Fundamental studies to improve analysis of accident progression at Fukushima Daiichi NPP

International Experts Meeting on Strengthening Research and Development Effectiveness in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant (IEM8)
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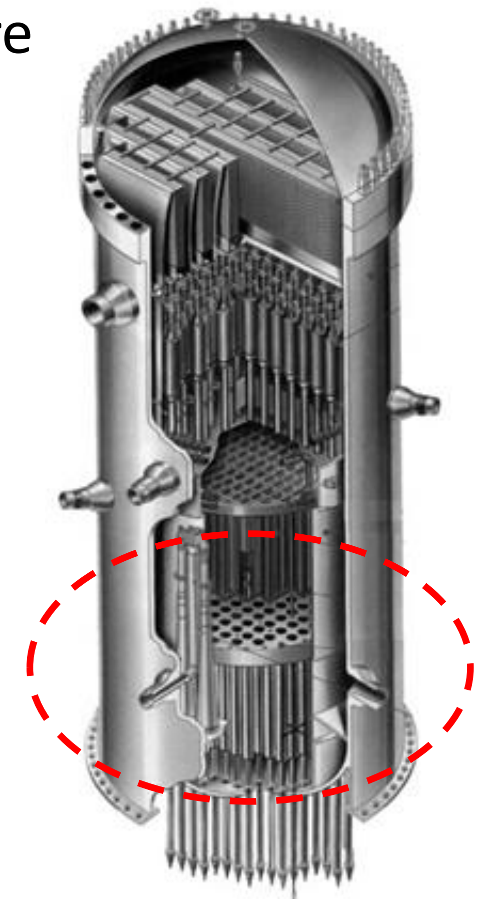
Background

- Data acquisition and model improvement for various phenomena during a severe accident (SA) are still required to develop/improve SA simulation codes and to evaluate effectiveness of accident mitigation measures in LWRs.
- Better understanding of key phenomena and increase in accuracy of analyses are also useful in estimation of the accident progress and status inside the Fukushima-Daiichi NPP (1F) for decommissioning.

Main differences between accidents at 1F and TMI-2

(Points for practical use of previous knowledge and consideration of research subjects)

- In-reactor structure and fuel bundle
  - BWR has more complicated lower plenum structure
- Inventory of core materials
  - Ratio of Zircaloy to UO_2 is higher in BWR
  - Control rod consists of B_4C and stainless steel
    (cf. Ag-In-Cd, ss and Zry in PWR)
- Accident scenario
  - Overheat and cooling conditions
  - Atmosphere (Oxygen potential)
  - Coolant level during the accident
- Extent of accident progress
  - Core exposure time (much longer in 1F)
  - Failure of reactor pressure vessels
- Advancement of analysis and experimental techniques

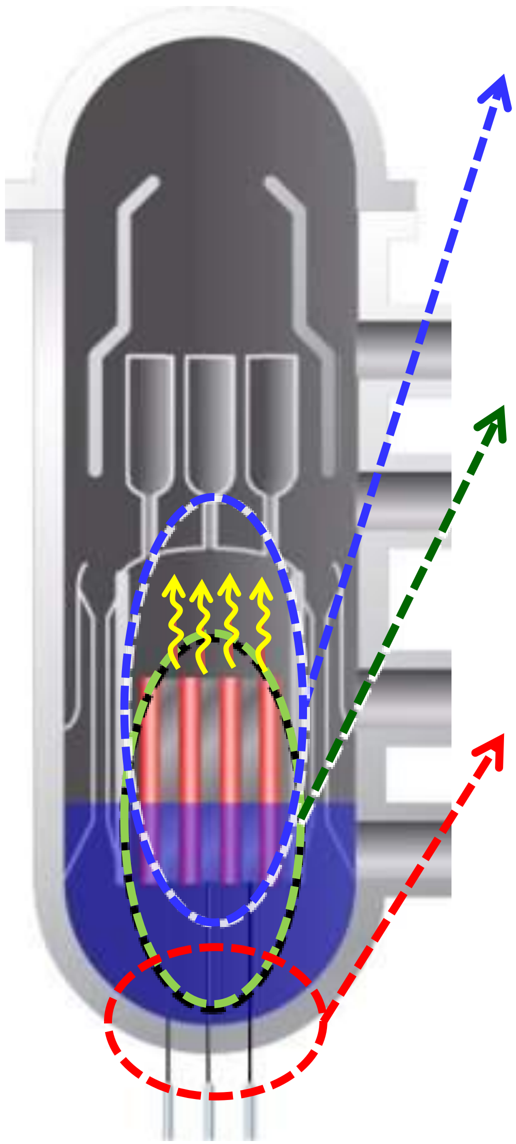


Research and development items

- JAEA conducts a wide range of R&D including
    - Thermal hydraulic behavior in RPV and PCV
    - Fuel and control rod damage and degradation process
    - Release, migration and deposition of fission products
    - Failure behavior of structural materials and pressure vessel
    - Molten materials relocation in the lower plenum region
    - Accident analysis and computer code development
    - Fuel debris characterization, etc.

  to meet the requirements from the reactor safety and decommissioning of 1F.
- *The R&D are mostly focused on phenomena in BWRs.*

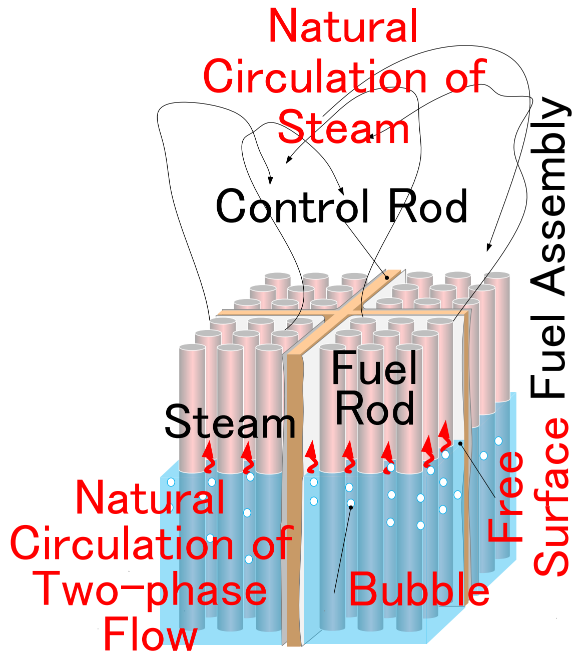
Today's Topics



- Thermal-hydraulics and coolability in reactor
  - Studies on effects of seawater on thermal-hydraulic characteristics in reactor core
  - Studies on behavior of molten fuel falling into coolant (Jet breakup)
- Fuel and control rod damage and degradation process
  - Fuel Melting and degradation test
  - Evaluation of FP release and transport behavior in the BWR system
- Failure of pressure vessel
  - Studies on behavior of structural materials and pressure vessel

