

■ 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





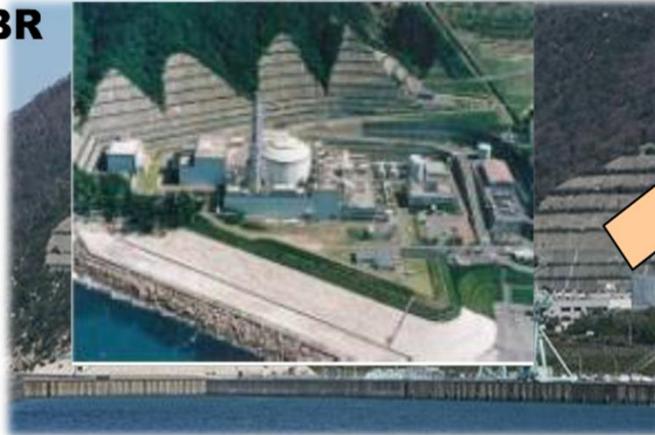
# Joyo and Monju toward Commercial Reactors

**To confirm the performance of sodium cooled FBR power plant**

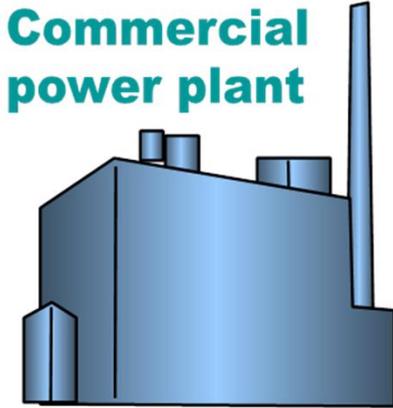
**Experimental fast reactor JOYO  
140MWt**



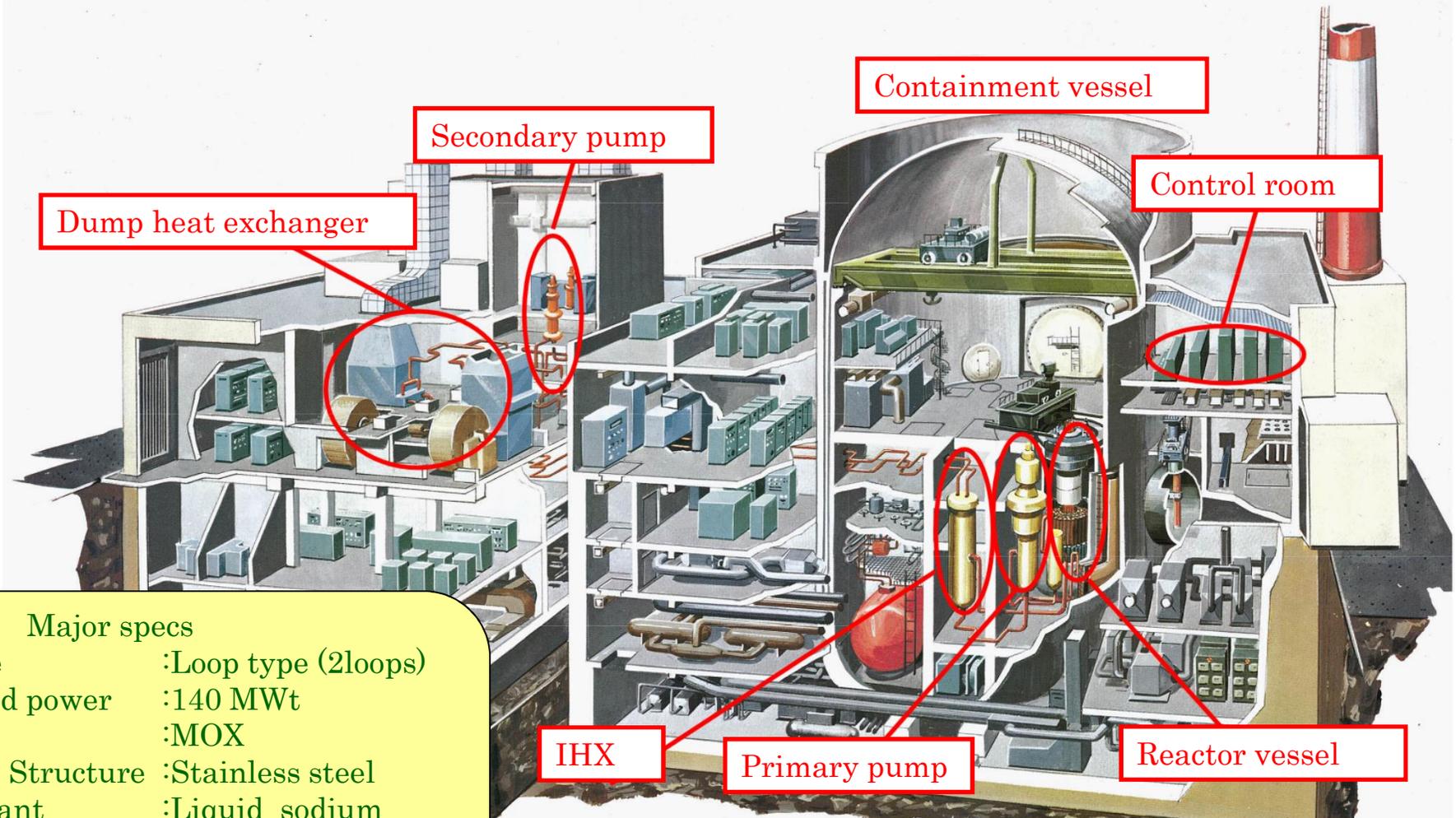
**Prototype fast breeder reactor power plant MONJU  
280MWe (714MWt)**



