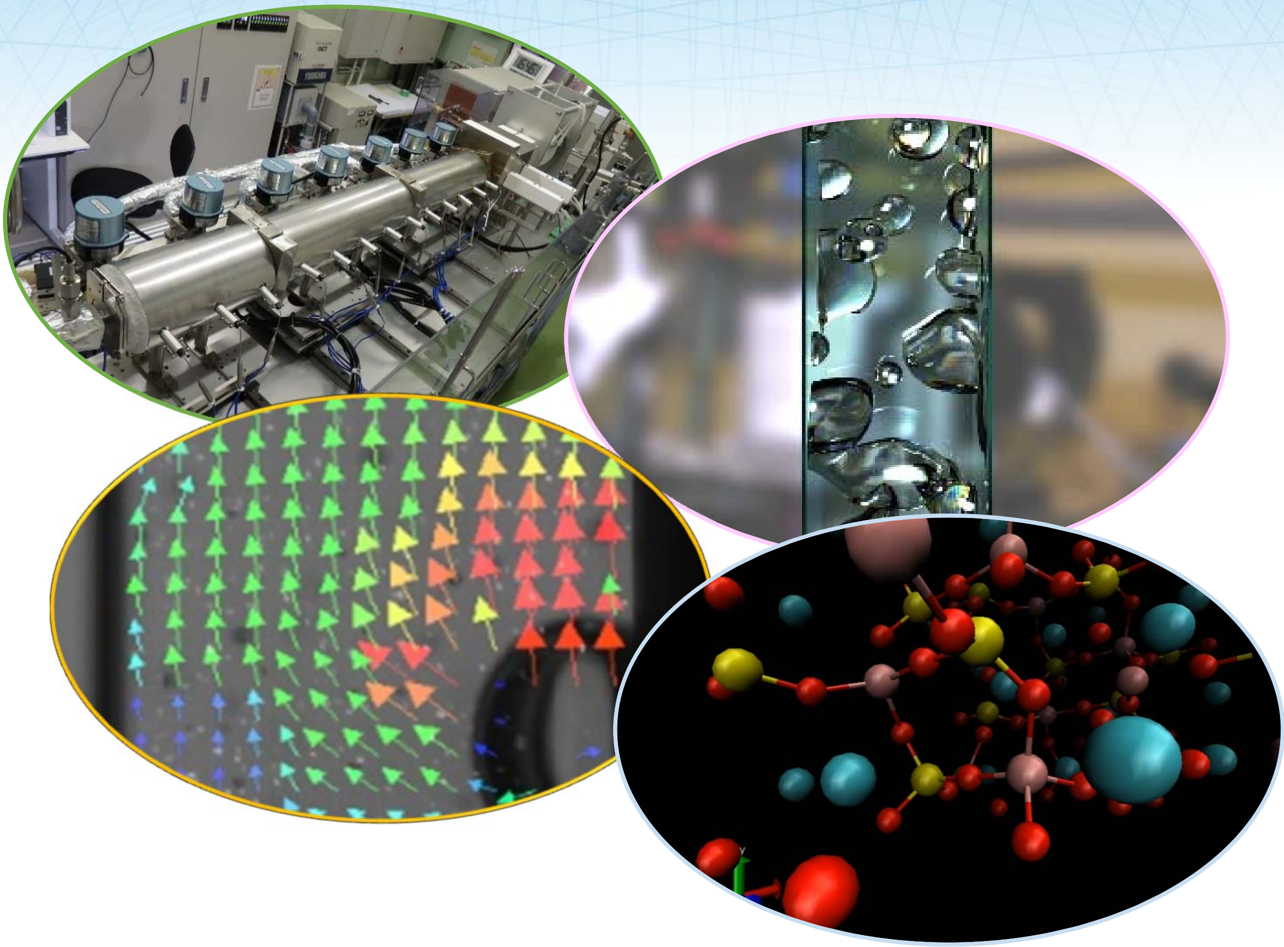
Fundamental Studies toward Safety Enhancement of LWR



Japan Atomic Energy Agency
Nuclear Science and Engineering Center
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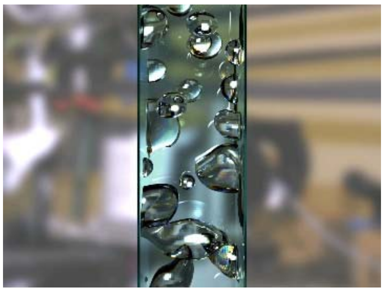
Development Group for Thermal-Hydraulics Technology

TPFIT Two-phase Flow Simulation Code



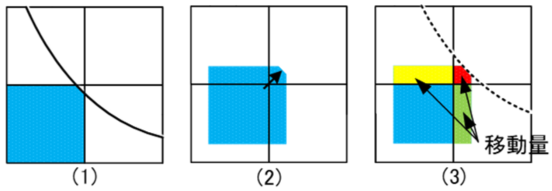
Two-Phase Flow with Interface Tracking

TPFIT is developed for two-phase flow in reactor systems, and can be applicable to CFD simulation including two-phase flow in any systems. TPFIT adopts the advanced interface tracking method developed in JAEA, and can be applied to the detail numerical simulation of two-phase flow including movement and deformation of the interface.

