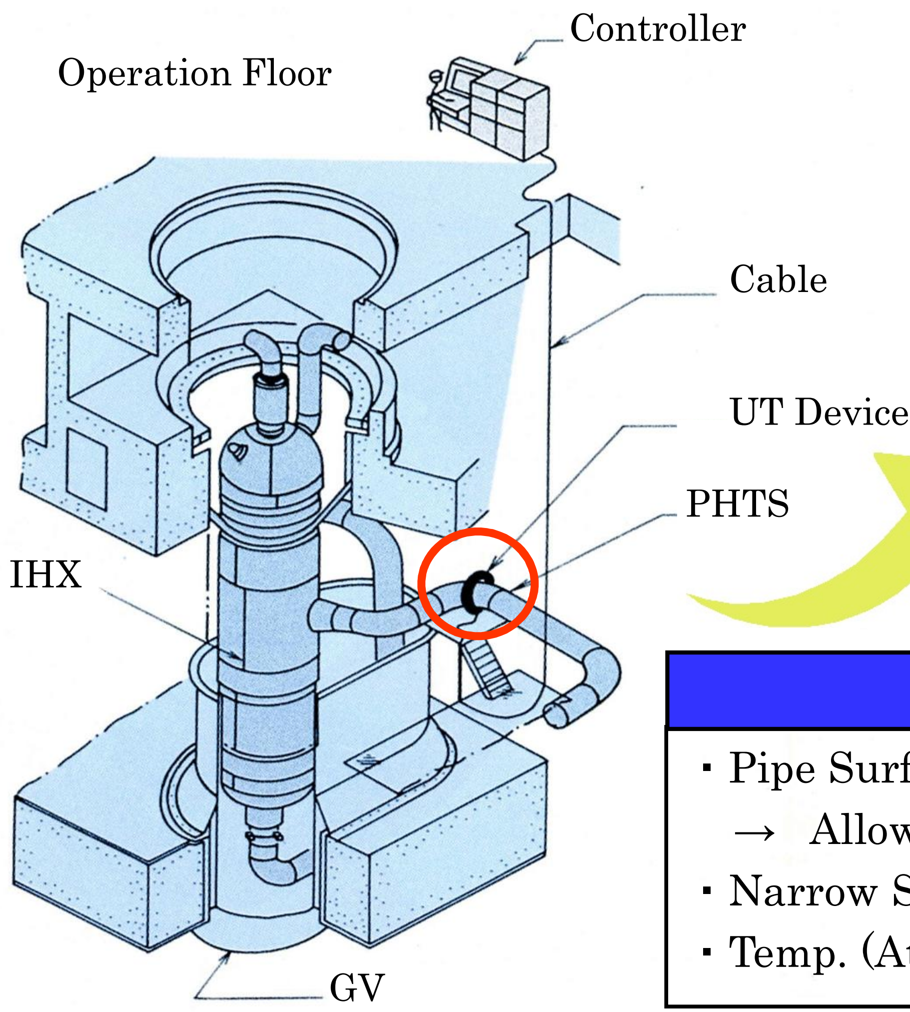
ISI Region	Circumferential Welds of Reactor Vessel Weld Joints of Inlet Nozzles and Vessel Weld Joints of Inlet Pipes and Nozzles
Methods	Vit : Fiber Scope → CCD Camera Vot : UT by EMAT(Electro Magnetic Acoustic Transducer) (R&D)





Inspection Conditions
・Temperature: 200 deg-c
・Radio Activity: 10 Sv/hr
・Atmosphere: Nitrogen Gas

ISI device for Primary Pipes



Testing Condition

- Pipe Surface (Dose Rate 15mSv/h)
  → Allowable Working : 5min
- Narrow Space with Obstacles
- Temp. (Atmosphere 55deg-c, Piping Surface 80deg-c)

Codes and standards

Surveillance

JSME Codes for Nuclear Power Generation Facilities
- Rules on Design and Construction for Nuclear Power Plants Part I: Light Water Reactors JSME S NC1 – 2008/2009
Article 12 Surveillance Testing

Monitoring and inspection

JSME Codes for Nuclear Power Generation Facilities
-Rules on Fitness-for-Service for Fast Reactor Nuclear Power Plants
Under deliberation for class1 component in JSME