**Commercial power plant**



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



**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

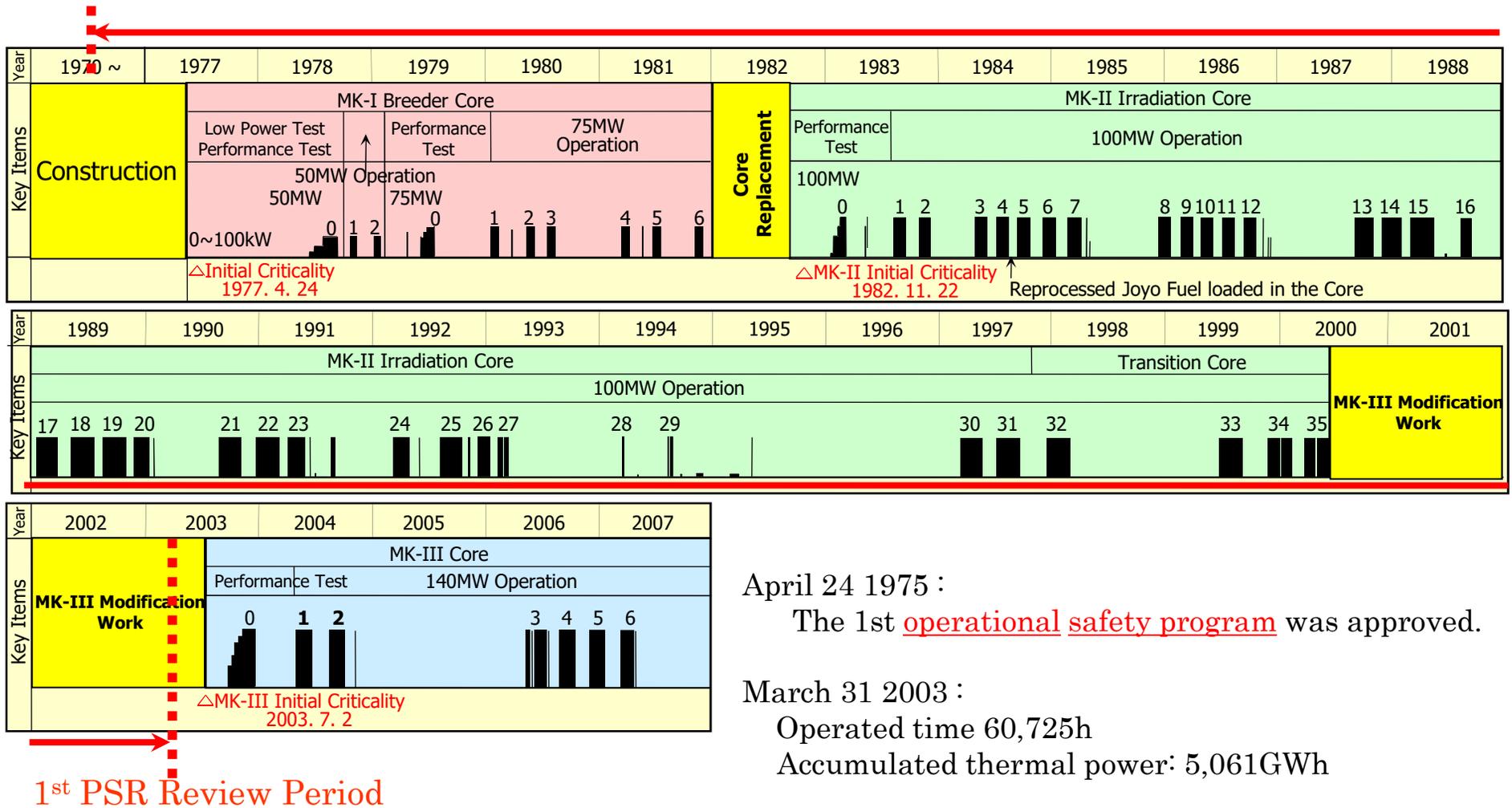
**MK-II**

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**

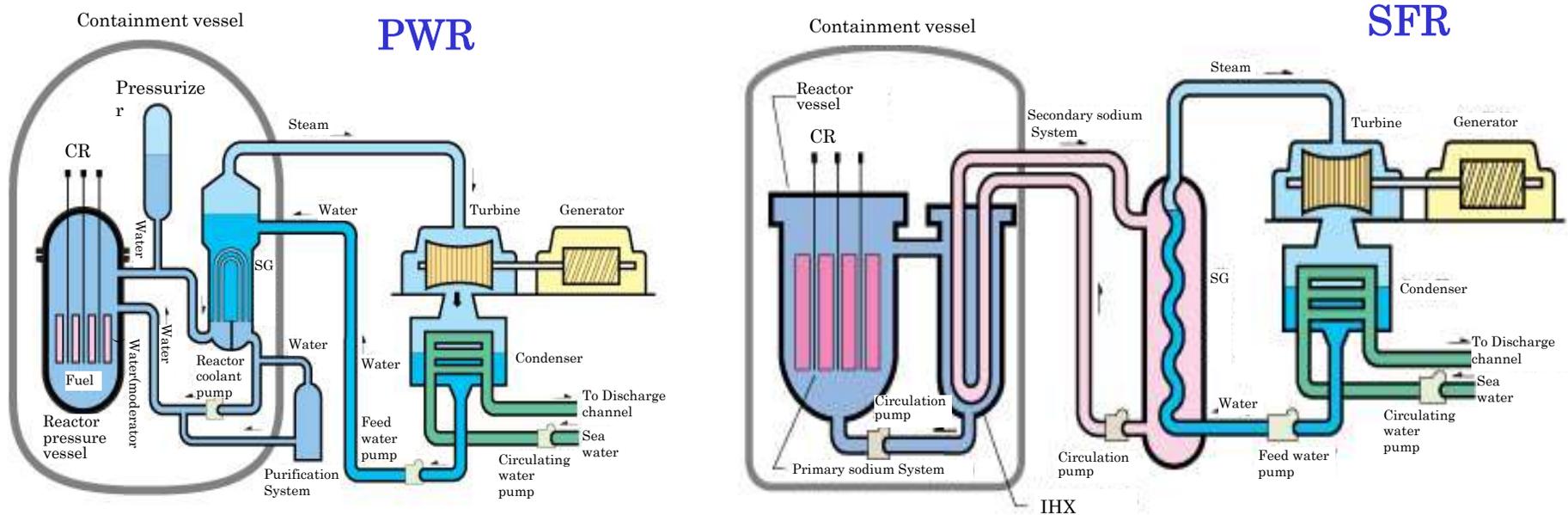
1981  
• Natural Circulation Test  
• Confirm Breeding Ratio  
• Accumulate Technical Experience Through Planning,  
Construction and Operation  
1977.4 Attain First Criticality

