December 7 - 11, 2009
Kyoto, Japan

Development of elevated temperature structural design methods to realize compact reactor vessels

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- Thermal Load Modeling Method
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Introduction（Background）

Innovative Technologies for JSFR (Japan Sodium Fast Reactor)

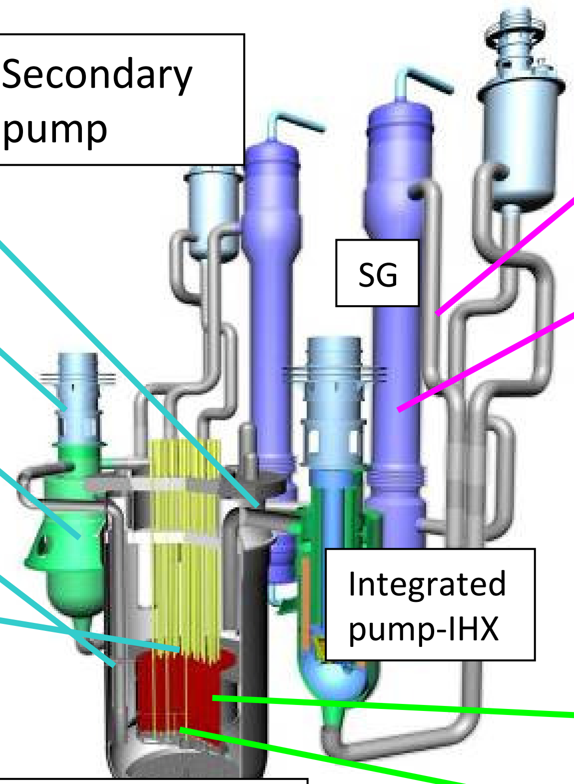
Economic Competitiveness

○ **Reduction of Mass & Volume**

- ① Shortened piping with high chromium steel
- ② 2 loop cooling system
- ③ Integrated pump-IHX component
- **④ Compact reactor vessel**
- ⑤ Simplified fuel handling system
- ⑥ CV with steel plate reinforced concrete building

○ **Long operation by high burn-up fuel**

- ⑦ Advanced fuel material



⑭ Plant design study (D-FR/C-FR)

⑮ Large-scale sodium test

Higher reliability

○ **Sodium technology**

- ⑧ Sodium leak tightness with double-walled piping
- ⑨ Higher reliable SG with double-walled tube
- ⑩ Higher inspection ability inside of sodium boundary

Higher safety

○ **Core safety**

- ⑪ Passive shutdown and decay heat removal
- ⑫ Re-criticality free core

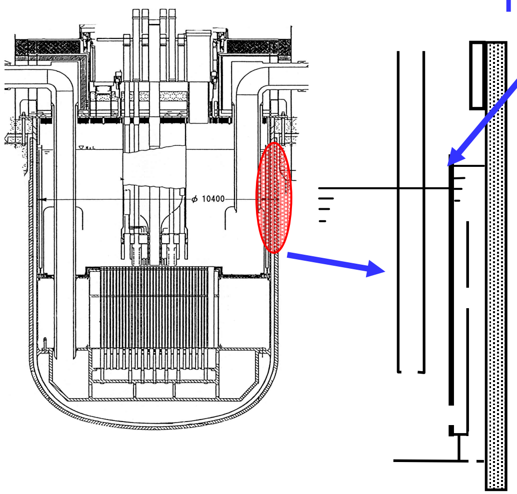
○ **Seismic reliability**

- ⑬ Seismic reliability in core assemblies

Introduction (Aim of Study)

Compact Reactor Vessel for JSFR

Elimination of Thermal Protection System

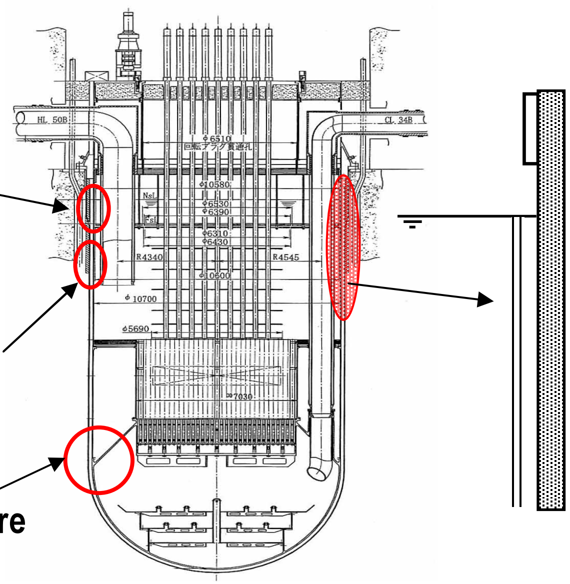


Thermal Protection System

Near Liquid surface

Thermal stratification

Core support structure



Scope; Evaluation of structural integrity against thermal loadings

Old design	660 MWe
RV diameter	Φ10.4 m
Thickness	50 mm
Life	40 years

DFBR

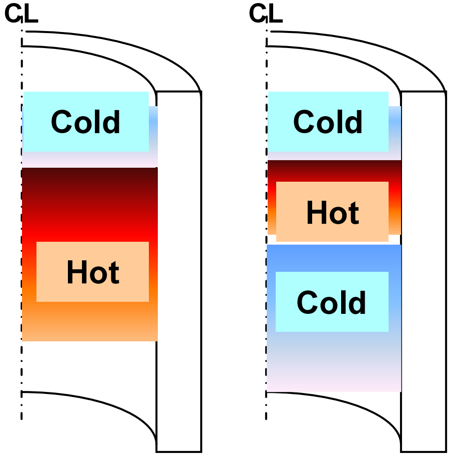
New design	500～750MWe (tentative)
RV diameter	φ8.3～8.85m (tentative)
Thickness	30mm～50mm
Life	60 years

JSFR

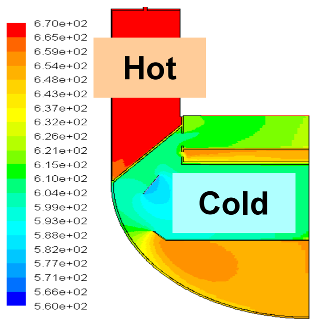
Expected failure modes of Reactor Vessels

Thermal loads

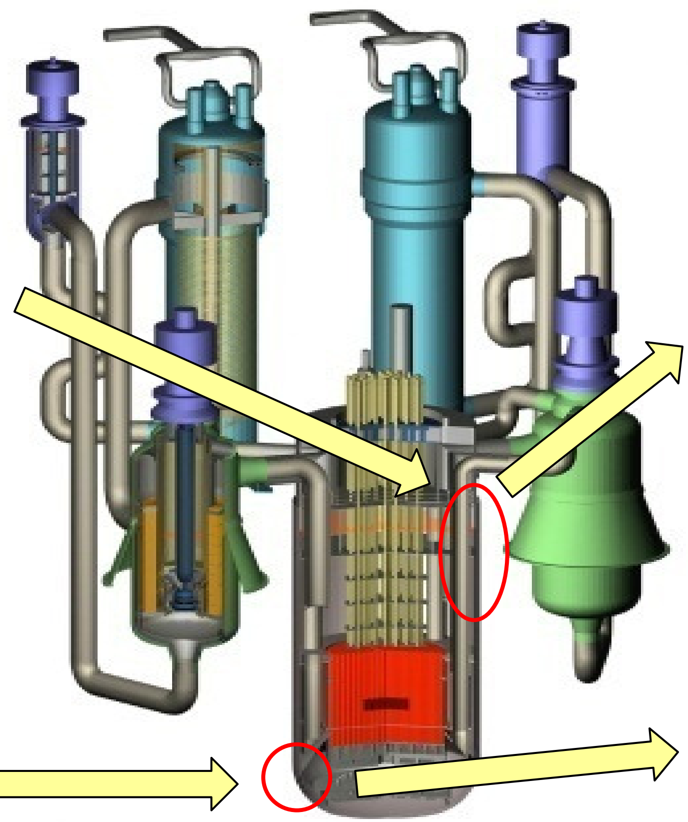
Main loads are thermal stresses induced by fluid temperature change at transient operation