Class 1 components examination methods

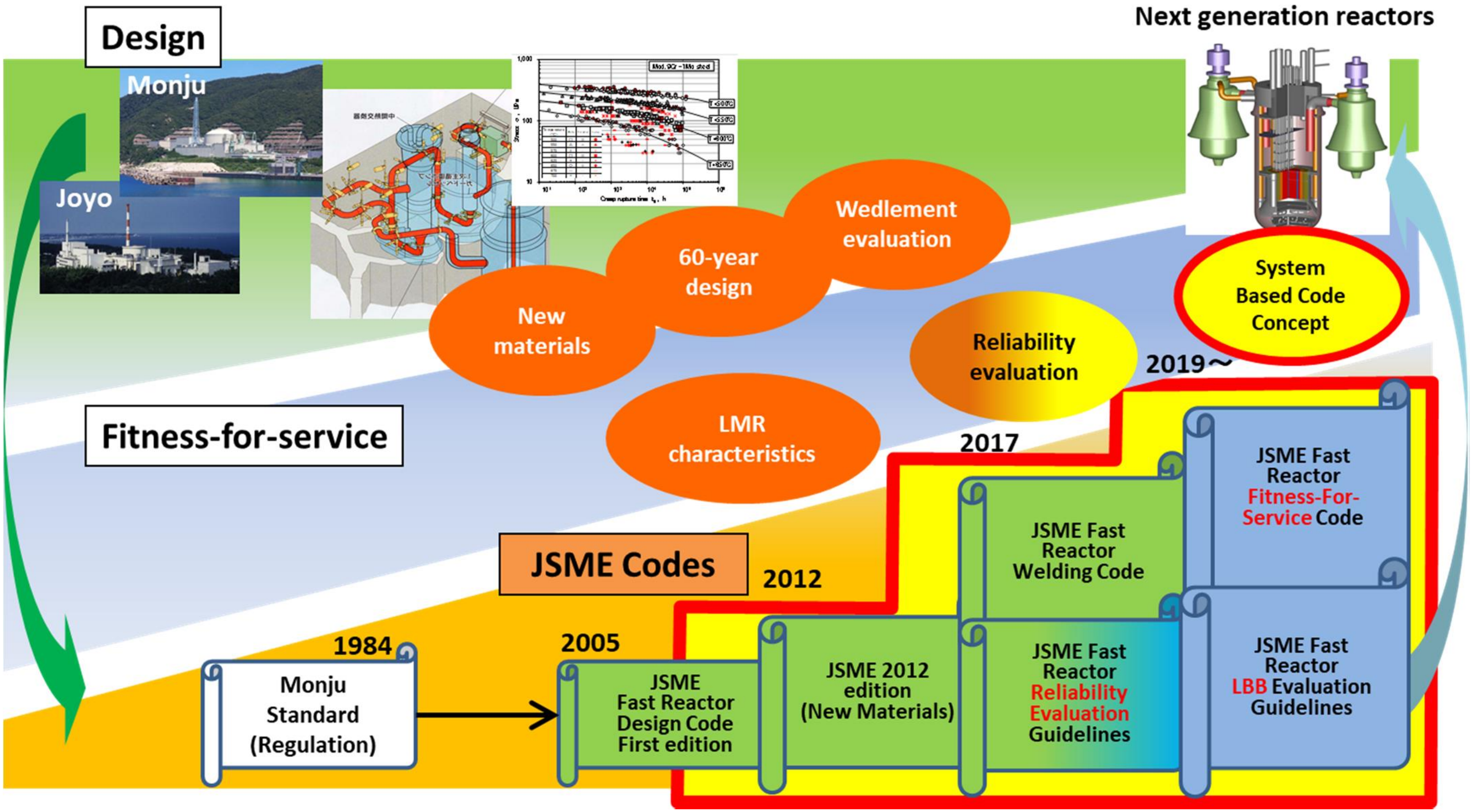
Parts Examined		Rules on Fitness-for-Service for Fast Reactor	LWR
Primary coolant boundary welds	Sodium retaining parts	Continuous monitoring 1 (CM－1) Small diameter pipe: Continuous monitoring 2 (CM－2)	Volumetric examination, surface examination Small diameter pipe: System leak test and VT-2
	Cover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