# 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



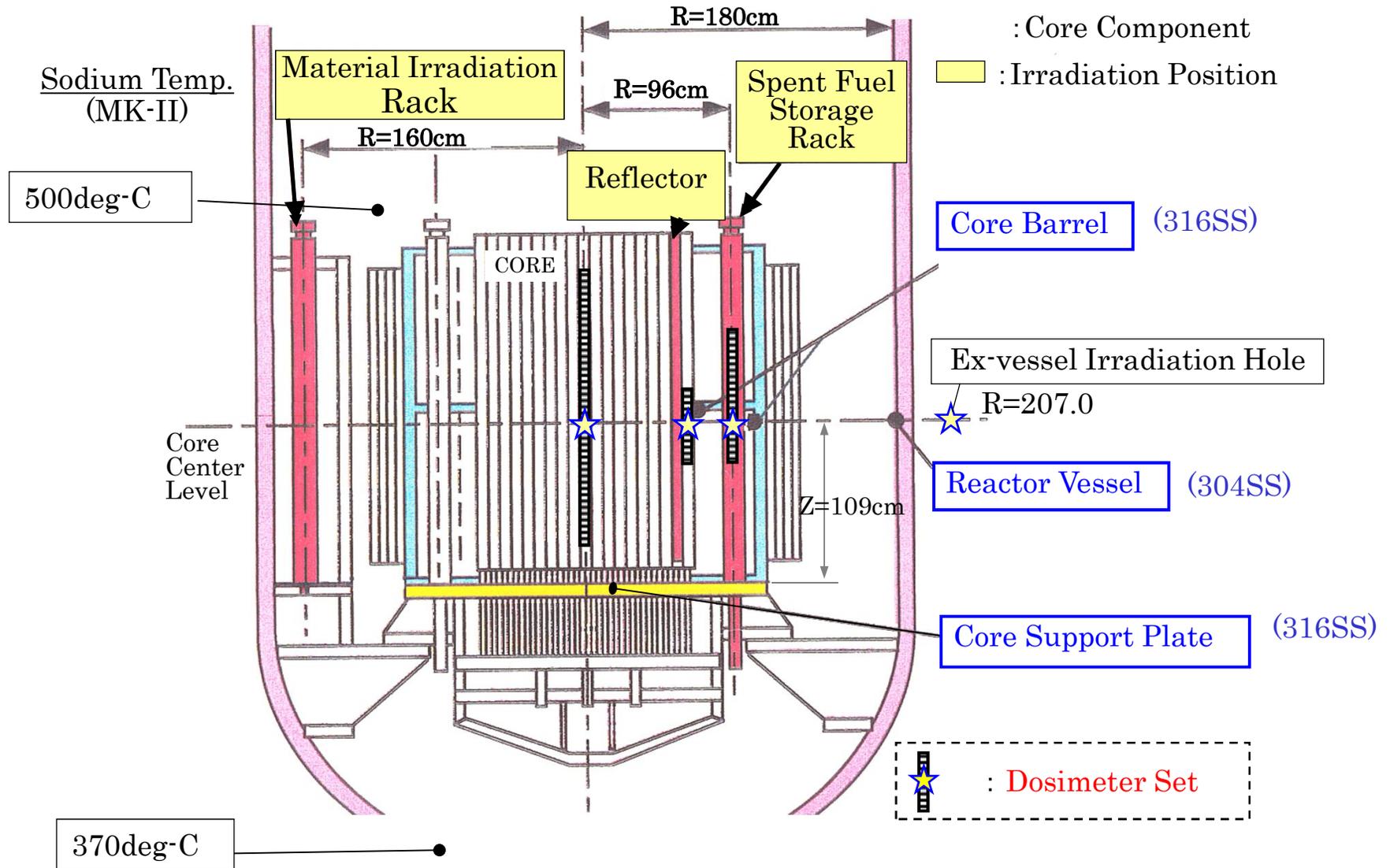
Feature of SFR comparison with LWR

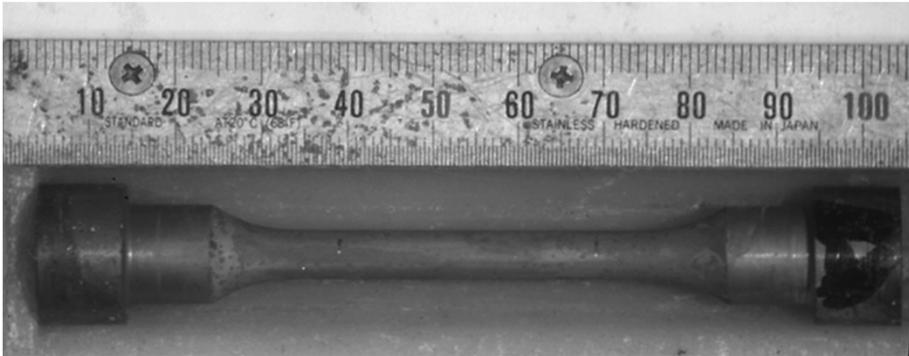
| Subject                            | LWR (PWR) | SFR (Monju) | SFR Feature                               |
|------------------------------------|-----------|-------------|---|
| Coolant                            | Water     | Sodium      | Large thermal stress                      |
| RV exit temp.                      | 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 inner 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.<br>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.<br>In the design, cumulative creep and fatigue damage were evaluated and satisfy with criteria of JSME.<br>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.<br>SCC is not degradation to care.   |
| Material property degradation | Neutron irradiation degradation | Damage of reactor vessel and internal structure increase with neutron irradiation.<br>In design, the degree of damage is evaluated using accelerated material irradiation test data by irradiation reactor.<br>To confirm material property change under the actual irradiation condition, the test pieces are loaded.<br>As the property change of SFR irradiation shows ductility reduction, elongation at fracture of unloaded test piece is measured. |

# Specimen Irradiation Positions

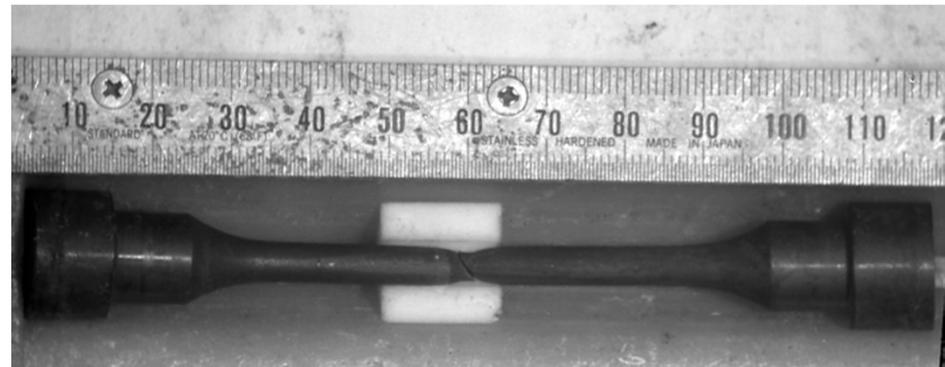




Before



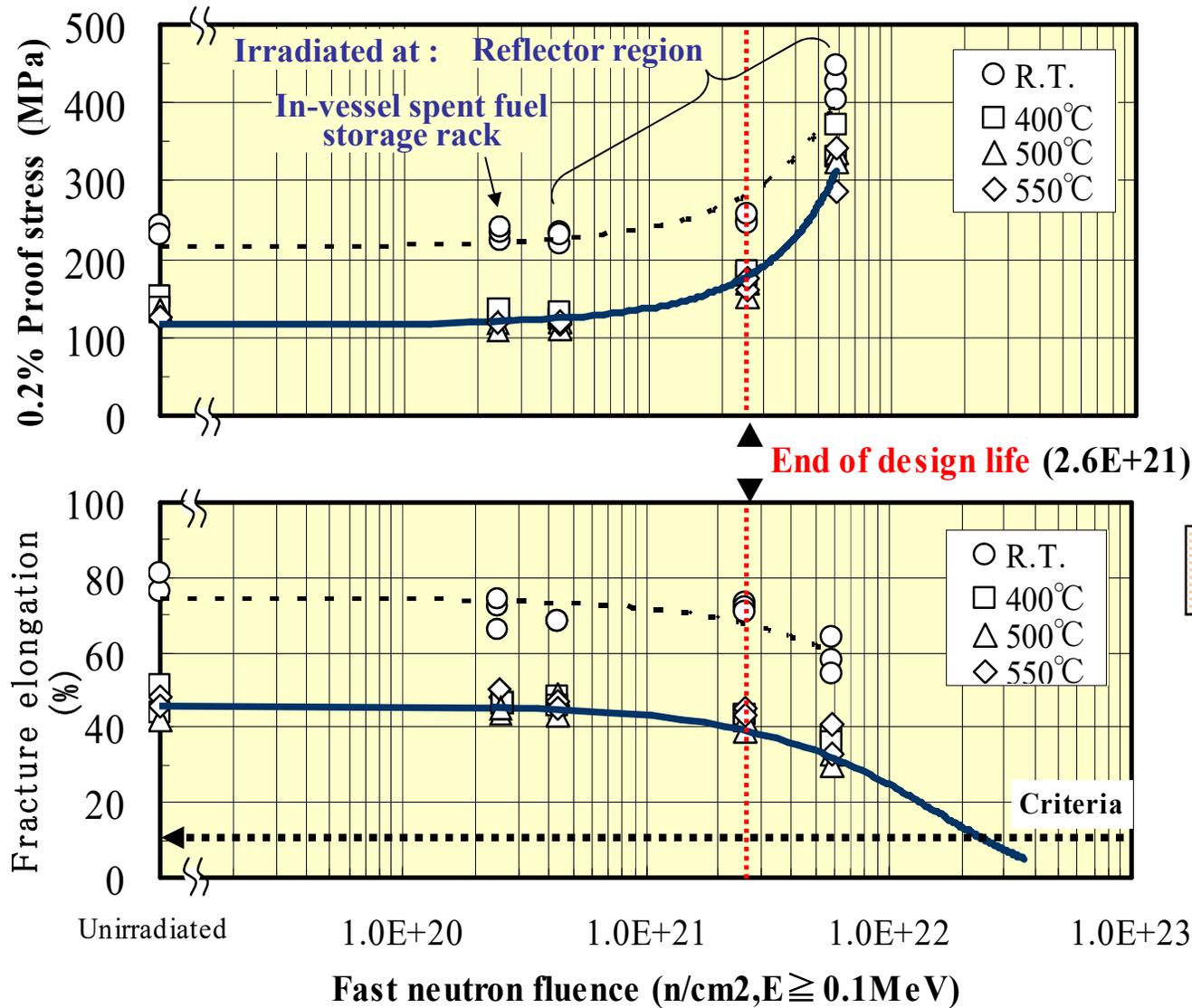
After



Material : 304SS (Reactor vessel material)