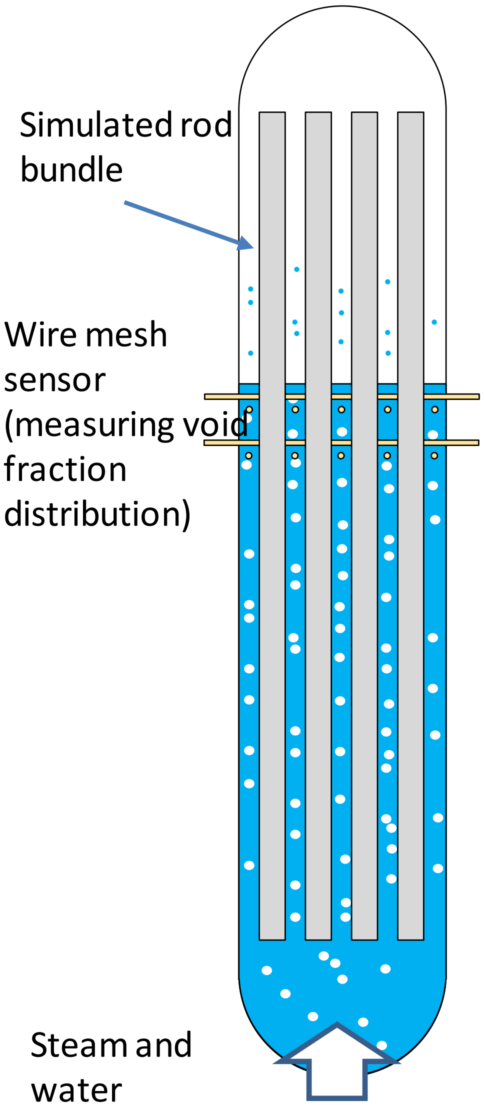
Thermal hydraulic behavior in reactor pressure vessel

- Development of numerical simulation method to evaluate temperature distribution in case of BWR rod bundle exposure
  - ✓ Effects of natural circulation of two-phase flow and steam flow
  - ✓ Construct experimental database by performing two-phase flow experiment with wire mesh sensor
- Thermo-hydraulic experiments for BWR core
  - ✓ Effects of internal structure (ex. CRGT) on jet breakup behavior (molten fuel behavior in coolant)
- Thermal-hydraulic experiments for effects of sea water (thermal hydraulic performance of sea water and effects of salt precipitation).



Heating of exposed BWR bundle

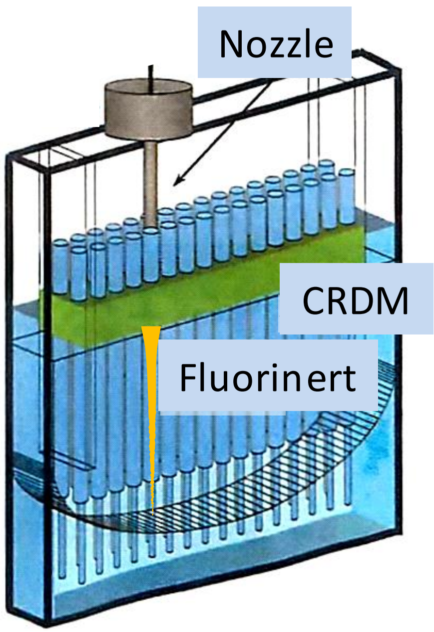
Obtained results are used for temperature evaluation of fuel bundle, reactor core, and RPV during SAs.



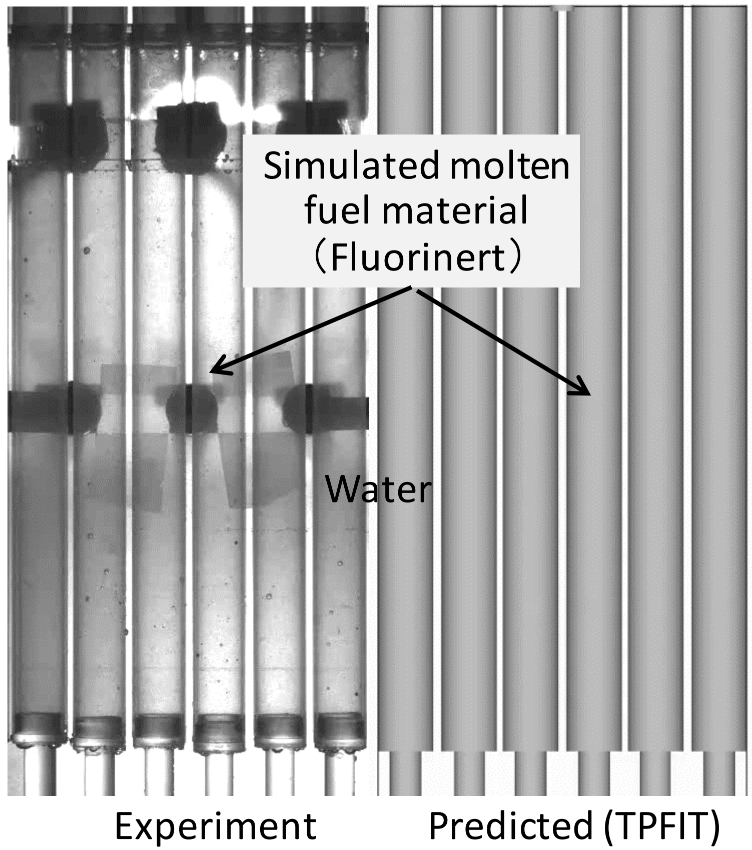
Outline of two-phase flow experiment

Behavior of the molten fuel falling into coolant (Jet breakup)

- Experimental and analytical studies to evaluate the behavior of the molten fuel falling into coolant and the influence of the complicated structures in the lower plenum of BWR

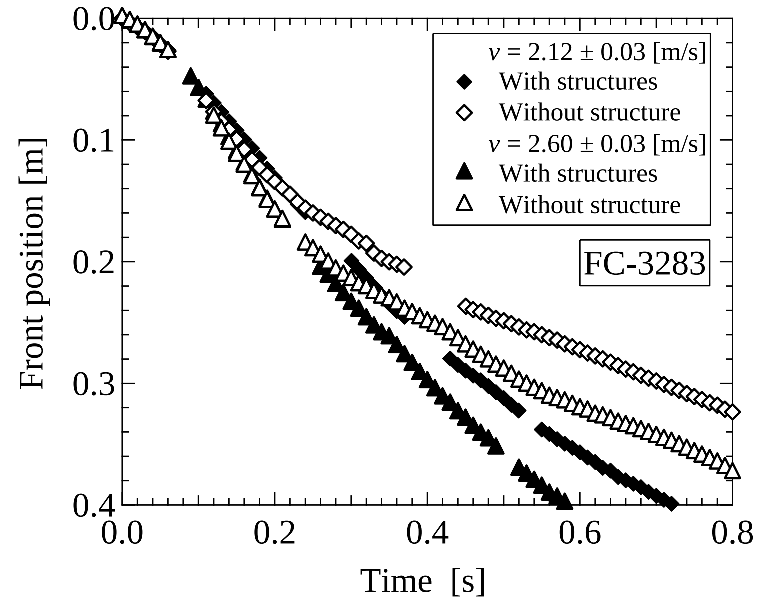


An experimental apparatus which simulates BWR lower structures (Cooperation with Tsukuba univ.)



