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

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- Introduction (background)
- Thermal Load Modeling Method
- Inelastic Design Analysis Method
- Elevated Temperature Strength Evaluation Method
- Conclusion
Innovative Technologies for JSFR (Japan Sodium Fast Reactor)

Introduction (Background)

**Economic Competitiveness**
- **Reduction of Mass & Volume**
  1. Shortened piping with high chromium steel
  2. 2 loop cooling system
  3. Integrated pump-IHX component
- **Compact reactor vessel**
  4. Shortened piping with high chromium steel
- **Simplified fuel handling system**
- **CV with steel plate reinforced concrete building**
- **Long operation by high burn-up fuel**
  7. Advanced fuel material

**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

**Plant design study (D-FR/C-FR)**

**Large-scale sodium test**
Introduction (Aim of Study)

Compact Reactor Vessel for JSFR

Elimination of Thermal Protection System

Scope: Evaluation of structural integrity against thermal loadings

Old design
- 660 MWe
- RV diameter: φ10.4 m
- Thickness: 50 mm
- Life: 40 years

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

DFBR

JSFR
Expected failure modes of Reactor Vessels

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

**Structural response**
Elastic plastic creep response under elevated temperature

**Failure**
Assumed Failure modes during long term operation

- Residual strain
- Strain concentration
- 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.
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**

1. **Design Conditions**
2. **Thermal hydraulic analyses**
3. **Conservative transient condition**
4. **Structural Analyses (Elastic Analyses)**
5. **Strength Evaluation for Each Failure Mode**
6. **Allowable design?**
   - No.
   - Yes.
   - One example of Conservative Design

**New Design Method**

1. **Design Conditions**
2. **Thermal hydraulic structure total analyses**
3. **The maximum stress condition**
4. **Structural Analyses (Inelastic Analyses)**
5. **Strength Evaluation considering interaction among Failure Modes**
6. **Allowable design?**
   - No.
   - Yes.
   - 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

Multi-linear approximation

| System parameters | Parameter adjustment
<table>
<thead>
<tr>
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<th></th>
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</thead>
<tbody>
<tr>
<td>Multi-linear approximation with design factors</td>
<td>Optimum condition</td>
</tr>
<tr>
<td>Conservative thermal transient condition</td>
<td>Modify</td>
</tr>
<tr>
<td>Structural analyses</td>
<td>Sensitivity analysis by Design of Experiments</td>
</tr>
<tr>
<td>Strength evaluation</td>
<td>The maximum stress with a combination of parameters</td>
</tr>
<tr>
<td>No</td>
<td>Sensitivity</td>
</tr>
<tr>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>One of satisfied design</td>
<td>Optimized design</td>
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Example prediction of thermal stress at the wall of upper part of reactor vessel:

- 206.2 MPa
- 176.0 MPa
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

<table>
<thead>
<tr>
<th>Uni-axial tests</th>
<th>Basic material properties</th>
<th>to determine parameters (thermal expansion test/ tensile tests/cyclic tests)</th>
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<tr>
<td></td>
<td>Temperature dependency tests</td>
<td>considering of the preliminary analysis results</td>
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<td>Basic structure element tests</td>
<td>Three bar ratcheting test</td>
<td>(two bar ratcheting theory)</td>
</tr>
<tr>
<td>Bi-axial ratcheting tests</td>
<td></td>
<td>considering of the preliminary analysis results</td>
</tr>
<tr>
<td>Structure model tests</td>
<td>Cylindrical specimen/movement of temperature distribution</td>
<td>similar phenomenon as the ratcheting in the reactor vessel</td>
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</table>
Verification of MK Model and MK-SRR Model

Temperature dependency tests

\[ \Delta \varepsilon / 2 \] (Tmax=550°C) - \[ \Delta \varepsilon / 2 \] (Tmin=200°C)

\[ \Delta \varepsilon / 3 \] (Tmax=550°C) - 2 \[ \Delta \varepsilon / 3 \] (Tmin=200°C)

[Graphs showing stress-strain relationship at different temperatures]
Structural model tests for validation

Measurements: strain on the surface, (creep) fatigue cracks
Measured strain on the outer surface of vessel wall during 200 cycle of thermal transient loads

No. A  Without hold          No. B  With intermediate hold

Axial ratchet strain $\varepsilon_R$ [%]

Cycle number
Elevated Temperature Strength Evaluation Method
Creep-fatigue damage evaluation with consideration of following effects

**Intermediate stress hold creep-fatigue**

**Strain Concentration**

Strain concentration due to plasticity and creep

**Aging**

Strength tests after aging

Fatigue test

Failure cycle $N$ (cycles)

Elastic solution

Conventional elastic follow-up factor $(q=3)^*$

Actual stress redistribution curve

Elastic-plastic strain

Elastic-plastic solution by SRL method

Elastic-plastic solution by Conventional method

Normalized Stress

Normalized Strain
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

<table>
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<th>Load conditions</th>
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<tr>
<td>Temparature</td>
</tr>
<tr>
<td>Thermal stress</td>
</tr>
<tr>
<td>Primary stress</td>
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<tr>
<td>Hold time</td>
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</table>

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

Strength reduction is small in creep-fatigue + ratchet tests compared with fatigue + ratchet ones. From the later results, allowable strain limit is determined as 2%.
Structural Design Evaluation Method for Reactor Vessels

Design evaluation methods

- Thermal load modeling
- Inelastic analysis
- Strength Evaluation

Database

Structural tests

Material tests

Numerical algorithm

Strain measurement

Constitutive equation

Thermal load modeling + Inelastic analysis + Strength Evaluation

Apply

Check

Sodium free surface

Thermal stratification

Core support structure

Extension of applicable area
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