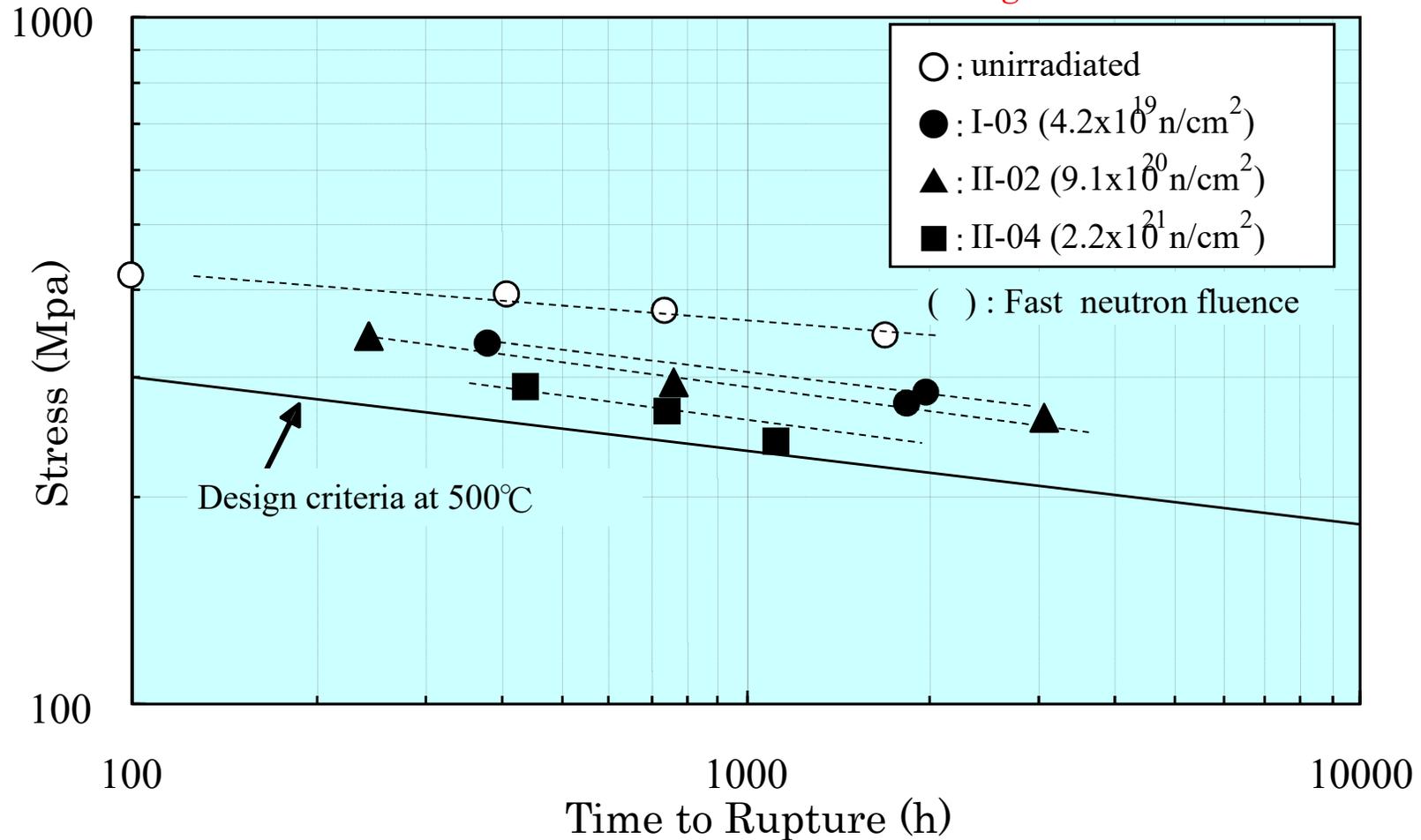
Irradiation subassembly No. : II-04 (Irradiated at in-vessel spent fuel storage rack)

Test temp. : 400 deg-C



Integrity of core support plate material was confirmed

End of design life:  $3.48 \times 10^{19} \text{ n/cm}^2$ .