Comparison between the calculation and experimental results

Detailed interface shape of jet is predicted by TPFIT qualitatively.

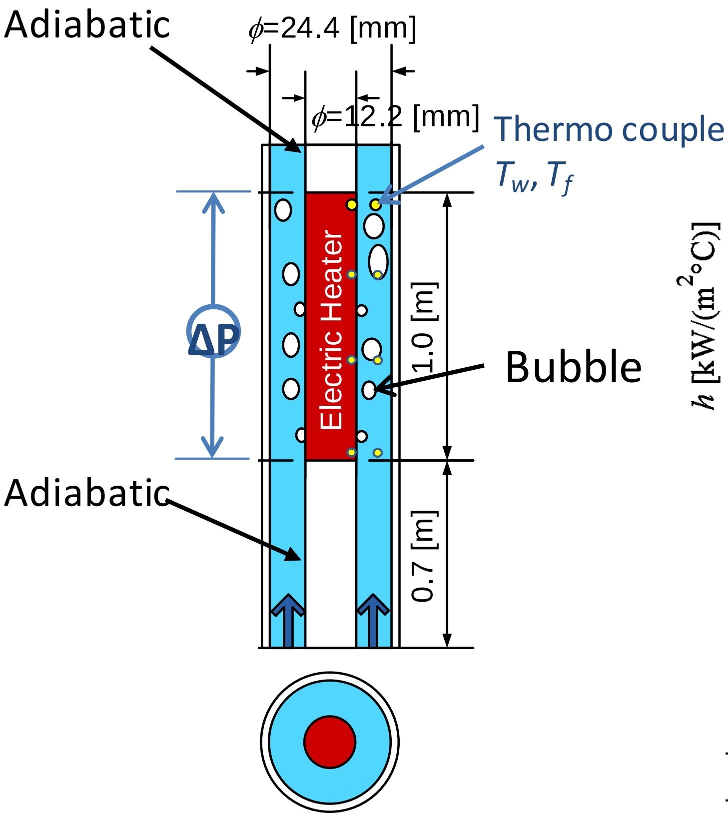


Effects of Structures on Front Position of Jet

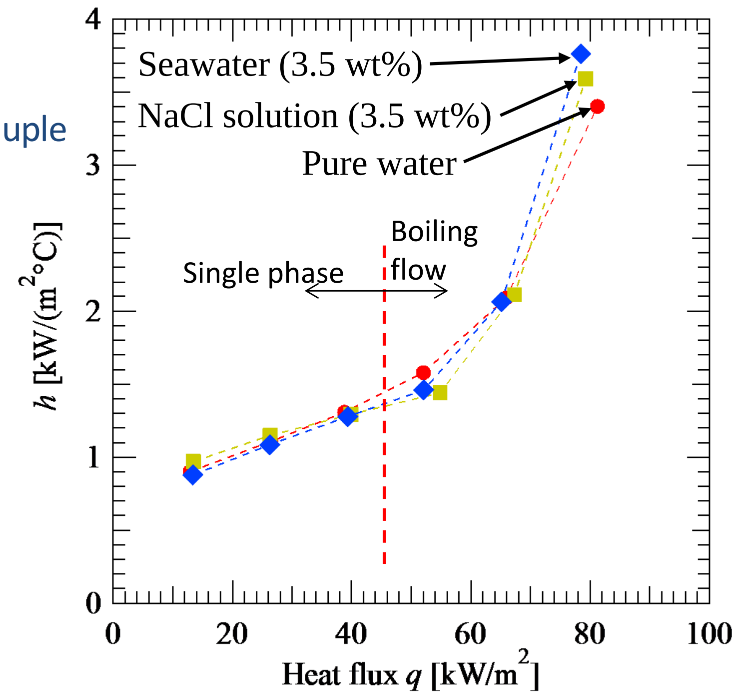
Front velocity of jet of "with structures" case, is higher than that of "without structure".

Thermal hydraulic characteristic of seawater

- Experimental study to evaluate the effects of seawater on thermal-hydraulic behavior in fuel bundle and debris bed.

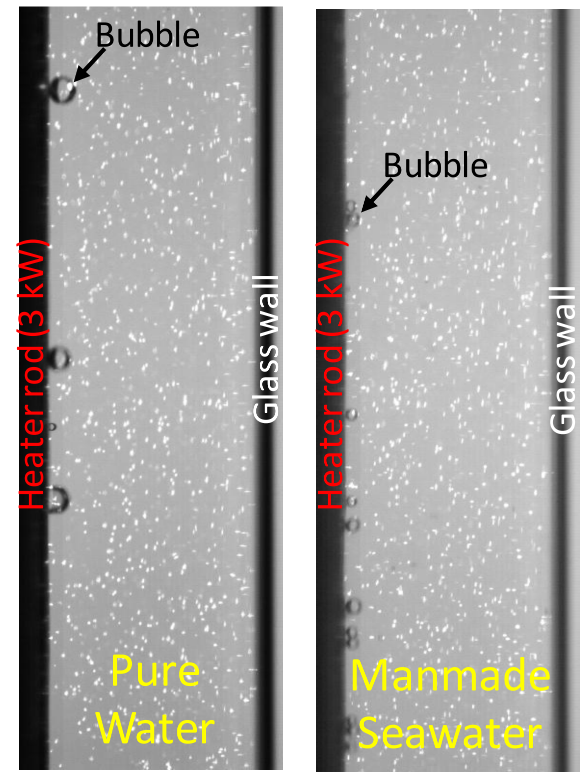


<u>Test Section used in the experiment.</u>
Temperature, pressure difference and velocity distribution were measured. Interface shape of boiling bubbles was also observed.