Start up **Shut down**

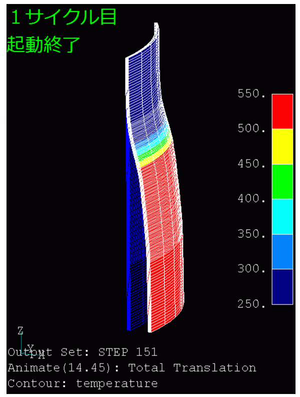


Shut down

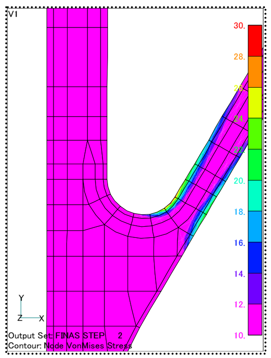


Structural response

Elastic plastic creep response under elevated temperature



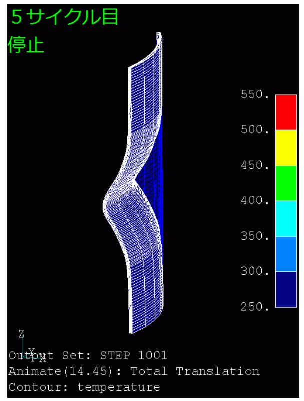
Residual strain



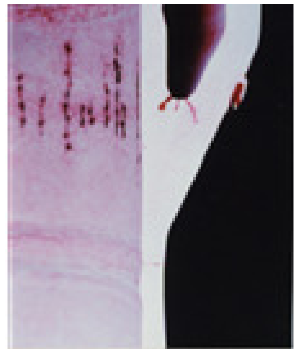
Strain concentration

Failure

Assumed Failure modes during long term operation



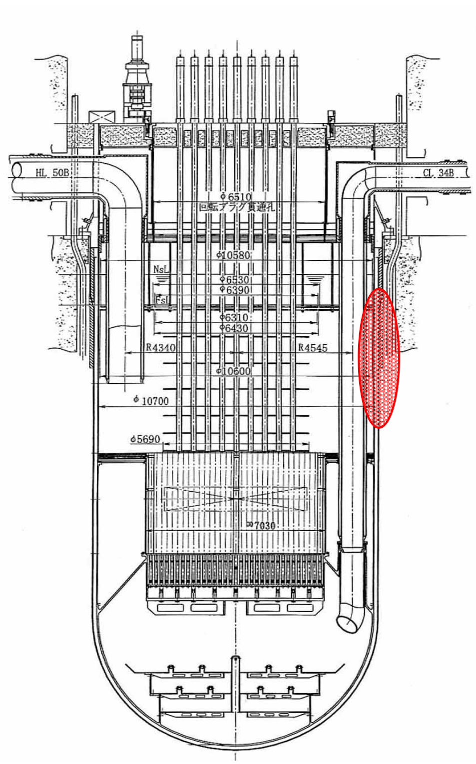
Incremental deformation



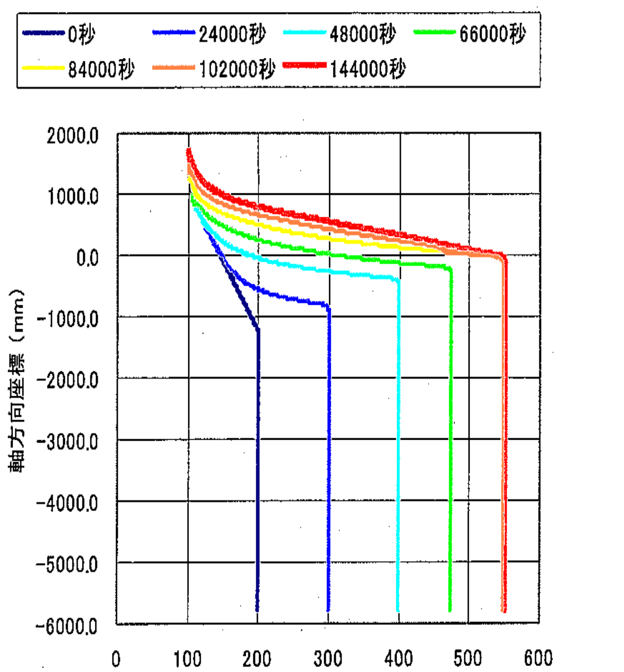
Creep fatigue crack

Characteristics of thermal transient loads in reactor vessels

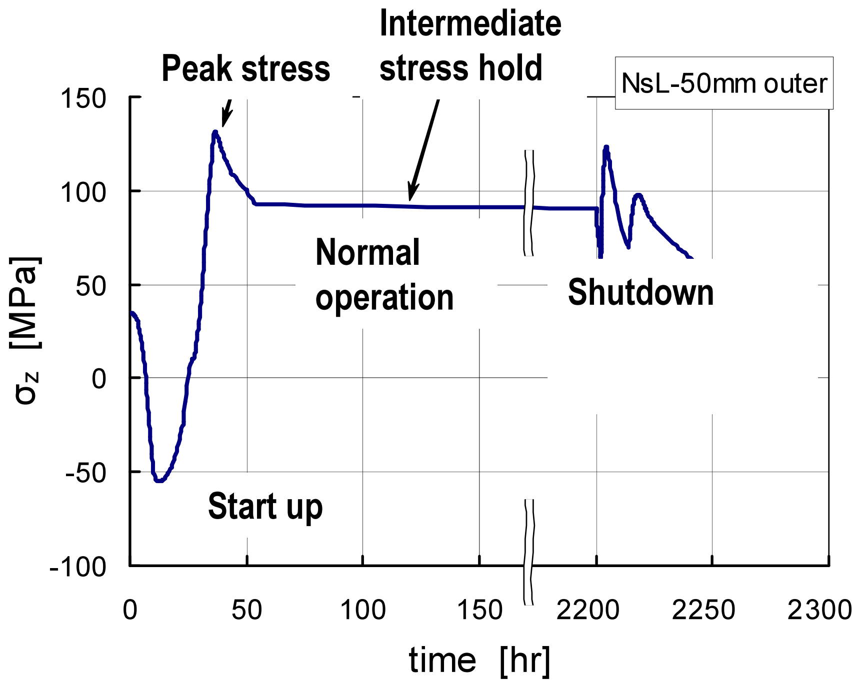
Thermal transient load histories of reactor vessels exhibit intermediate stress hold after the maximum peak stress. By changing hypothesis of peak stress hold to consideration of intermediate stress hold for creep damage calculation, accuracy of damage prediction will be much improved.