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

## 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.

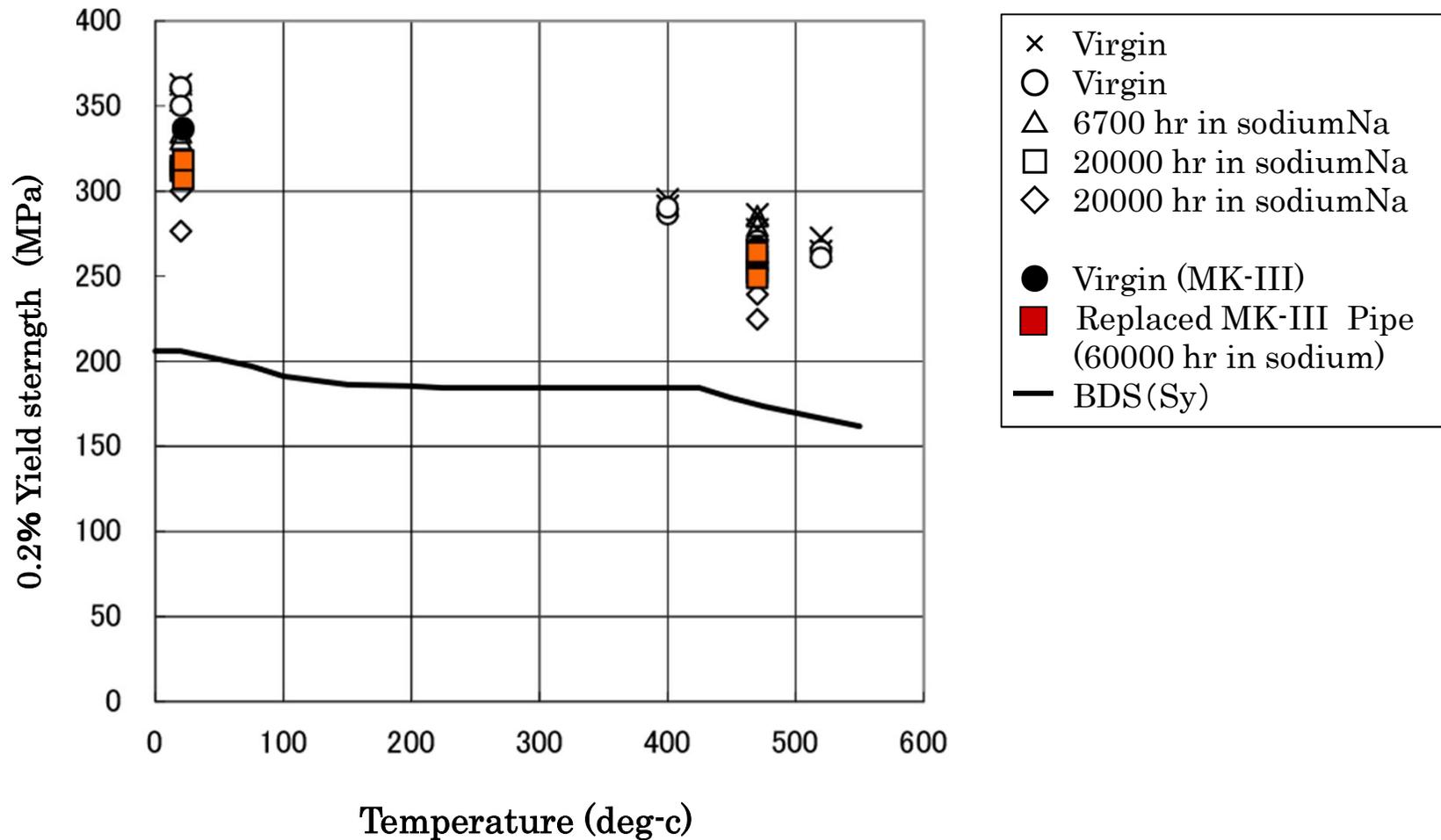


IHX outlet elbow inner and outer surface

Material: 2 1/4Cr-1Mo steel

# Yield strength of removed secondary pipe)

Secondary hot leg pipe tensile characteristic  
(Base metal)



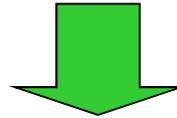
# Sodium Leak Continuous Monitoring

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## 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* <sup>1</sup>                                | Manually Scram      |
|                  | DPD (Differential Pressure Detector)  |   |                     |
|                  | CLD (Contact Type Sodium Leak Detector)   |   |                     |
|                  | Process Data Changing<br>(Process instrumentation : O/T Sodium Level)   | Middle Scale* <sup>2</sup><br>to<br>Large Scale* <sup>3</sup> | Automatically Scram |
|                  | Process Data Changing<br>(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) |   |                     |
| Secondary System | RID (radioactive Ionization Detector), CLD  | Very Small Scale<br>to<br>Small Scale                         | Manually Scram      |
|                  | Smoke type Detector<br>(Fire Alarm System)  |   |                     |
|                  | Process Data Changing<br>(Sodium Level or Temp.)  | Large Scale   | Automatically Scram |

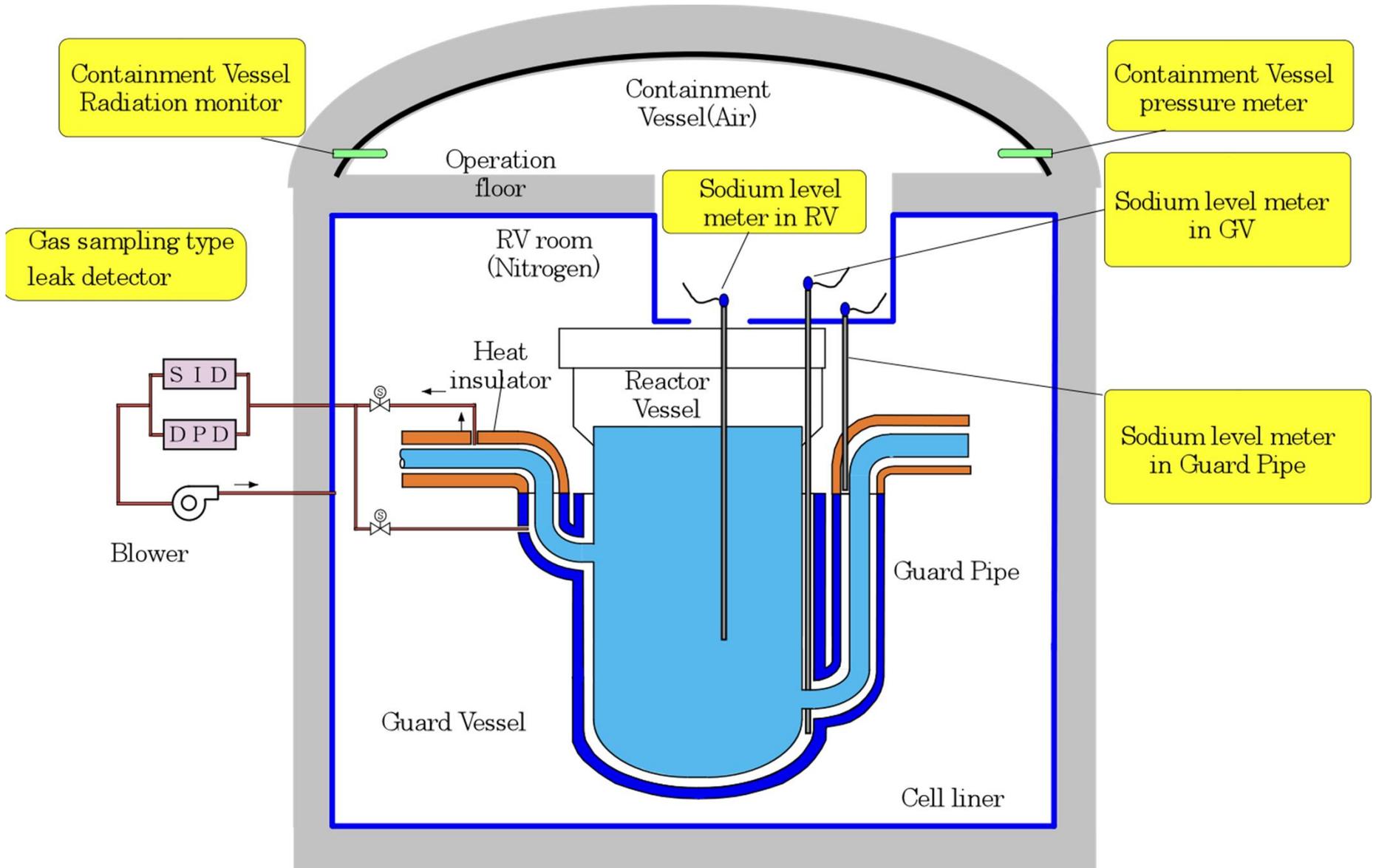
\*1: Sodium leak rate  $>1 \times 10^{-3}$  kg/h.

\*2: Sodium leak rate 1 to  $50 \times 10^3$  kg/h.

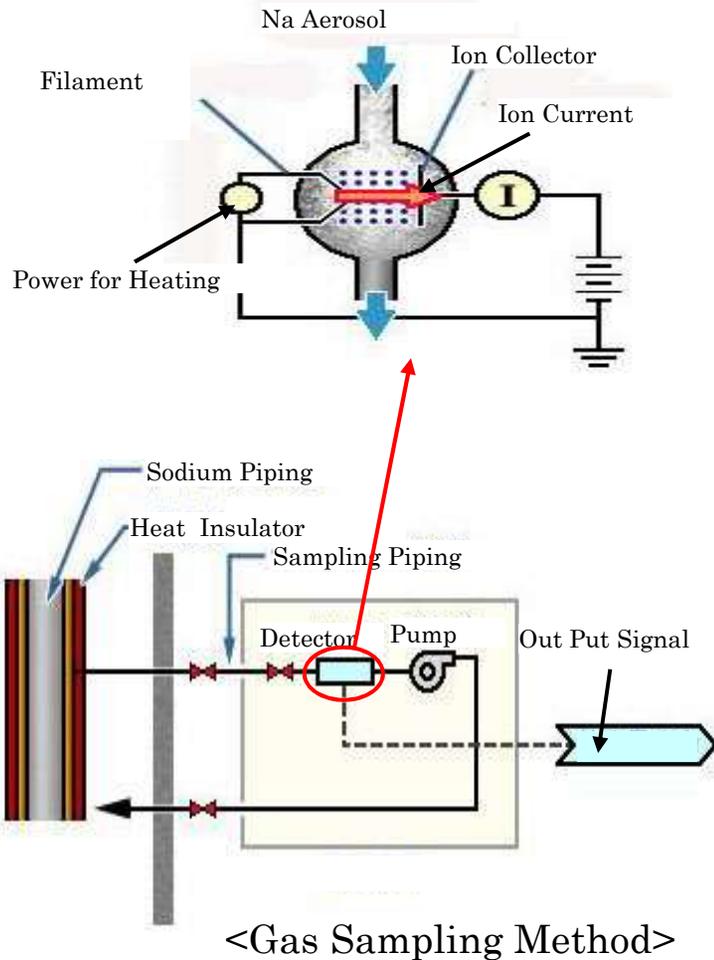
\*3: Sodium leak rate  $>50 \times 10^3$  kg/h.