<u>Effect of salinity on heat transfer coefficient</u>

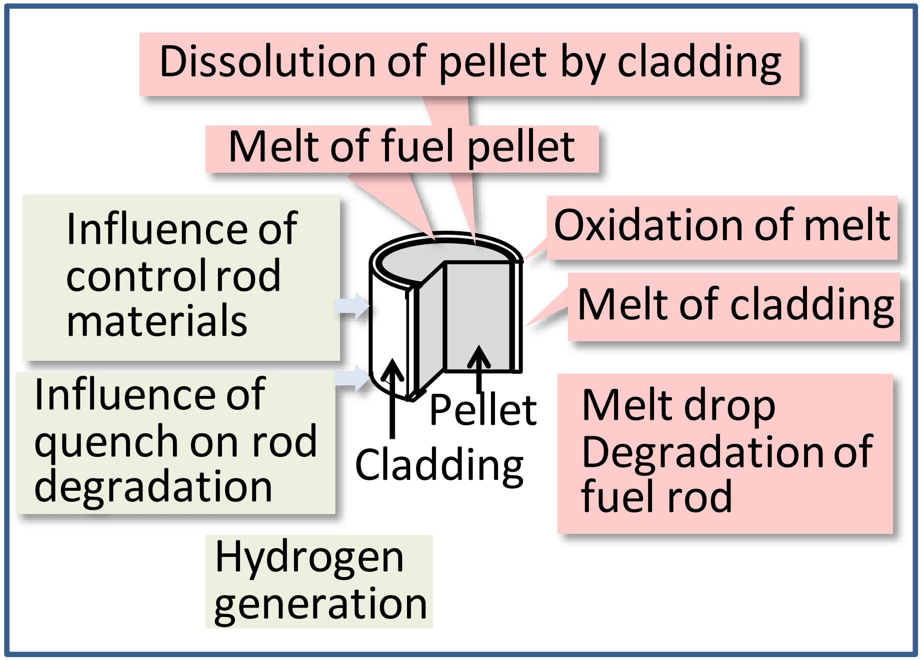
- Heat transfer coefficient of seawater is higher than that of pure water, but difference is negligible if the difference in physical properties of liquids are taken into account in single phase condition.



<u>Effect of seawater on boiling phenomena</u>. Size of Boiling bubbles in manmade seawater is smaller than that in pure water.

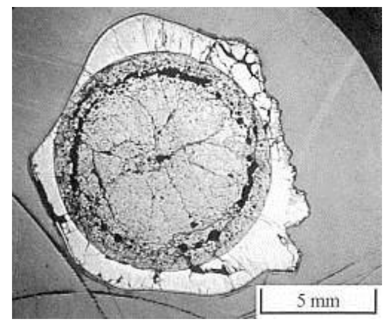
Fuel damage and degradation process

- To obtain knowledge concerning melting and degradation process of fuel rod under LOCA condition



Various phenomena and influential factors in fuel rod degradation during loss of coolant

- Melted down or degraded due to severe oxidation?
- What is the criterion of fuel degradation? *E.g.*
  - Temperature?
  - Oxygen potential in the surroundings of the fuel?



NSRR exp.

QUENCH exp.
NED, 237(22), 2007, 2157

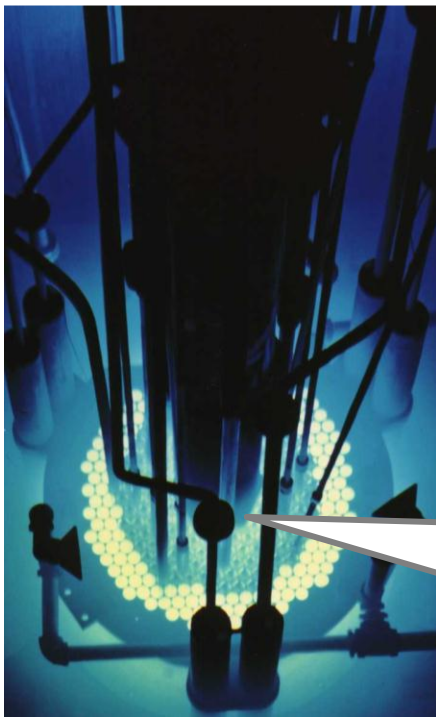
Irradiation test

Reproduce fuel rod degradation in a research reactor

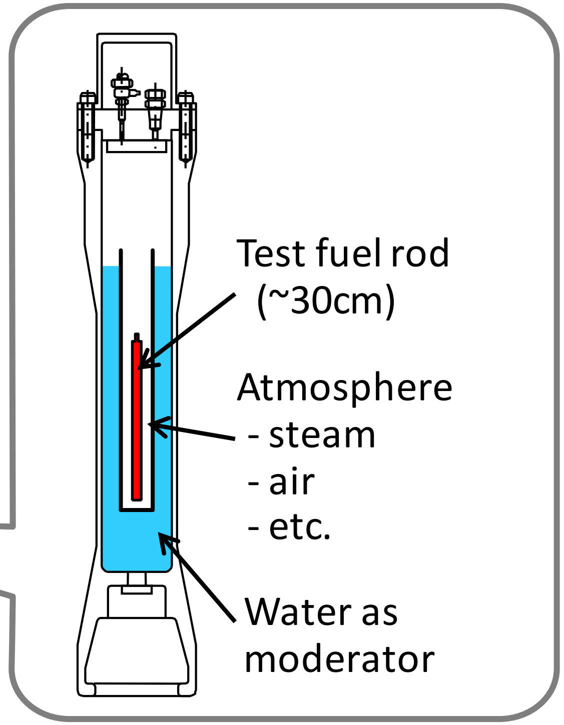
Fuel Melting and Degradation Tests in a Test Reactor

● Fuel melting and degradation experiment using the NSRR in JAEA

✓ Preliminary experiments and the analyses of their results are now being conducted in order to find suitable experimental conditions.



NSRR core at operation



Test capsule
(height: ~1.2 m, ID: 120 mm)

【 Experimental plan 】

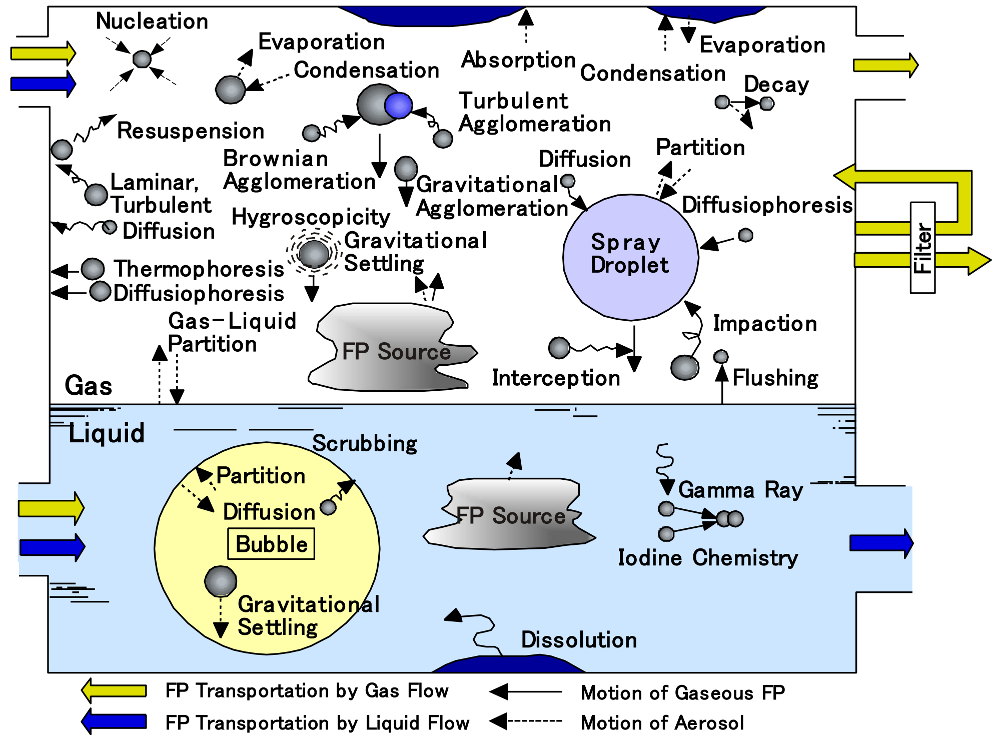
A test fuel rod is heated in gas in order to simulate loss of coolant conditions.
The onset conditions of fuel melting and fuel degradation behavior will be investigated.