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

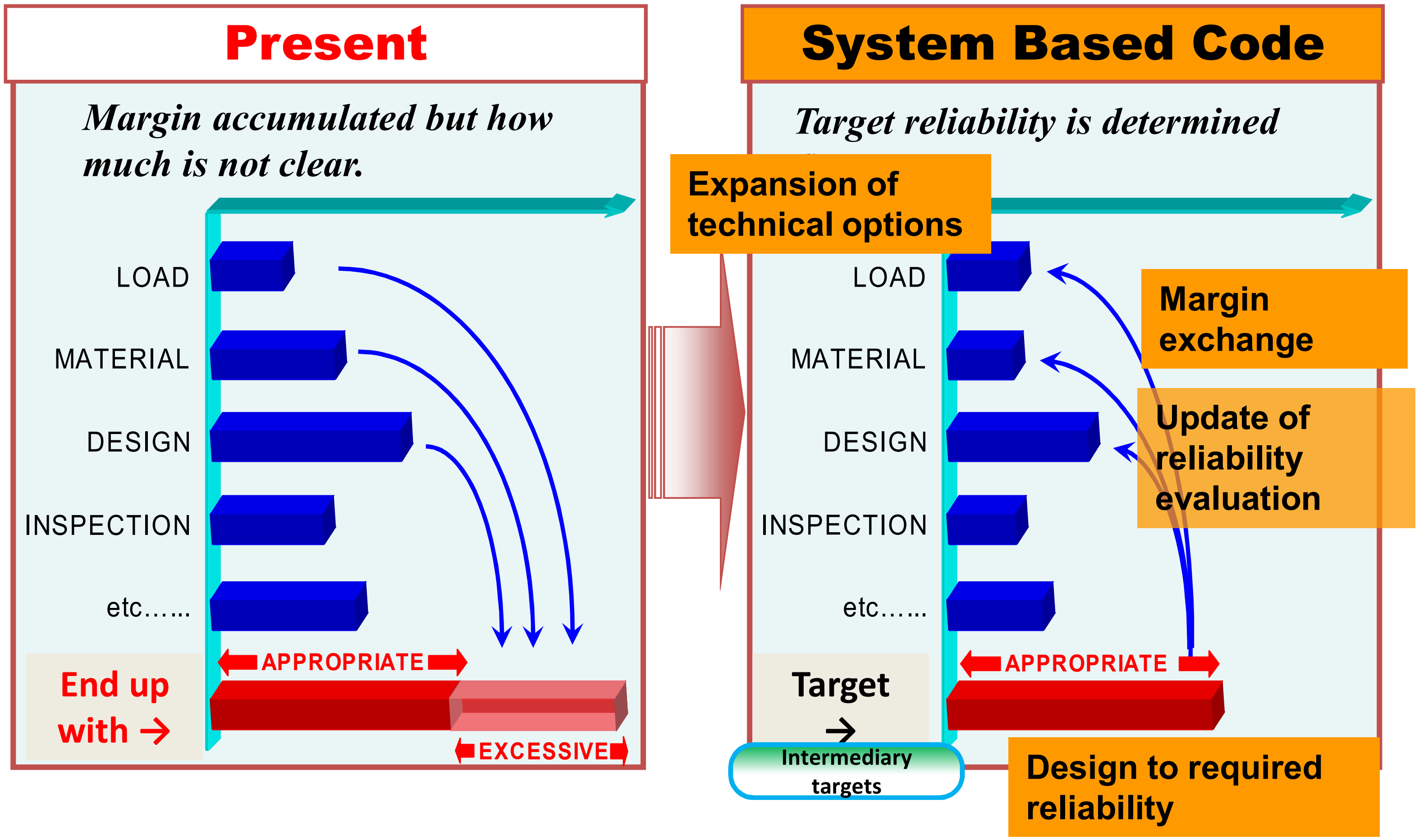
Influence of damage

- Leak detection sensitivity is decided by LBB evaluation.
- JSME code “Rules on Leak Before Break Assessment for Sodium Cooled Fast Reactor Plants” that is under deliberation, may be applied.
- As for Monju, primary main piping satisfies LBB assessment by using sodium leak detector that can detect leak rate of 1kg/h sodium leak.