R/V: Reactor Vessel    O/T: Overflow Tank

G/V: Guard Vessel    C/V: Containment Vessel



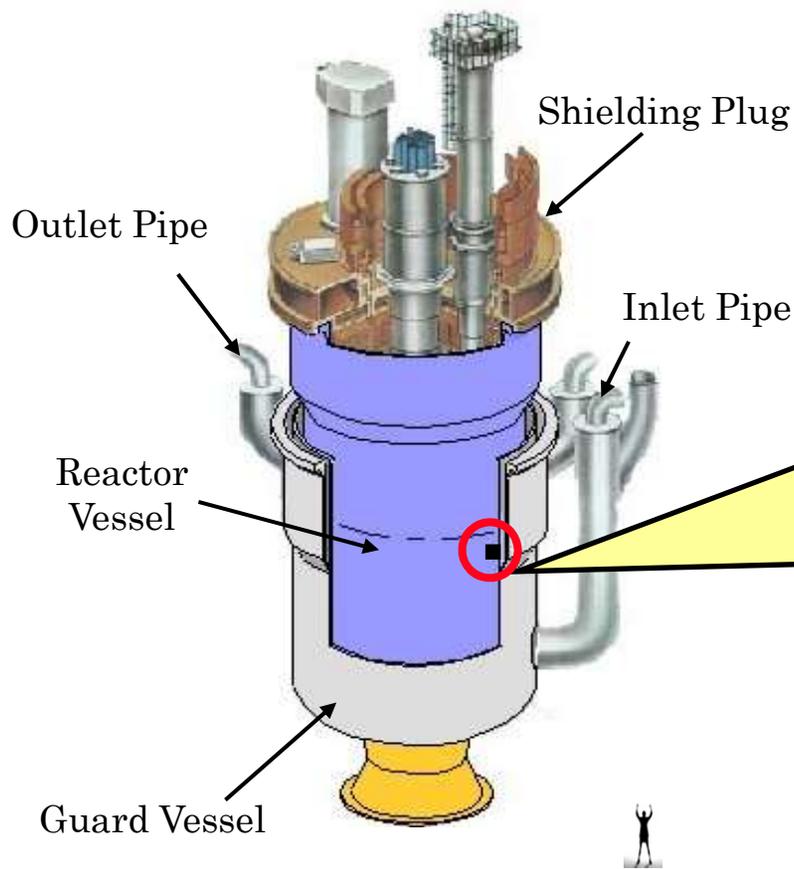
## ➤ SID (Sodium Ionization Detector)



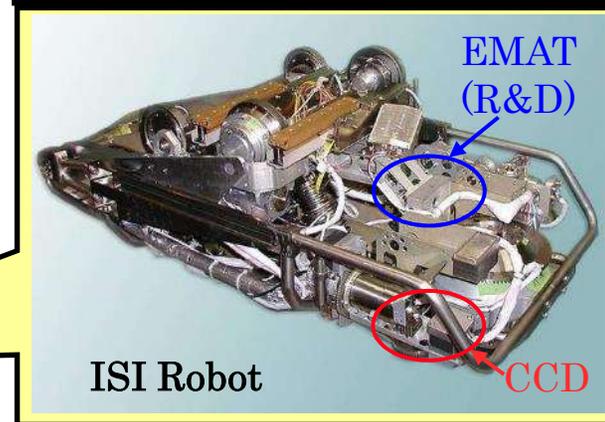
◆ **SID** is best suitable to detect sodium leak in the inert gas atmosphere (detector would burn if used in air).  
 (Sensitivity:  $\geq 1 \times 10^{-10}$  g Na/cc (minimum))

# ISI Technology Development

## ISI device for Reactor Vessel

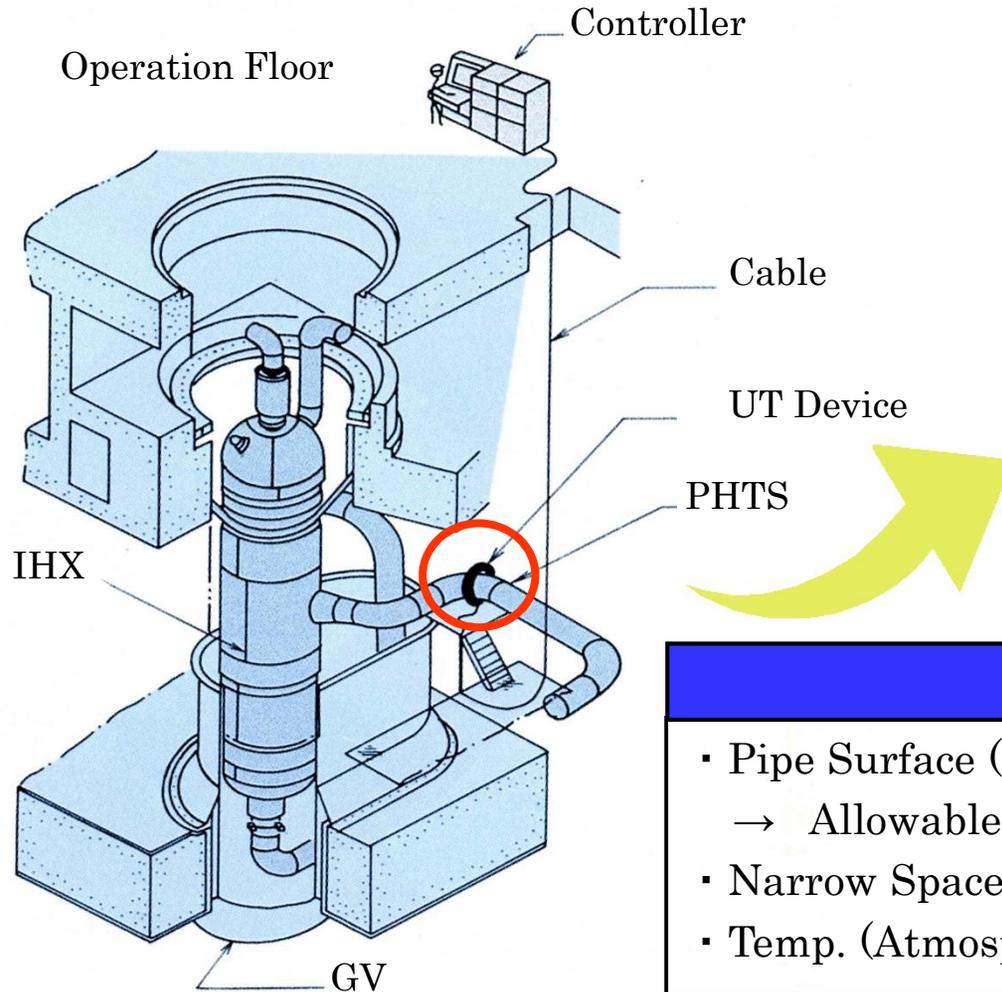


|            |  |
|------------|--|
| ISI Region | Circumferential Welds of Reactor Vessel<br>Weld Joints of Inlet Nozzles and Vessel<br>Weld Joints of Inlet Pipes and Nozzles |
| Methods    | Vit: Fiber Scope → CCD Camera<br>Vot: UT by EMAT(Electro Magnetic Acoustic Transducer) (R&D)               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



| Inspection Conditions  |
|--|
| <ul style="list-style-type: none"> <li>▪ Temperature: 200 deg-c</li> <li>▪ Radio Activity: 10 Sv/hr</li> <li>▪ Atmosphere: Nitrogen Gas</li> </ul> |