Axial temperature distribution



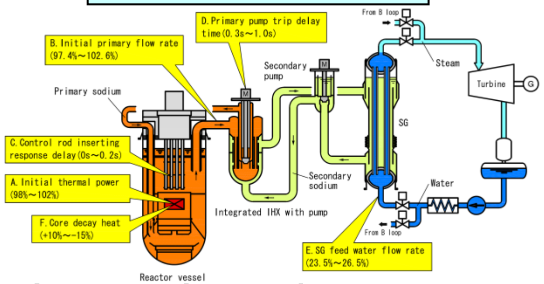
Temperature (℃)distribution



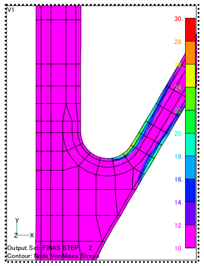
Development of the New Design Method for Reactor Vessels

Design Procedure

Load Modeling

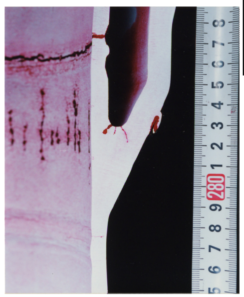


Conservative Modeling against Variety of Design Conditions



Structural Analysis

Prediction of Inelastic Response under High Temperature Condition



Strength Evaluation

Prediction of Ratchet Deformation and Creep-fatigue damage during 60 years

Conventional Design Method

Design Conditions
↓
Thermal hydraulic analyses
↓
Conservative thermal transient condition
↓ Fluid temperature
Structural Analyses (Elastic Analyses)
↓ Stress/Strain
Strength Evaluation for Each Failure Mode
↓
Allowable design ? — No. → back to Structural Analyses
↓ Yes.
One example of Conservative Design

Conservative but not Optimized Design

New Design Method

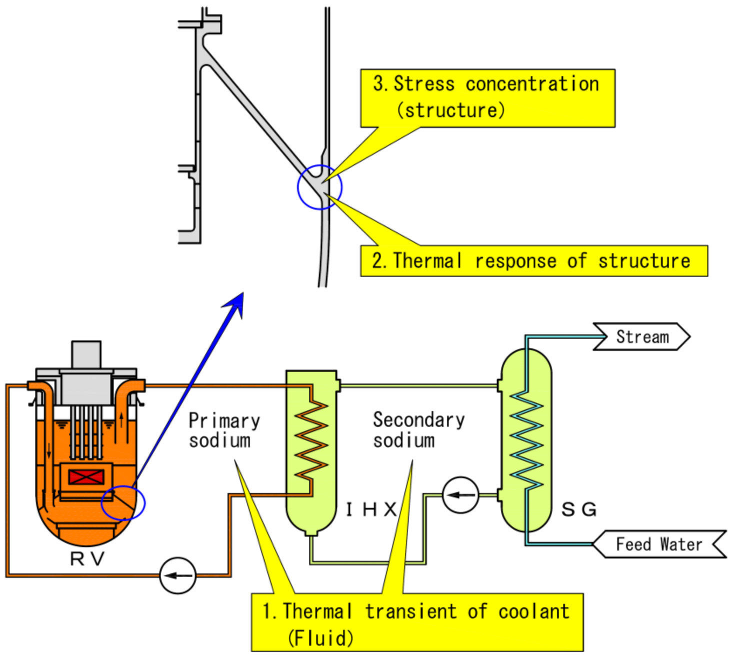
Design Conditions
↓
Thermal hydraulic structure total analyses
↓
The maximum stress condition
↓ Fluid temperature
Structural Analyses (Inelastic Analyses)
↓ Stress/Strain
Strength Evaluation considering interaction among Failure Modes
↓
Allowable design ? — No. → back to Design Conditions / The maximum stress condition / Structural Analyses
↓ Yes.
Optimum Design

Positively Optimum Design

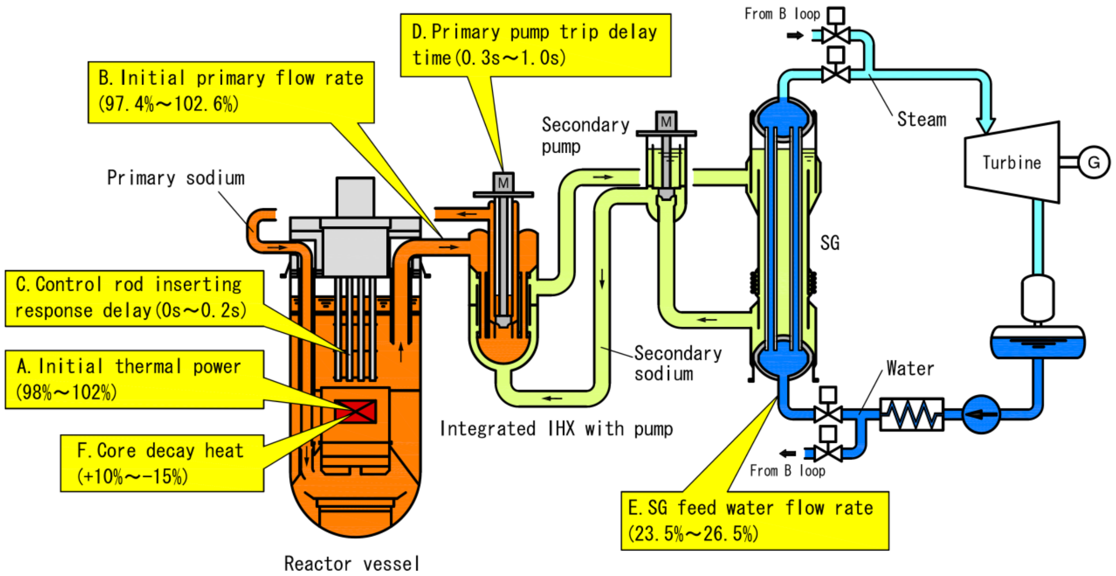
Thermal Load Modeling Method

◆Thermal load is a main loading of fast reactors. Thermal transient loads are generated by both thermal-hydraulic and structural phenomena and have many influence parameters.

◆New thermal load modeling method is developed based on direct evaluation of relationship between thermal-hydraulic behaviors and thermal stress.