【 Outcomes expected 】

This test provides data to be used for verification and accuracy improvement of SA analysis codes.

Evaluation of FP release and transport behavior in the BWR system

- Objective
  Establishment of "database for FP chemical form under SA conditions" (namely, data acquisition and mechanism clarification) towards more accurate evaluation of source term, as well as FP distribution inside reactor
  - Especially for Cs and I, mainly from the viewpoints of public exposure, decay heat,
  - Based on experimentally obtained information, and
  - Focusing on influences of control rod materials of BWR



Various phenomena on fission product release, transport and deposition

Research items for FP-chemistry study in JAEA

Research items	Related important phenomena	Stages in NPP			
		RPV	RCS	PCV	R/B
1. Chemical form of FP	Increase of gaseous iodine fraction by the presence of B_4C (or absence of SIC)	←→	←→	←→	←→
1.1 Aqueous properties in S/C	Change of gas-liquid partition coefficient of iodine by pH change			←→	
1.2 Characteristics of FP deposits in R/B	Variety of contamination in R/B of each unit of 1F-NPP				←→
2. FP deposition and re-vaporization behavior	Change of chemisorption and re-vaporization behaviors of Cs and I by the presence of B_4C (or absence of SIC)		←→	←→	
3. FP release kinetics	Change of FP release kinetics by the presence of B_4C	←→			
4. Fundamental data acquisition of FP compound	—	←···→	←···→	←···→	←···→
Experimental device for the evaluation	TG-DTA-MS	━	━		
	KC-MS	━	━		
	Test device for FP release and transport	━	━	━	
	Test device for FP chemistry		━	━	━

FP deposition and re-vaporization behavior - Cs-chemisorption onto the stainless steel (SS) -

Background

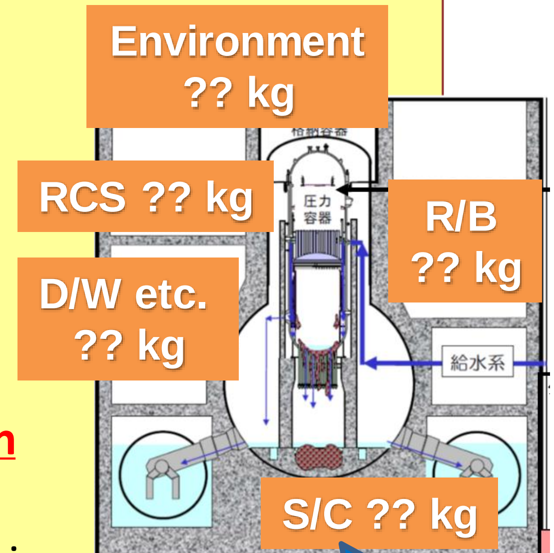
- **Cs-chemisorption onto the stainless steel (SS)**
  - An important phenomena for the evaluation of source term
  - Key issue toward 1F-decomissioning work: a possible long-term radiation and heat sources when fixed onto structural materials
- Research subject:
  - **No knowledge for BWR-SA conditions, B_4C including system**
  - Uncertainty in the model: **fundamental mechanism should be known**

Objective

Fundamental study on the Cs-chemisorption behavior under BWR-SA conditions

⇒To improve the Cs-chemisorption model in the SA-analysis code

- Basic experiment using non-radioactive surrogate materials
- Thermodynamic/ab-initio analysis for the chemisorption process



- Cs distribution is essential for dose evaluation during decommissioning work.
- Cs deposition to the structures a key phenomenon for Cs distribution.

The preliminary evaluation are presented here:

- Assumption and visualization of the Cs-chemisorption process model based on the review of previous related works
- Validation of the assumed model and prediction of effects of boron on Cs-chemisorption behavior with the aid of a chemical equilibrium calculation

Cs-chemisorption behavior
- Assumption and visualization of process -

➢ Summary of review of previous experimental works on Cs-chemisorption

	Simulant FP[1,2]*	
Temperature condition of heating test	Relatively low (below 873 K)	Relatively high (beyond 1073 K)
Type of Cs compound (Solubility for water)	Cs-Cr-O (Soluble)	Cs-Si-O (Insoluble)

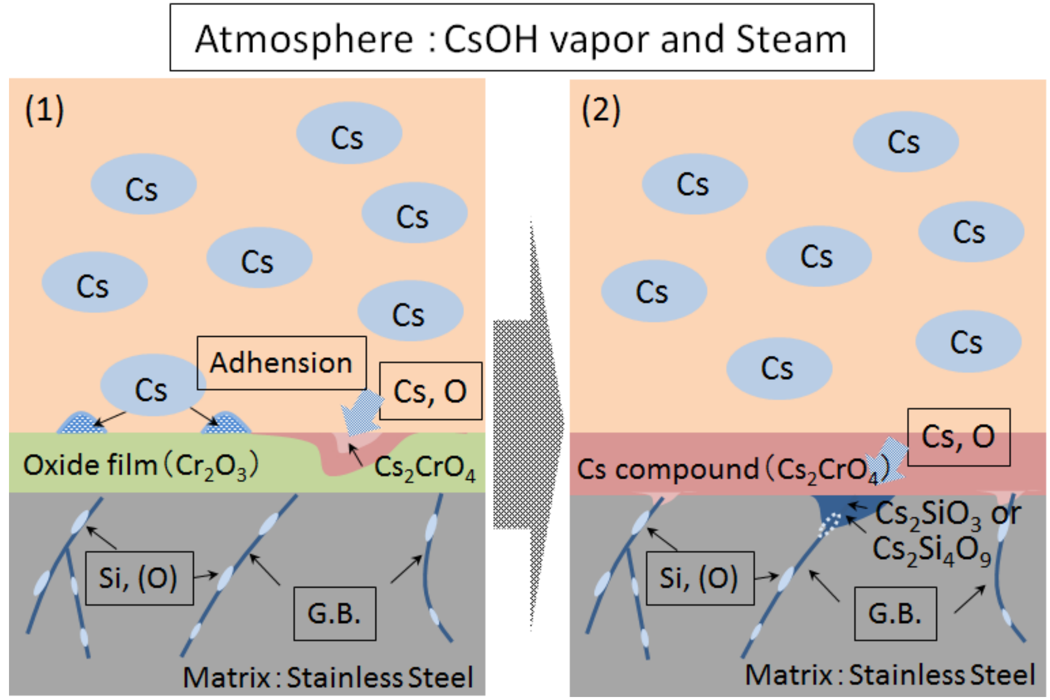
*Cesium hydroxide (CsOH)