# 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)

## Surveillance

JSME Codes for Nuclear Power Generation Facilities

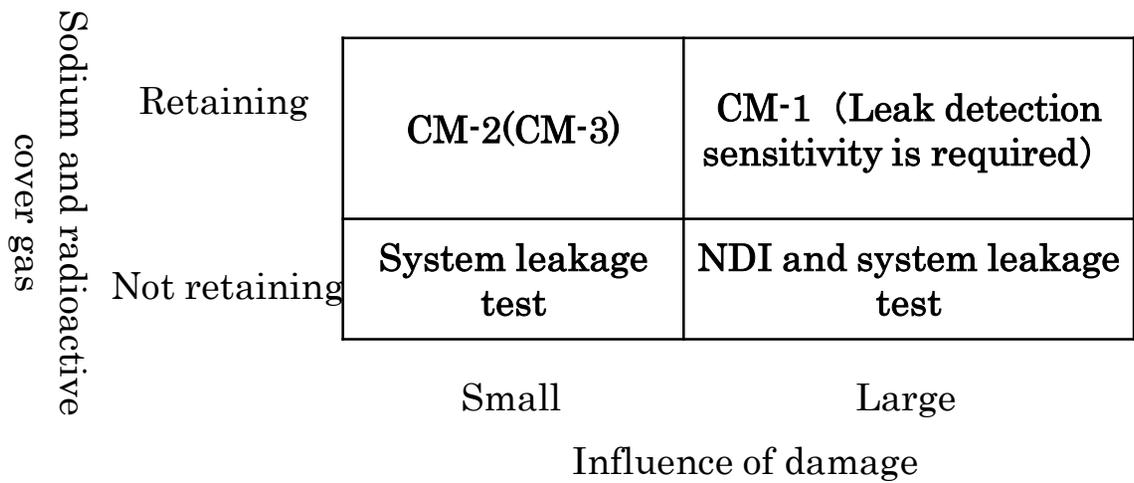
- Rules on Design and Construction for Nuclear Power Plants Part  
I: Light Water Reactors JSME S NC1 – 2008/2009  
Article 12 Surveillance Testing

## Monitoring and inspection

JSME Codes for Nuclear Power Generation Facilities

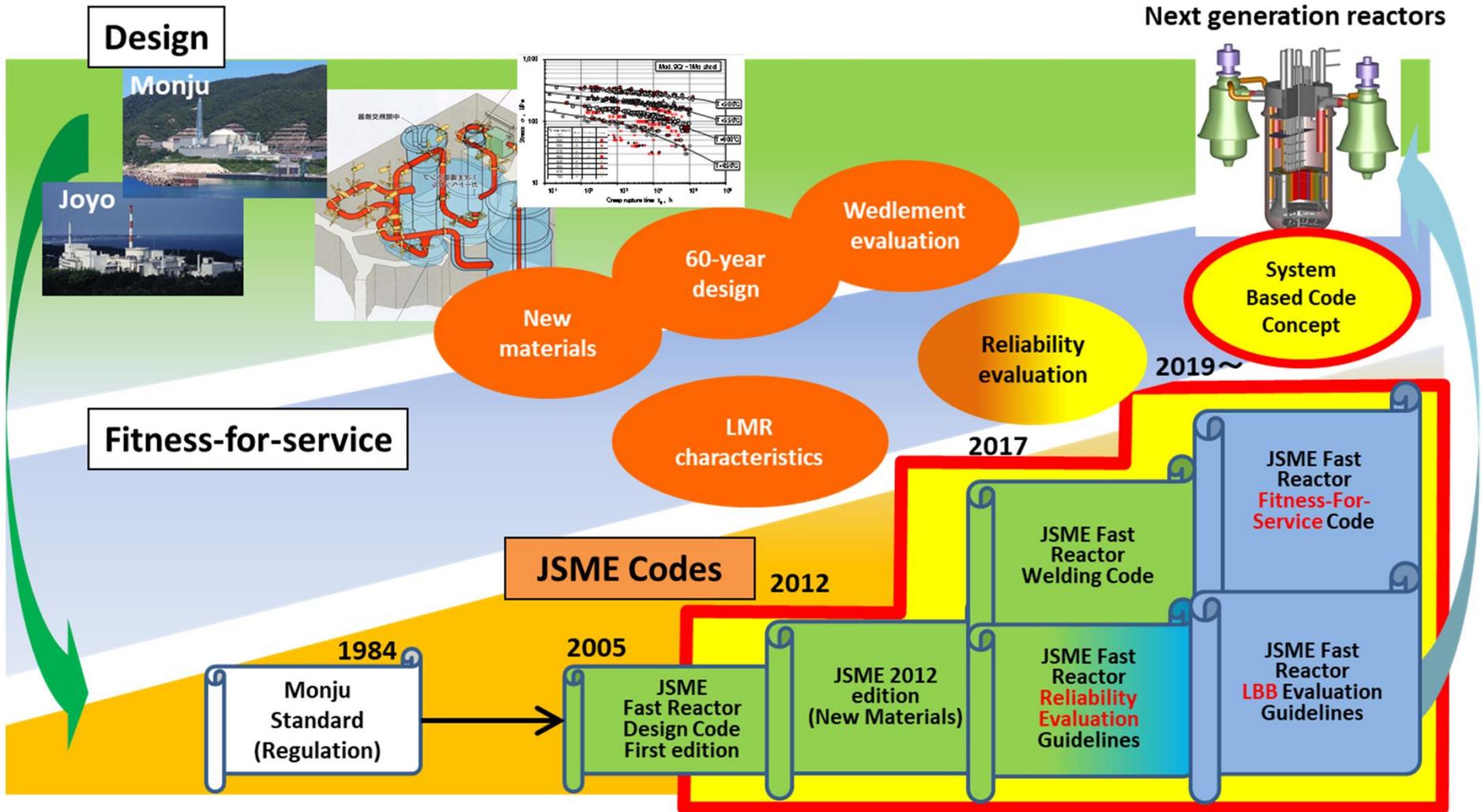
- Rules on Fitness-for-Service for Fast Reactor Nuclear Power Plants  
Under deliberation for class1 component in JSME