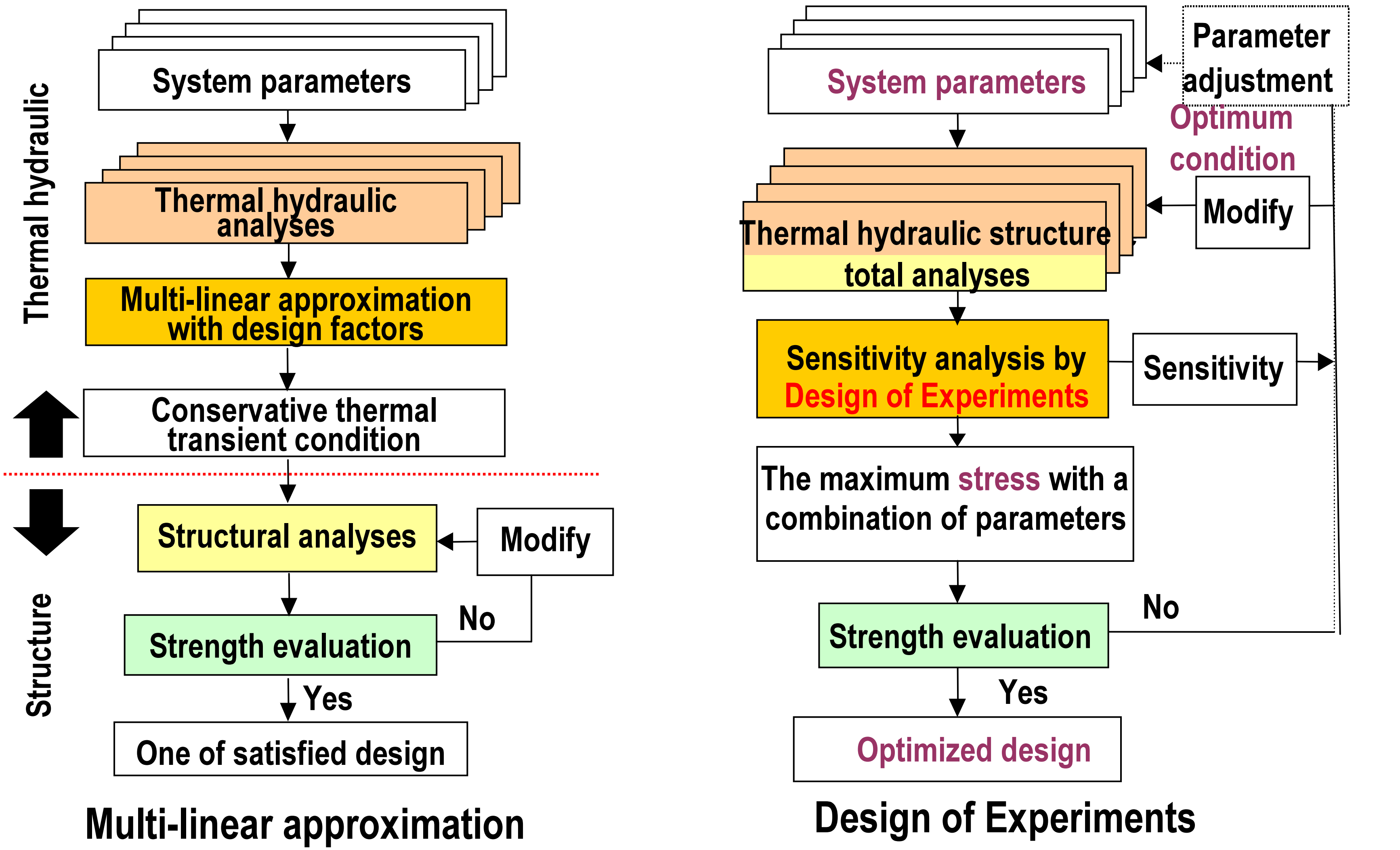


Thermal loads induced by thermal-hydraulic and structural phenomena



Many influence parameters on thermal loads

Comparison of thermal load modeling methods



206.2 MPa

176.0 MPa

Example prediction of thermal stress at the wall of upper part of reactor vessel

Inelastic Design Analysis Method

Basic policies

○ More rational evaluation than conventional methods based on elastic analysis

○ Conservative evaluation taking design uncertainty into account

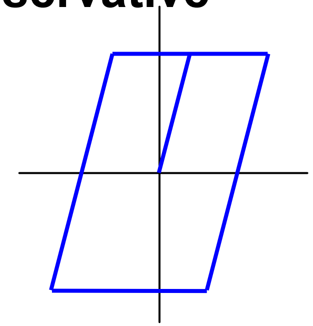
○ Limitation of applicable area to ensure conservative results

← General application requires too many parameters which affect on inelastic calculation results

Development of advanced constitutive equations for design

Elastic Perfectly plastic model

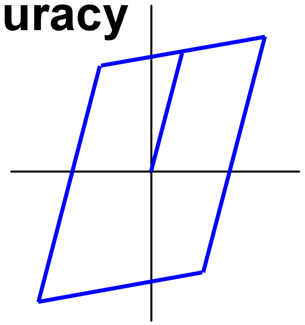
・Conservative



Bi-linear kinematic hardening model

・Dependency on assumed strain

・low-accuracy

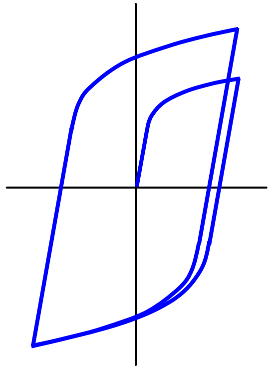


Improvement of models

Simplified model

・Stress reverse reset model(MK-SRR)

・Multi-linear kinematic hardening model(MK)



・Nonlinearity of Stress-Strain relation

・Temperature dependency

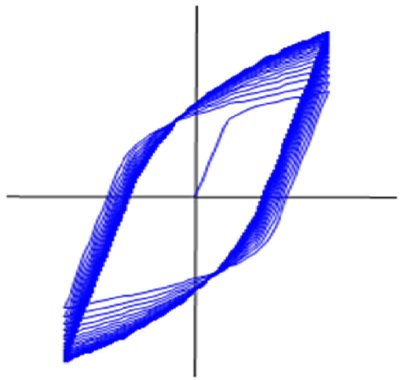
Non-proportional Multi-axial Behaviors

High-accuracy model

・Multi-linear kinematic hardening model(MCP)

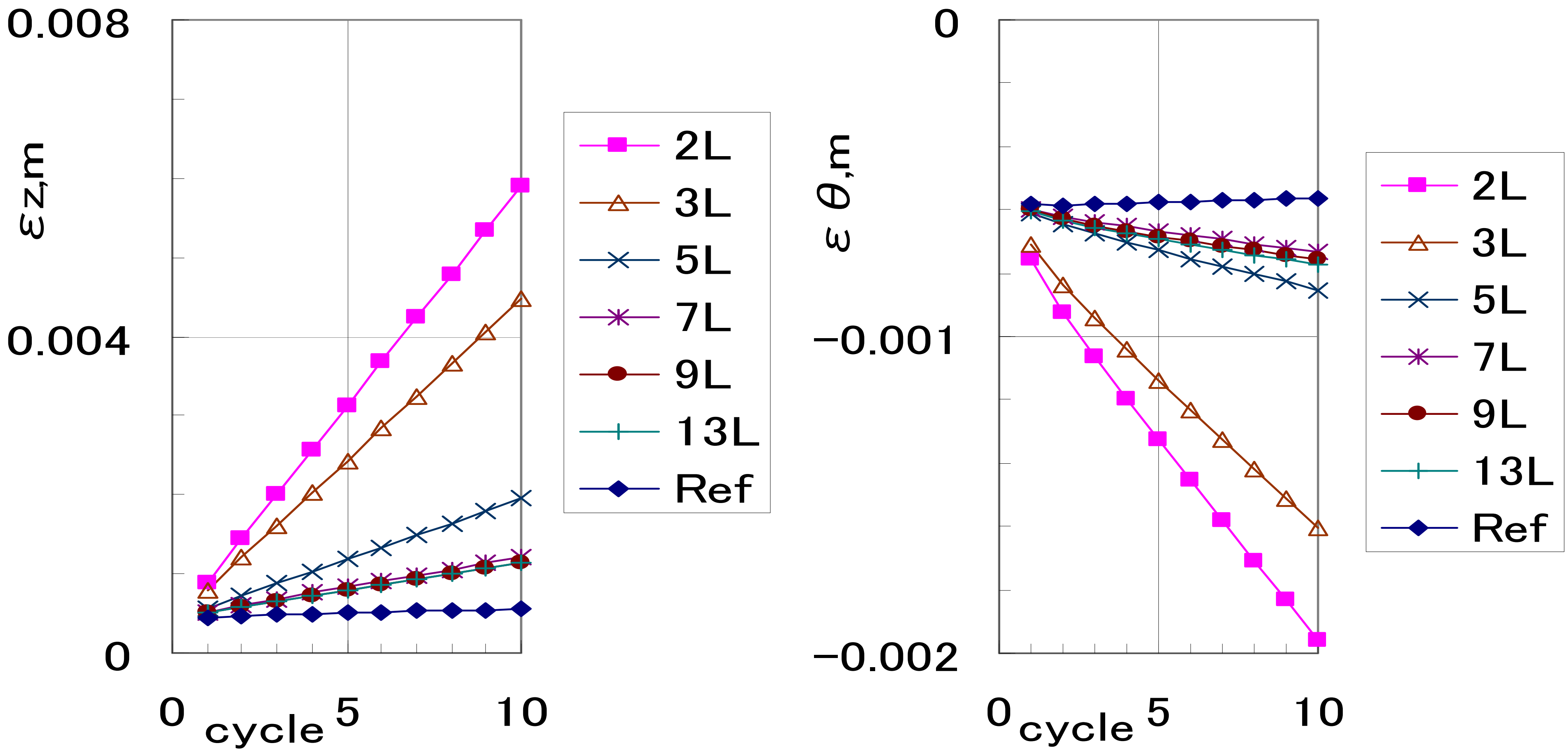
・Two-surface model(TCP)