Reference

[1] A. M. Beard et al, Reactor Safety Programme 1985-1987 Final Report, EUR12844EN

[2] R. M. Elrick, NUREG/CR-3497

➢ Assumed model of Cs-chemisorption process for unused SS



(1) Cs-chemisorption onto Cr_2O_3 layer formed on unused SS

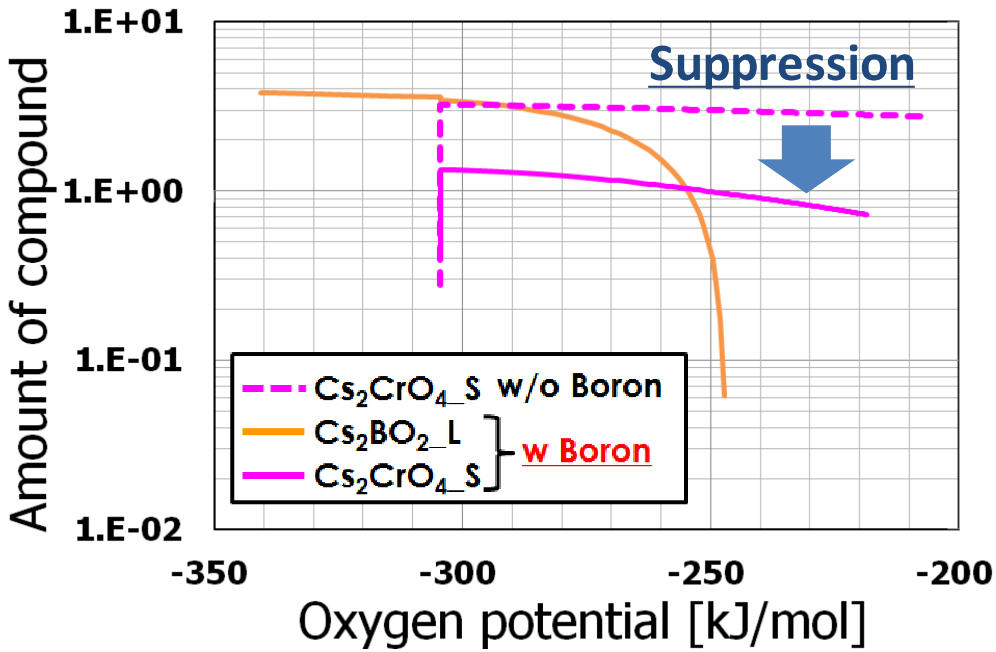
(2) Interaction of chemisorbed Cs with Si included unused SS

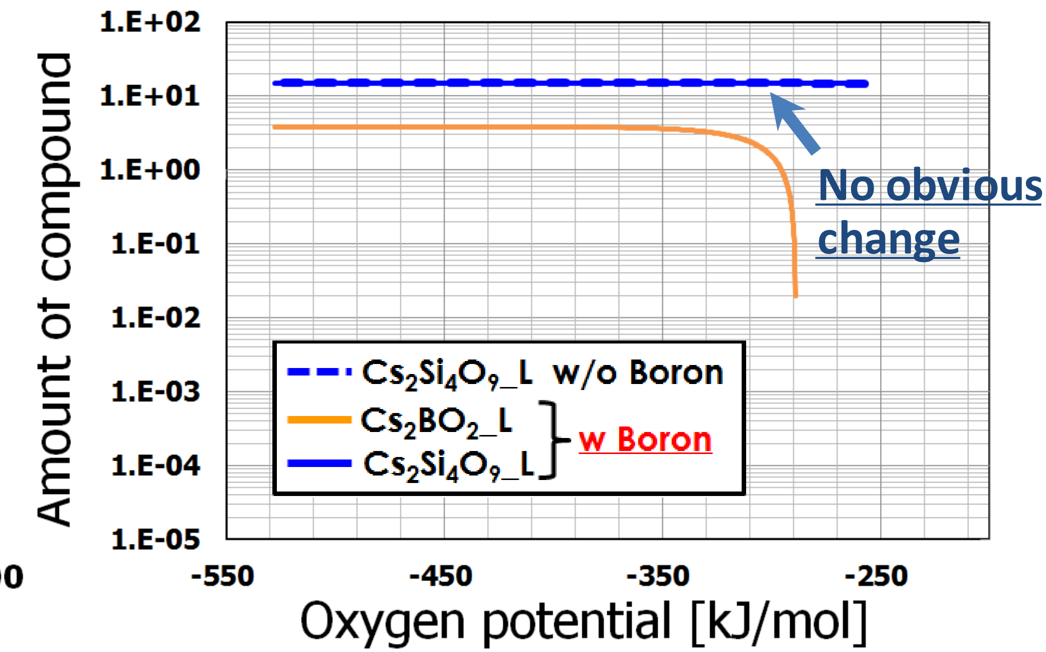
✓ Model of Cs-chemisorption processes for unused SS was assumed.

✓ A chemical equilibrium calculations showed a consistent trend with the previous experimental results.

Cs-chemisorption behavior - Prediction of B effects -

➢ **Effect of boron on Cs-chemisorption onto the unused SS surface (1273K, 0.1MPa)**





(1) Cs reacts with Cr_2O_3 on the unused SS and resultantly formed Cs-Cr-O phase. The formation of Cs-Cr-O phase was highly possible to be suppressed under B including environment.

(2) Chemisorbed Cs interacts with silicon (Si) included in unused SS and formed Cs-Si-O phase. There is no effect of B on the formation of Cs-Si-O phase.

Implication of the possibility that existence of B affect the Cs-chemisorption onto the SS surface.

Behavior of structural materials and pressure vessel

Objective

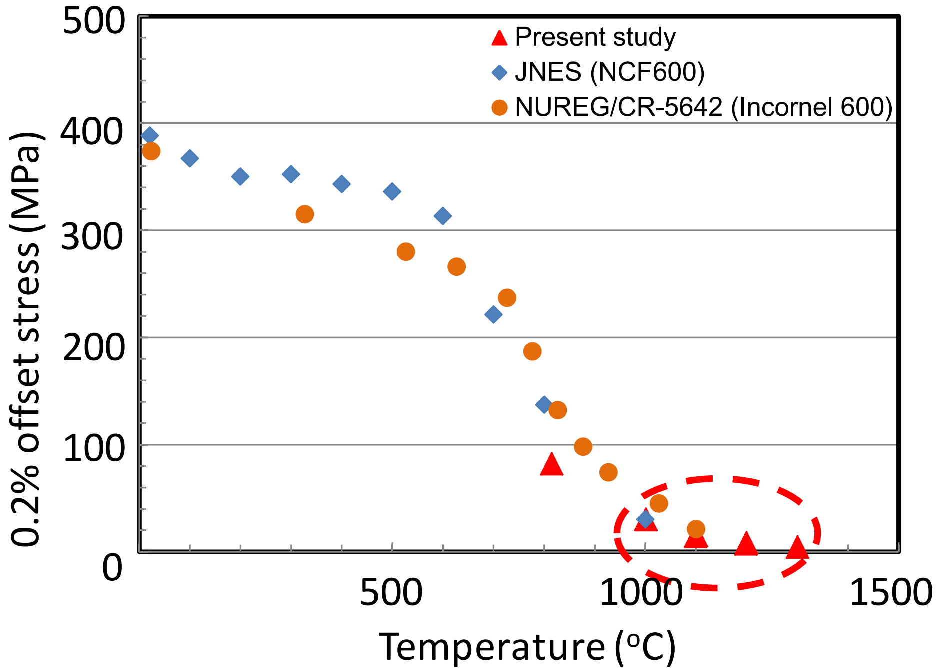
- To predict time and location of RPV lower head rupture of BWRs more precisely, which are valuable to the analysis of fuel debris distribution in the Fukushima Daiichi NPP, investigations including material data acquisition and computer code analysis are conducted.