JSME Fast Reactor Codes



System Based Code Concept



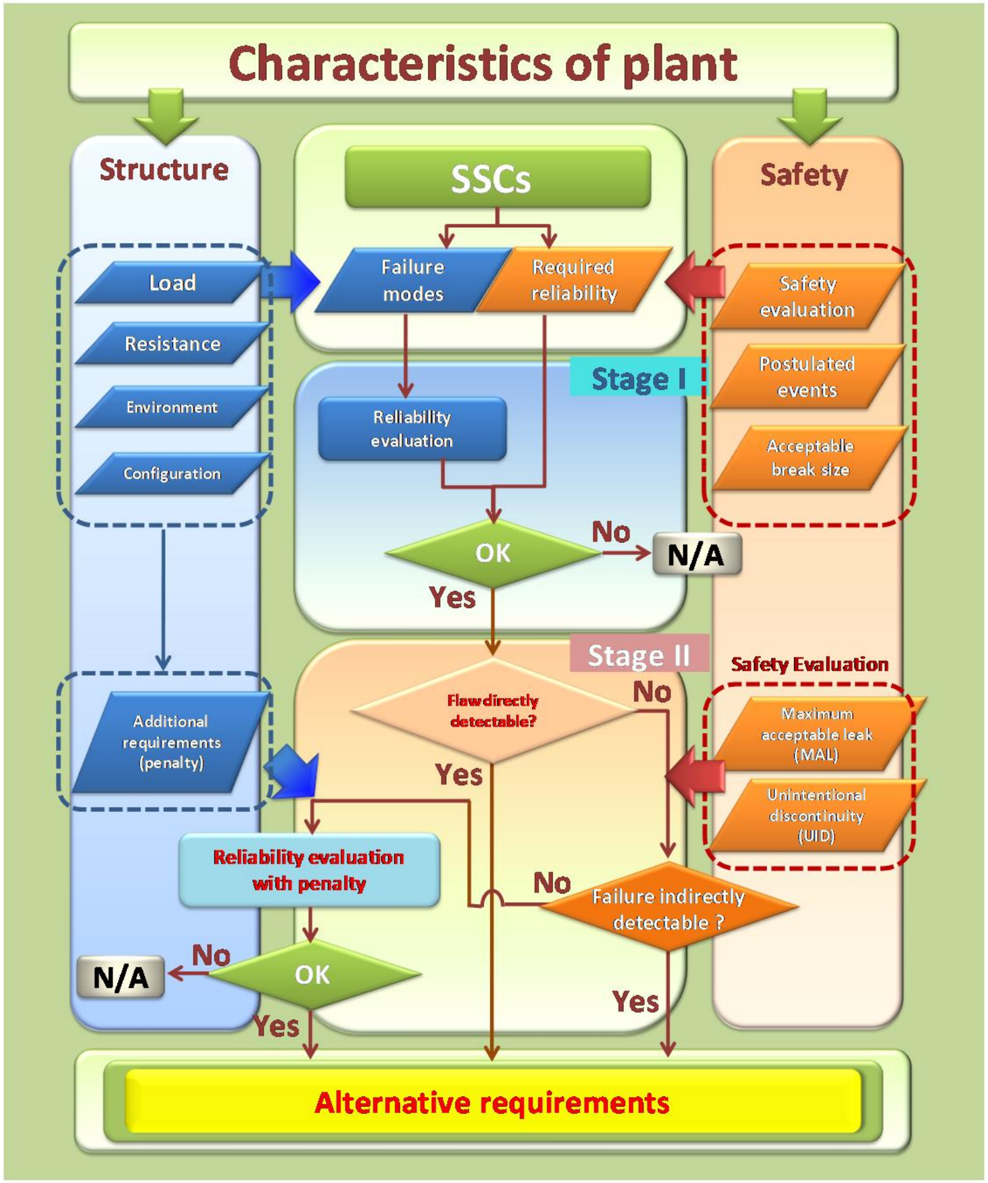
Asada, Y., Japanese Activities Concerning Nuclear Codes and Standards – Part II, Journal of Pressure Vessel Technology, ASME 128 (2006) 64.

ASME Code Case N-875

- The SBC process consists of Stage I and II evaluations that have different objectives.

  (a) Stage I is a structural reliability evaluation which considers the component level structural integrity and the probability of failure of the component at design basis conditions The contribution of inservice inspections is not taken into account at this stage.

  (b) Stage II is a safety-related evaluation of detectability of a flaw which ensures that the plant can be safely shut down before the flaw reaches the maximum acceptable size. The evaluation is performed taking into account the component safety functions during plant operation and the events that have been postulated in the safety evaluation of the plant. Any flaw can be detected either directly or indirectly. Indirect detection includes the detection of a leak or an unintentional discontinuity of plant parameters such as temperature and leak rate. If a flaw is not detectable, then additional margins in structural integrity will be required



Conclusions

- The experience of the experimental fast reactor Joyo and the prototype reactor Monju provides valuable information for the development of next generation liquid metal reactors.
  - The Joyo operation and maintenance (O&M) experiences over 30 years and integrity confirmation results of PSR will be reflected to future SFR not only O&M but also design and construction.
  - The R&D and design for Monju, the SSC developed for Monju such as leak detectors, ISI device are taken in codes and standards.
- Codes and standards development is ongoing within JSME utilizing the experience. Collaboration with ASME is concurrently underway.