・Cyclic hardening behavior



Application of MK Model and MK-SRR Model

Analysis of Reactor Vessel Model



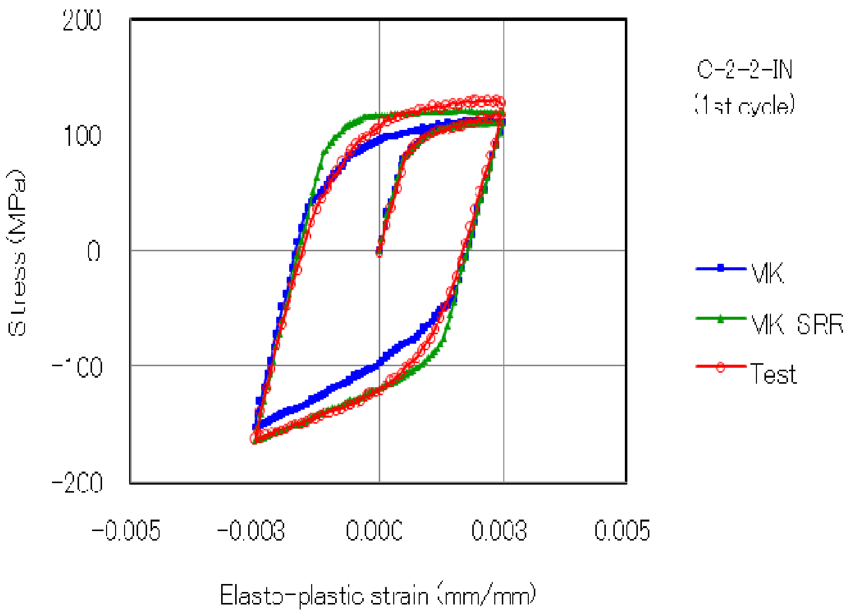
Strain behaviour by MK-SRR model with various number of multi-linear approximation

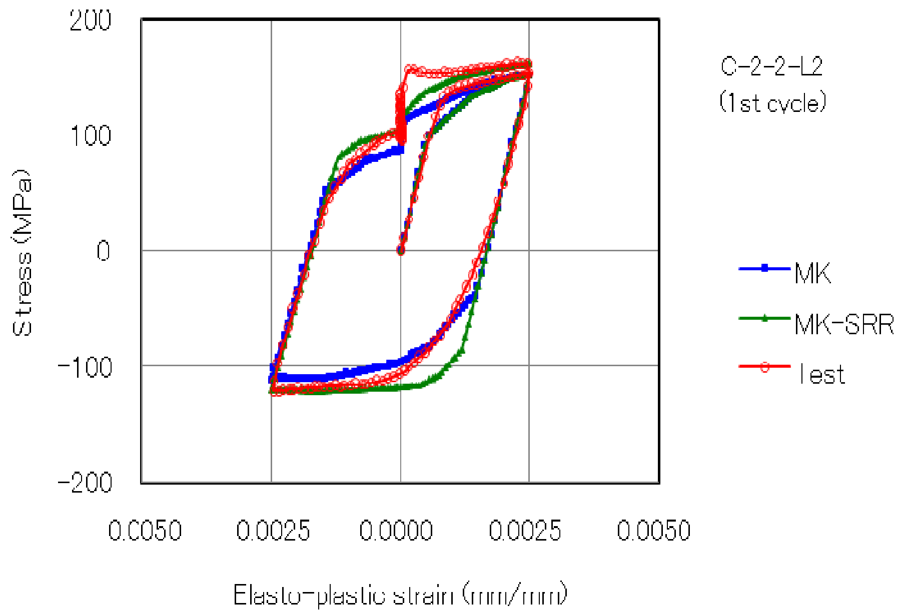
Experiments for verification and validation

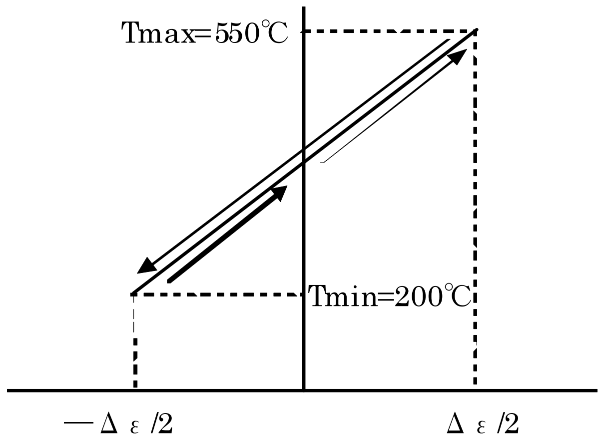
Uni-axial tests	Basic material properties	to determine parameters (thermal expansion test/ tensile tests/cyclic tests)
	Temperature dependency tests	considering of the preliminary analysis results
Basic structure element tests	Three bar ratcheting test	(two bar ratcheting theory)
	Bi-axial ratcheting tests	considering of the preliminary analysis results
Structure model tests	Cylindrical specimen /movement of temperature distribution	similar phenomenon as the ratcheting in the reactor vessel

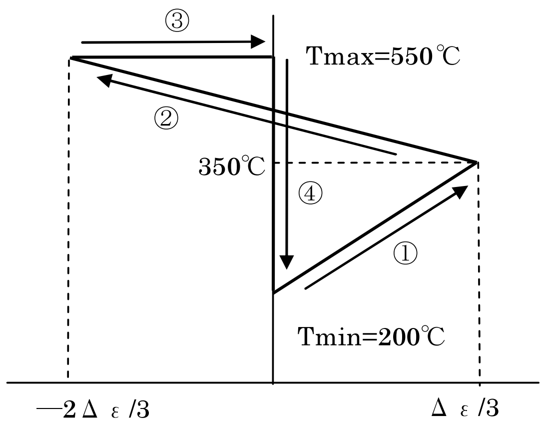
Verification of MK Model and MK-SRR Model