Research items

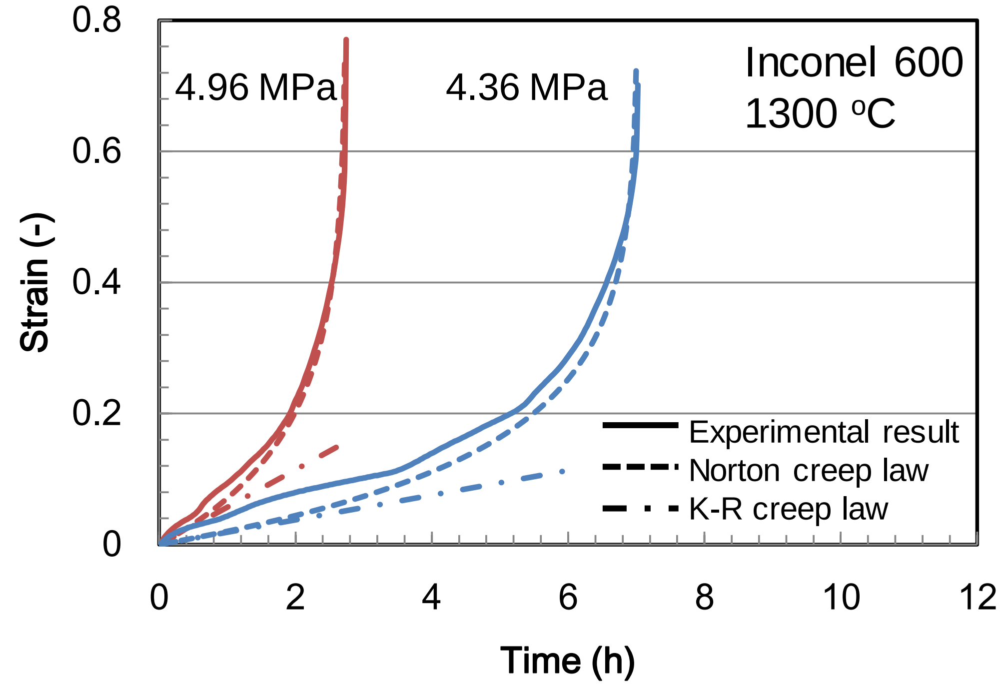
- Re-evaluation and expansion of materials data such as mechanical properties, creep deformation/rupture properties is made for low alloy steel, Ni-based alloy and stainless steels based on past research activities.
- Applicability evaluation of the FEM modeling using uni-axial material data for multi-axial deformation analysis.
- To investigate the inhomogeneous temperature and stress distribution by geometrical complex of BWR lower head, the detailed 3D model of RPV lower head with control rod guide tubes (CRGTs) and shroud supports are constructed and the 3D coupled thermal hydraulic-structural simulation are performed using ANSYS Fluent/Mechanical and FINAS CFD/STAR finite element codes.

Behaviors of structural materials and pressure vessel

(1) Expansion of materials database such as mechanical properties, creep deformation / rupture properties



Tensile properties (0.2% offset stress)



Investigation of creep constitutive law and creep damage criterions

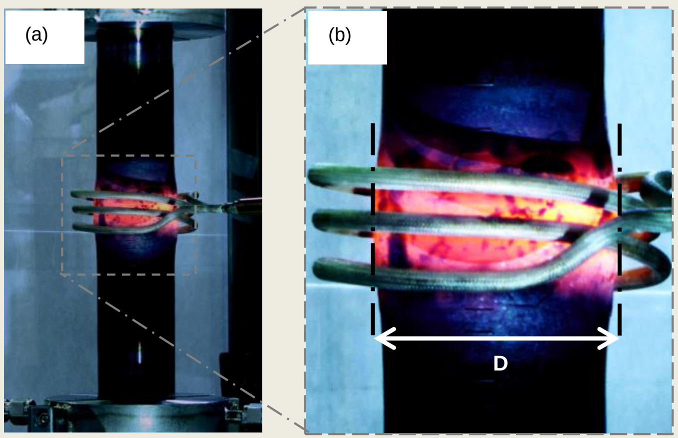
- Materials database in high temperature region where there is no data in past literatures were expanded.
- To predict failure behaviors of lower head due to creep deformation, creep constitutive law creep damage criterions were investigated through the comparison in rupture time between analytical predictions and experiments.

Behaviors of structural materials and pressure vessel

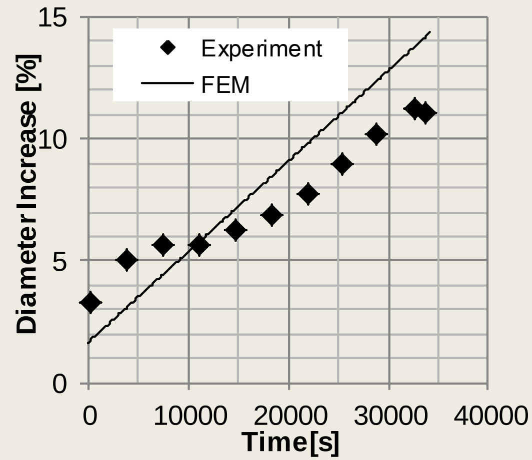
(2) Applicability evaluation of the FEM using uni-axial material data for multi-axial deformation analysis

- Internal pressure creep test
- FEM analysis

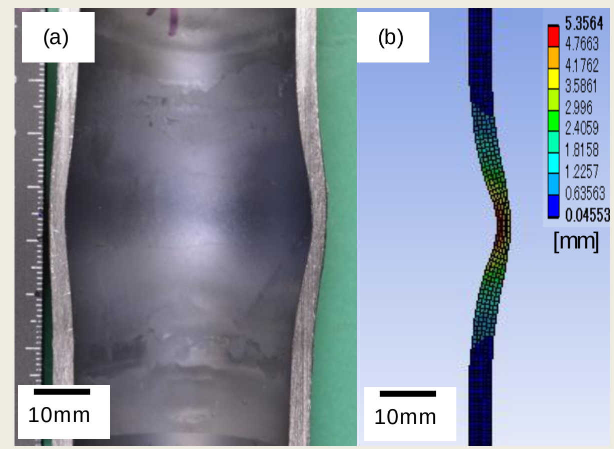
The results were compared to evaluate the reproductivity



(1) A533B specimen during the internal pressure creep test.