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

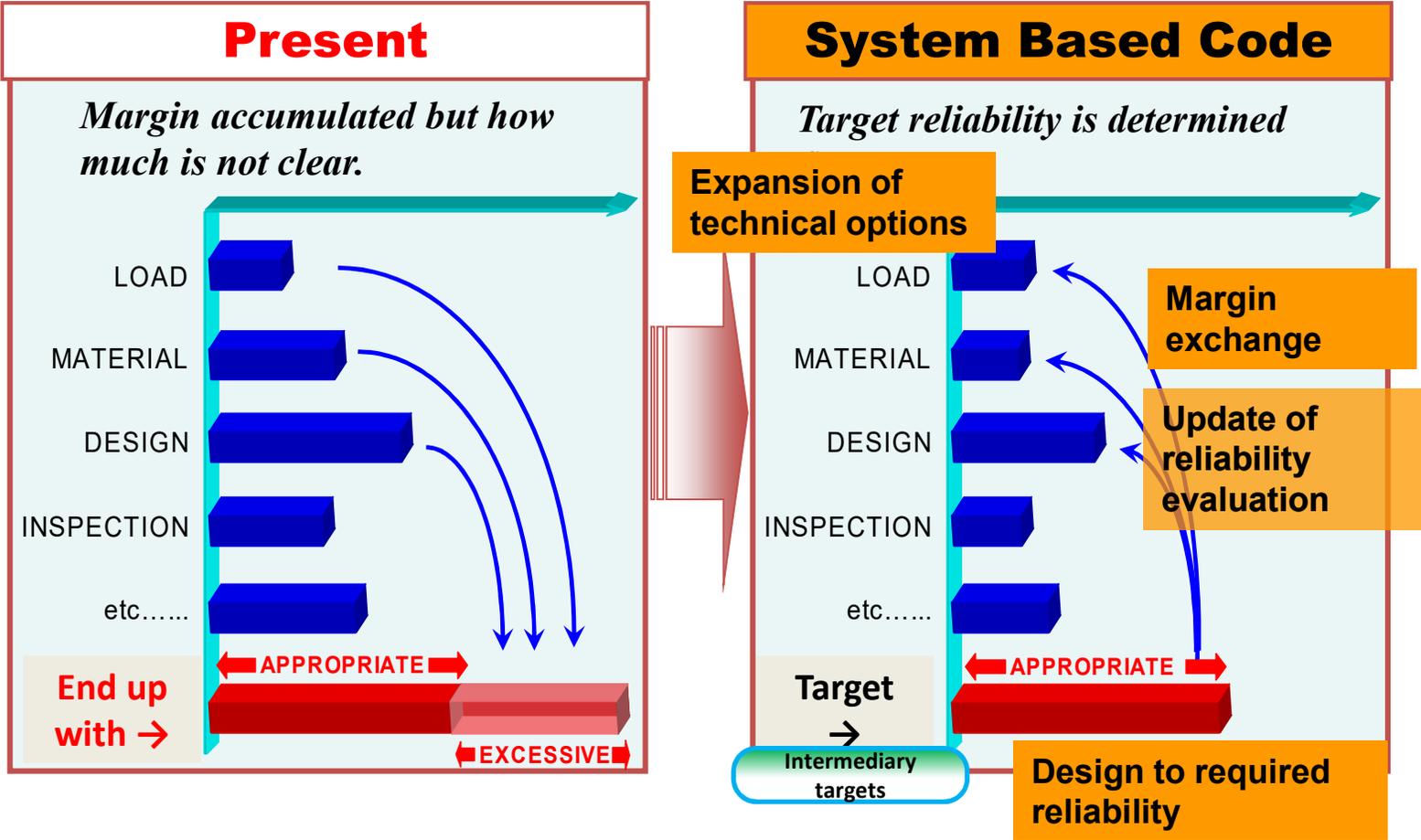


- Leak detection sensitivity is decided by LBB evaluation.
- JSME code “Rules on Leak Before Break Assessment for Sodium Cooled Fast Reactor Plants” that is under deliberation, may be applied.
- As for Monju, primary main piping satisfies LBB assessment by using sodium leak detector that can detect leak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

# JSME Fast Reactor Codes



# System Based Code Concept

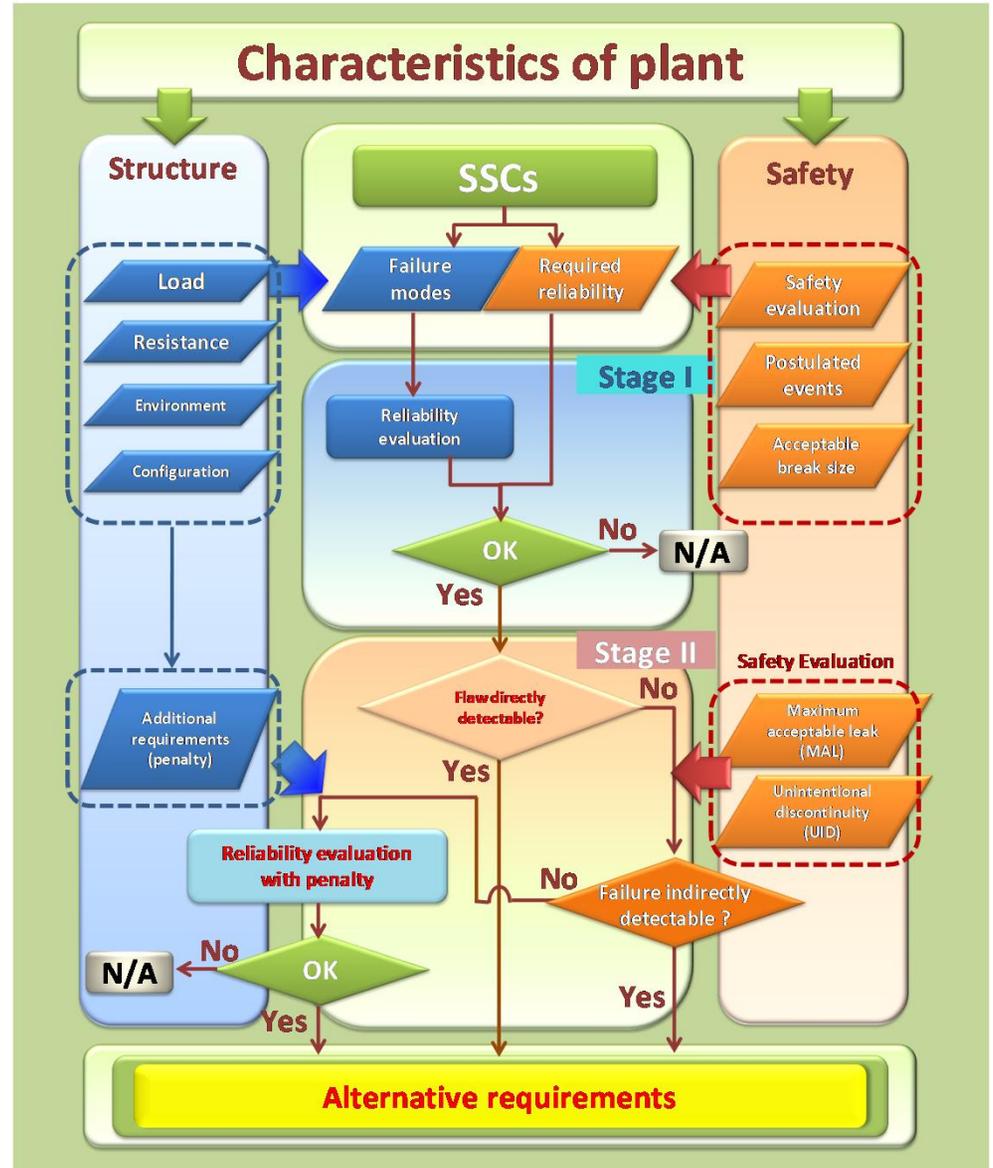


Asada, Y., Japanese Activities Concerning Nuclear Codes and Standards – Part II, Journal of Pressure Vessel Technology, ASME 128 (2006) 64.

- 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

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- 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.