Temperature dependency tests



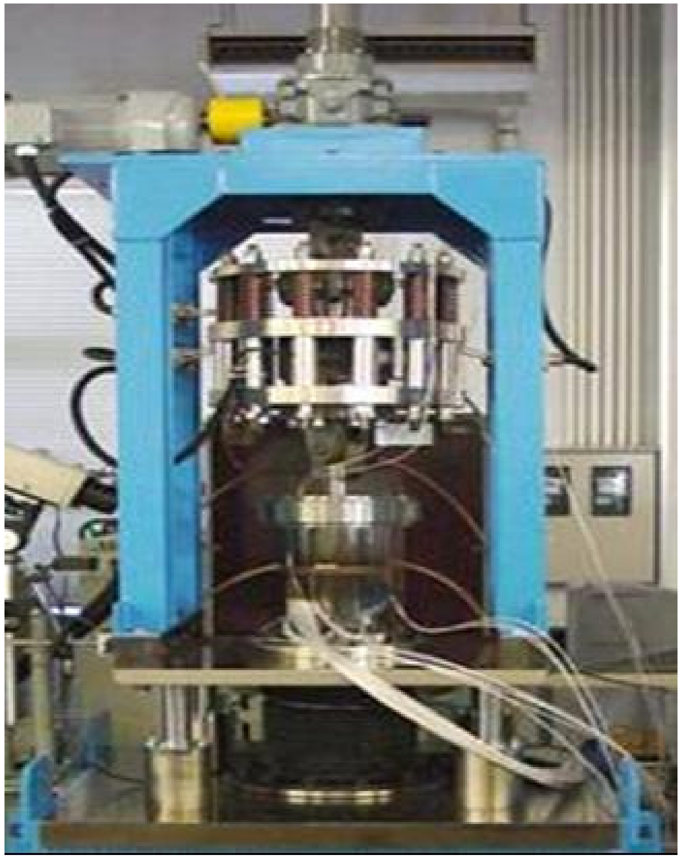


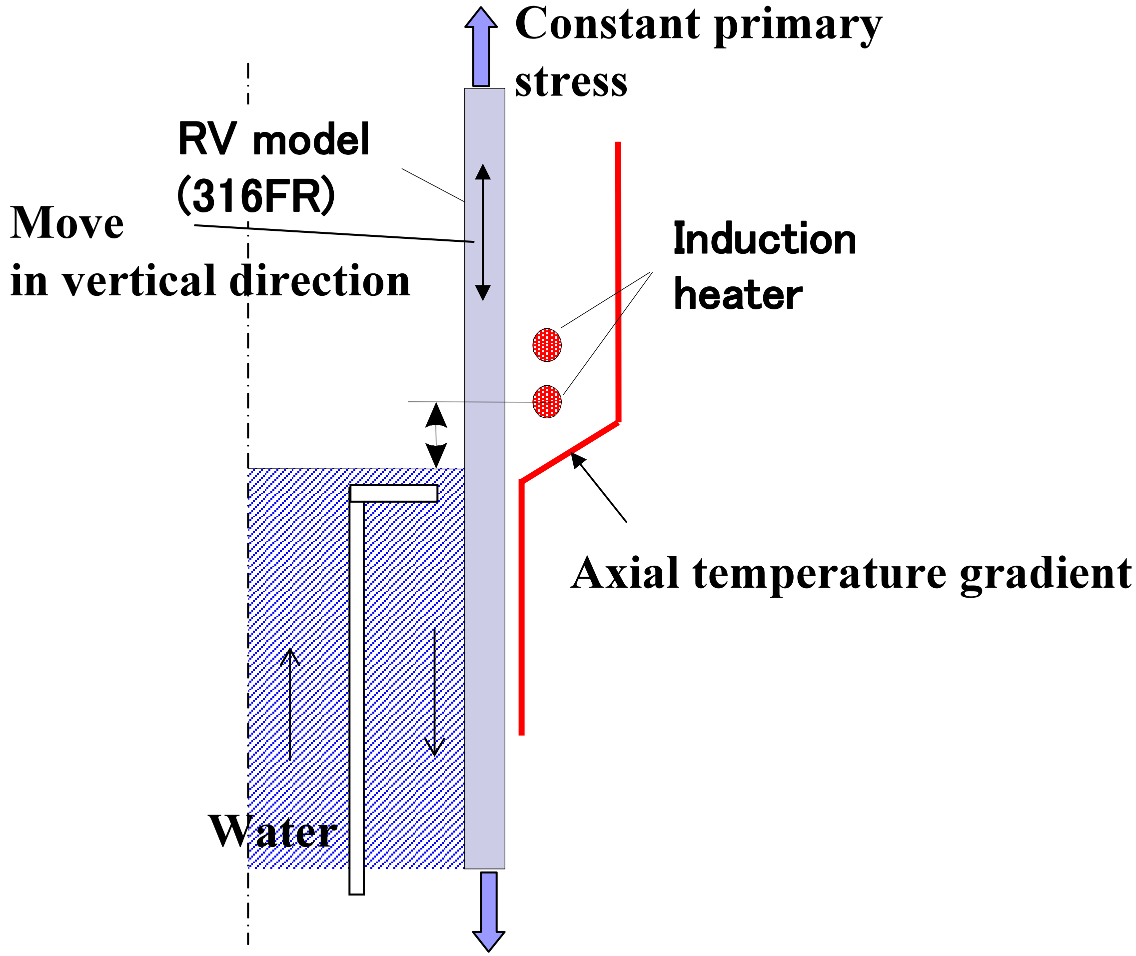




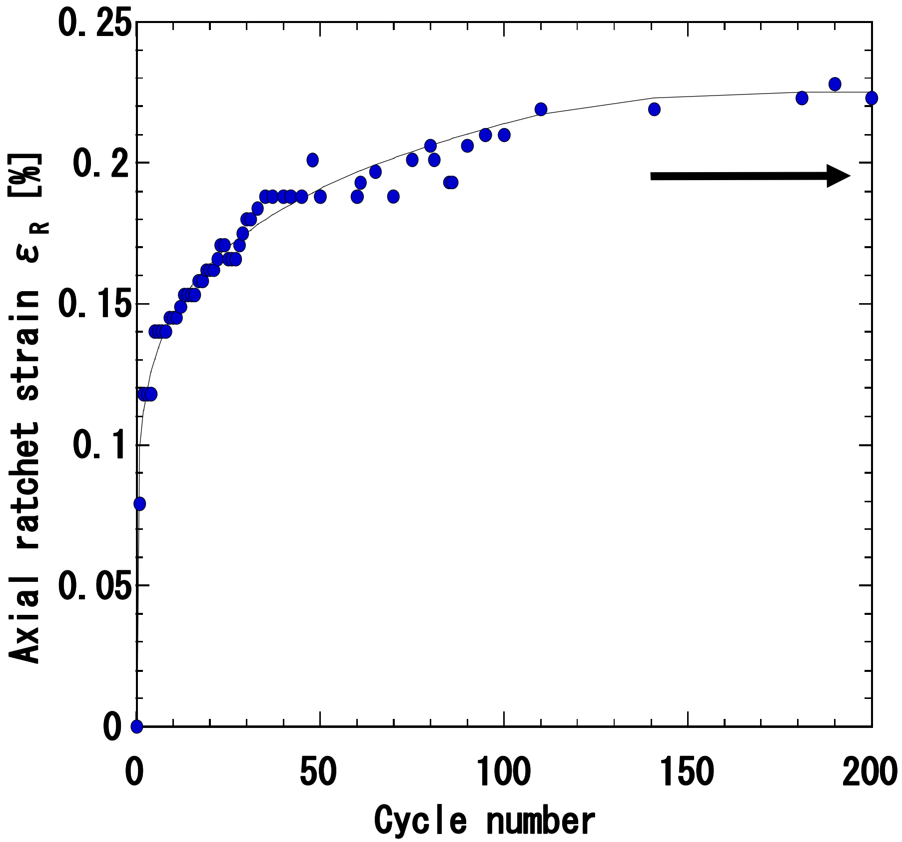
Structural model tests for validation

Measurements: strain on the surface, (creep) fatigue cracks

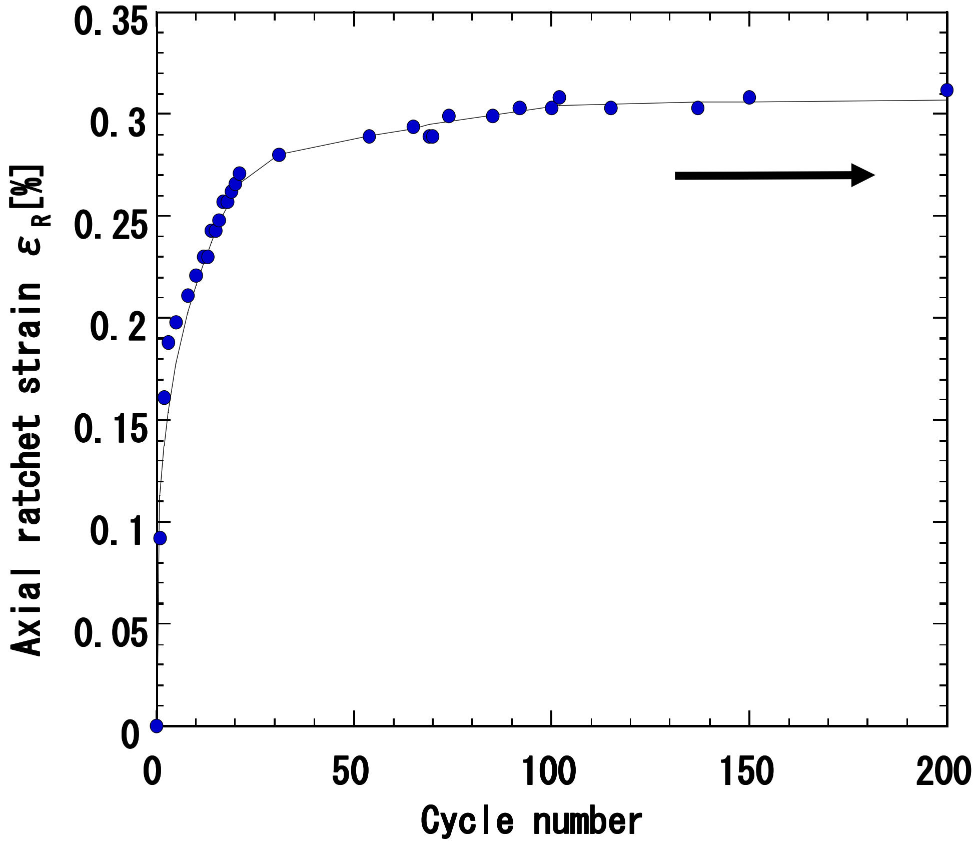




Measured strain on the outer surface of vessel wall during 200cycle of thermal transient loads



No.A Without hold

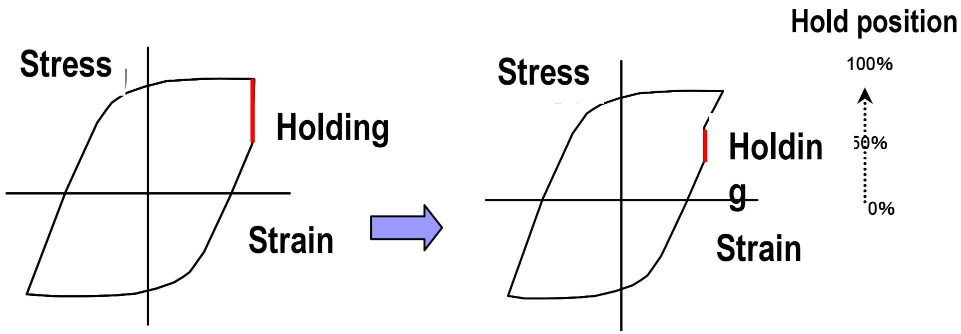


No.B With intermediate hold

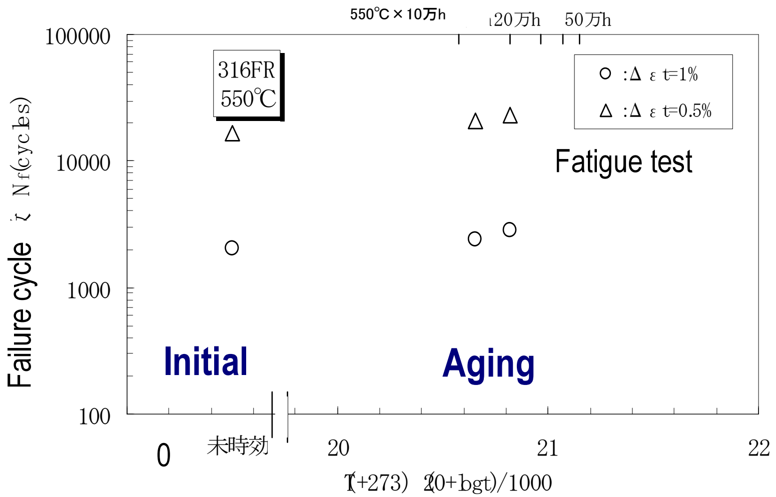
Elevated Temperature Strength Evaluation Method