(2) Diameter increase of the specimen (experiment & FEM)



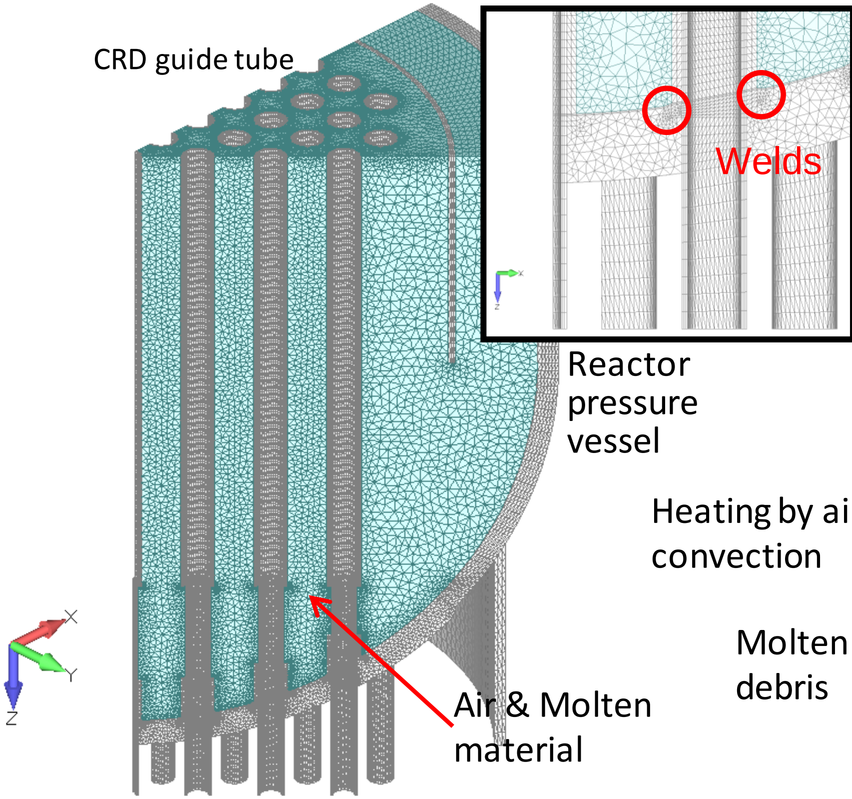
(3) Cross-section of the tested specimen and its modeling.

Typical results of experiments and analyses
(simulated RPV internal pressure : 2.5MPa, the highest temperature : 900C)

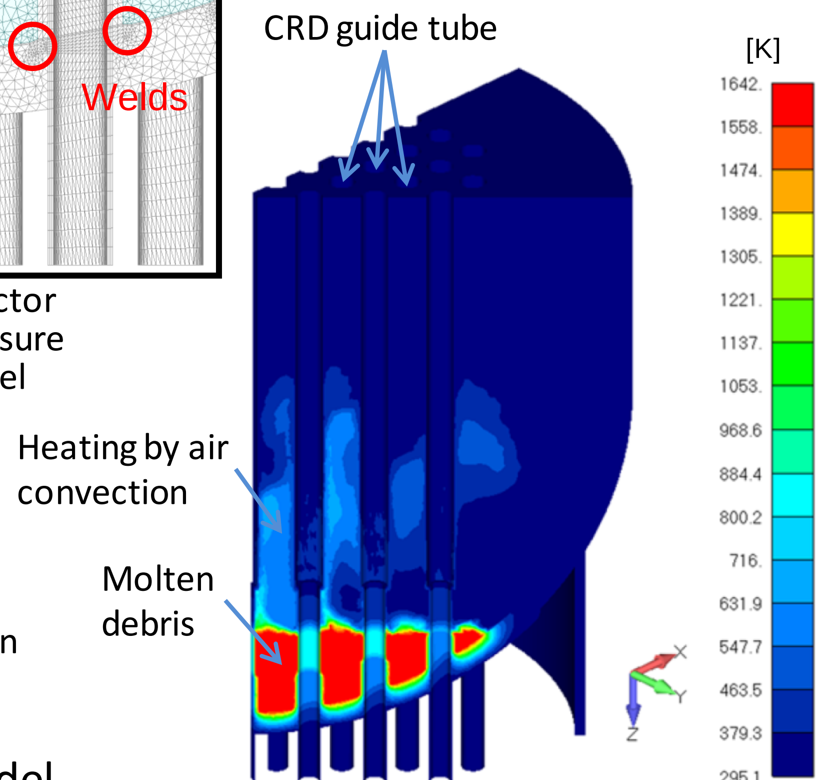
- The modeling well reproduced multi-axial deformation of the RPV material (A5333B steel) during the internal creep test if experimental error was taken in account.
- The analytical model using uni-axial data for multi-axial deformation analysis was thought applicable.

Behaviors of structural materials and pressure vessel

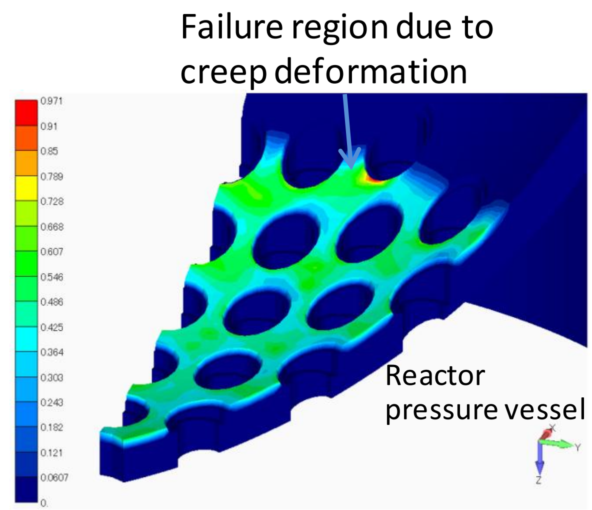
(3) Coupled thermal hydraulic-structural simulation by FINAS CFD/STAR



General view of analysis model



Temperature distribution



Creep damage evaluation based on LMP criterion

- Simulation model considering the behaviors of air convection and molten materials is established for obtaining an accurate prediction results.
- Failure region of lower head can be predicted on the basis of several damage criterions including melt-through, creep damage mechanisms, Larson-Miller parameter, etc.

Summary

- JAEA conducts various R&D to obtain the information for decommissioning of the Fukushima Daiichi NPP as well as developing/improving SA simulation codes and evaluating effectiveness of accident mitigation measures.
- The R&D covers thermal hydraulic behavior, fuel rod degradation process, failure behavior of pressure vessel, molten materials relocation in the lower plenum region of BWR during accidents, fuel debris characterization and computer code development.
- The experiments and analyses are being progressed and technically interesting results are being obtained.