Creep-fatigue damage evaluation with consideration of following effects

Intermediate stress hold creep-fatigue



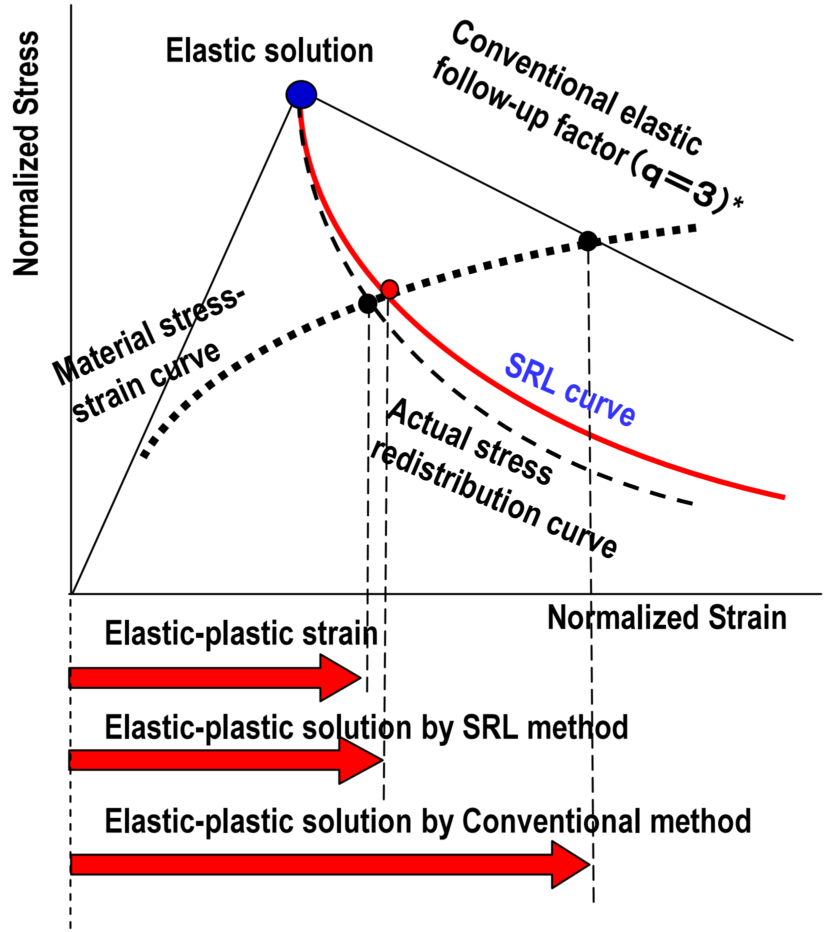
Aging



Strength tests after aging

Strain Concentration

Strain concentration due to plasticity and creep

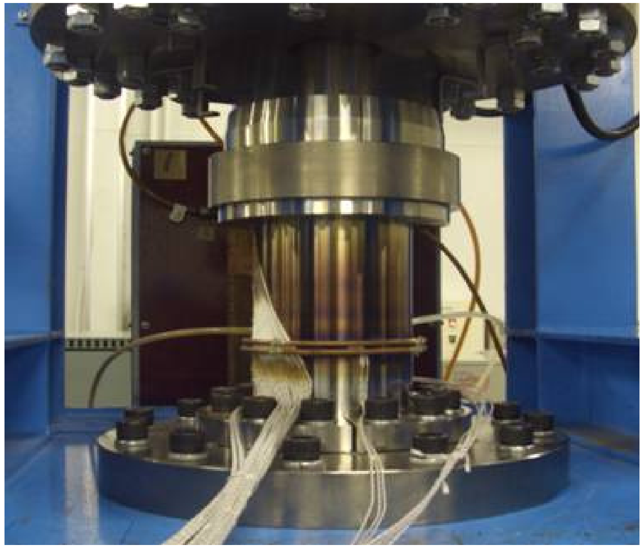


Strength tests of reactor vessel models for validation

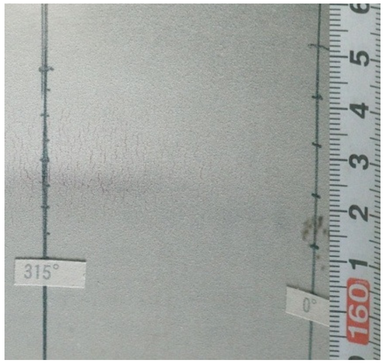
Objective : Total validation of creep fatigue strength evaluation method by reactor vessel models subjected to simulated design load conditions

Load conditions

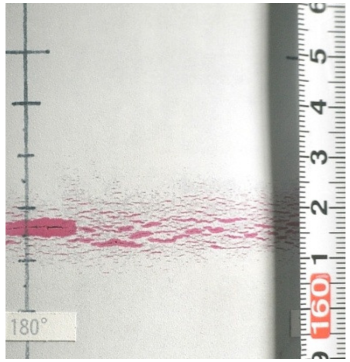
Temperatur	600℃		
Thermal stress	1000MPa		
Primary stress	50MPa	30MPa	50MPa
Hold time	0hr	3hr	3hr



Reactor vessel model



Inner surface



Outer surface

Observed cracks after 2000cycles (30MPa, 3hr)

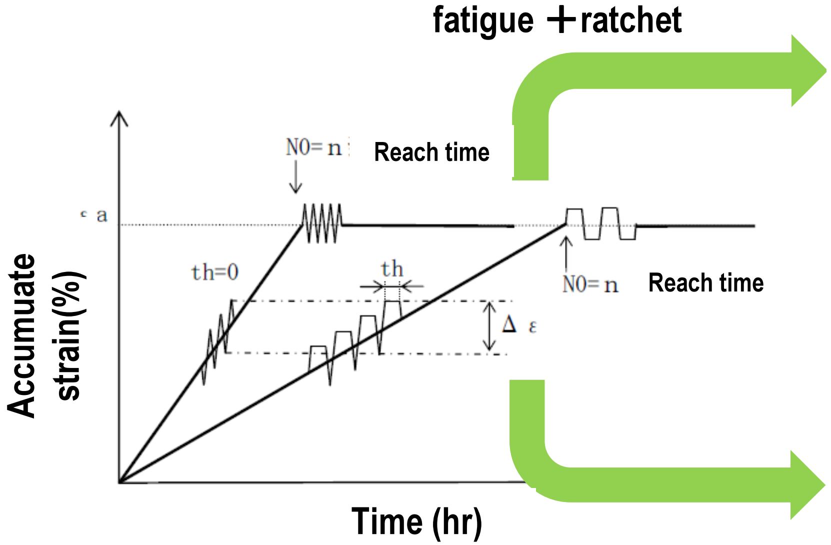
Allowable Strain Limit

Accumulate strain is limited within the value which has no impact on fatigue and creep-fatigue strength.

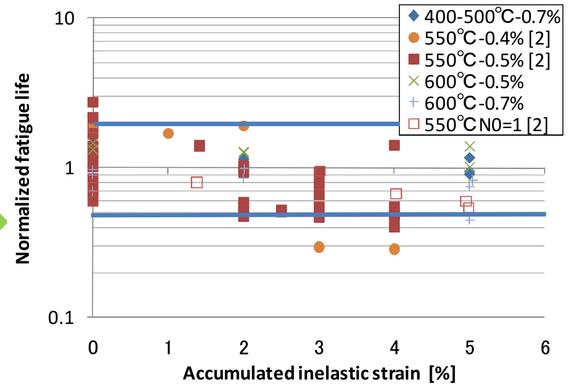
Fatigue strength tests with ratchet strain

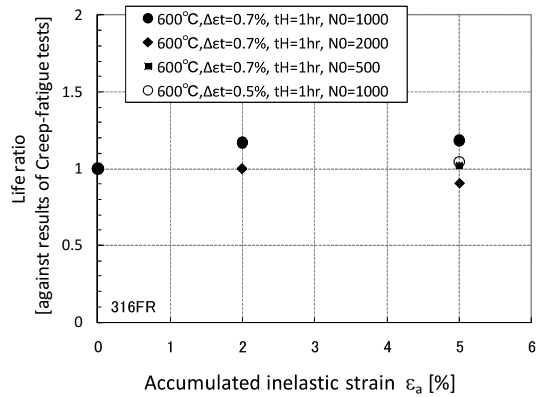
Creep-fatigue strength tests with ratchet strain

Simulated loading of reactor vessels



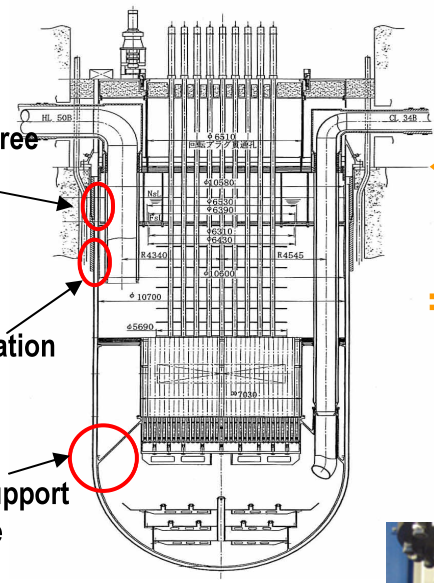
Creep-fatigue +ratchet





Strength reduction is small in creep-fatigue + ratchet tests compared with fatigue + rachet ones. From the later results, allowable strain limit is determed as 2%.

Structural Design Evaluation Method for Reactor Vessels



Sodium free surface

Thermal stratification

Core support structure

Extension of applicable area

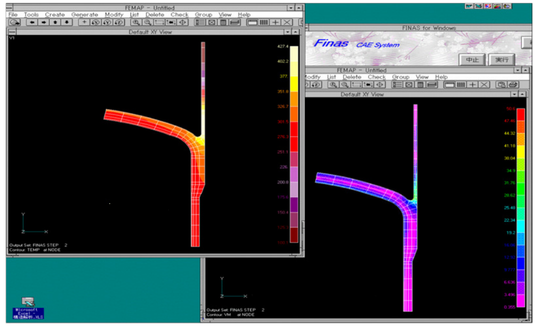
Apply

Check

Design evaluation methods

- Thermal load modeling
- Inelastic analysis
- Strength Evaluation

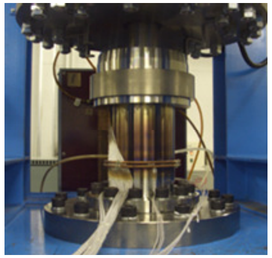
+



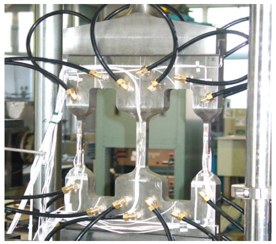
Structural analysis program

+

Database

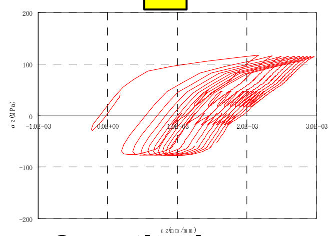


Structural tests



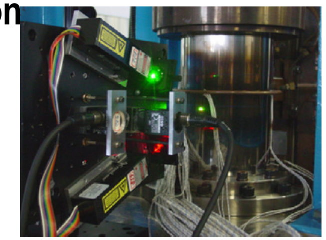
Structural element tests

Material tests



Constitutive equation

Numerical algorithm



Strain measurement

Conclusions

The Structural Design Evaluation Method for Reactor Vessels of FBRs was proposed with :

- **Thermal Load Modeling Method**

  Thermal hydraulic - structure total analysis with experimental design

- **Inelastic Design Analysis Method**

  Design evaluation method based on recommended inelastic analysis

- **Elevated Temperature Strength Evaluation Method**

  Creep-fatigue damage evaluation method considering intermediate stress hold, aging and strain concentration

  Strain limit within negligible effects on creep-fatigue strength