■ Features

Advanced interface tracking method

Existing numerical simulation codes to evaluate the interface shape and movement directly has a problem of numerical diffusion. In the case of deformation and/or speed of the interface is large, numerical diffusion will become a serious problem, and to avoid this problem, a lot of existing numerical simulation codes adopt non-physical treatments around the interface. The advanced interface tracking method developed in JAEA can suppress numerical diffusion without non-physical treatments by using semi-Lagrangian treatment of fluid body. The advanced interface tracking method is also applicable to compressible two-phase flow, then, TPFIT can treat high-speed two-phase flow with high accuracy.

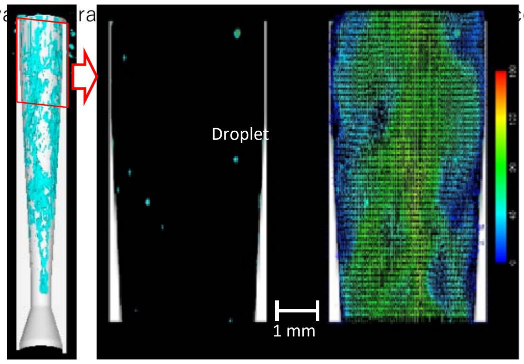


Outline of advanced interface tracking method

1. Approximate fluid body shape in the numeric cell by polygon
2. Transport polygon by accordance with the velocity field
3. By distributing polygon volume to overlapping cells, eva ... ra ... cells.

Applicable to high speed two-phase flow

The advanced interface tracking method is developed for compressible two-phase flow. Then, TPFIT can be applicable to high-speed two-phase flow, such as two-phase flow in Venturi scrubber, whose maximum velocity is more than 150 m/s. In the snapshot of numerical results, small droplets in high velocity gas flow are simulated directly.



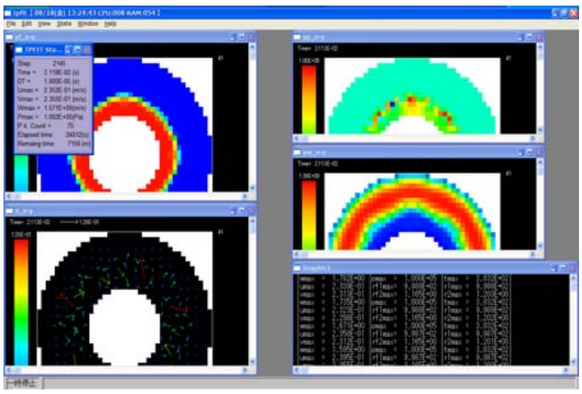
High speed two-phase flow in Venturi scrubber

Massively parallel computing

To perform the thermal-hydraulics simulation of wide ranged scale structure, the simulation code has to be sufficient performance on massively parallel computers. By adopting MPI and OpenMP parallelization technology, TPFIT can use sufficiently massively parallel computers with more than 4,000 cores and billion numerical cells.

Real time visualization function

At first trial of numerical simulation, we want to see numerical results as fast as possible. The real time visualization function of numerical results is implemented in TPFIT. This function is useful for not only beginners, but also professional of CFD. Its function is only available on windows PCs.



Real time visualization of numerical results

■ Specifications

Languages	Fortran 90	Surface tension model	CSF model
Parallelization	MPI, OpenMP	Applicable Fluid	Water, steam, air, any fluid (based on user input data)
Confirmed compiler software	Intel® Fortran, PGI® Fortran	Interface tracking	Advanced interface tracking method
Convective term discretization	CIP method		

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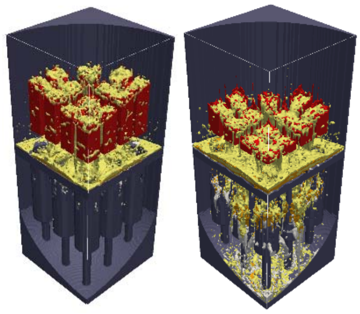
JUPITER Multiphase/Multicomponent Thermal Hydraulics Code



JAEA **U**tility **P**rogram for **I**nterdisciplinary **T**hermal hydraulics **E**ngineering and **R**esearch

JUPITER is developed for melting or solidification simulation for reactor core at severe accident, and can be applicable to CFD simulation including any number of fluid or solid component and various kinds of multiphase/multicomponent flow simulation. JUPITER adopts state-of-the-art computational technique, and massively parallel computing is available and then detail simulation regarding multiphase flow will be realized, which is never seen before.

■ Features

Massively parallel computing

To perform the thermal-hydraulics simulation of wide ranged scale structure, the simulation code has to be sufficient performance on massively parallel computers. By adopting MPI and OpenMP parallelization technology, JUPITER sufficiently can use the K-computer, that is one of the fastest supercomputers in Japan (see Fig.1).

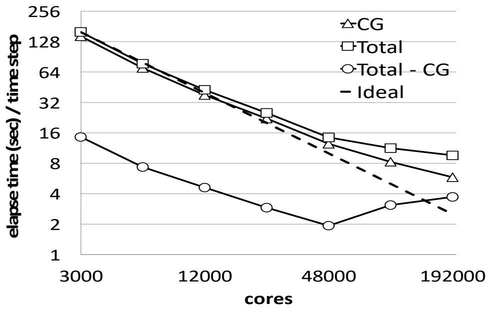


Fig 1. Strong scaling

Simple and robust interface capturing method

In order to simulate accurately and efficiently in massively parallel computers, numerical algorithms must be simple. In JUPITER, interface capturing scheme, which can capture interfaces easily and accurately, is adopted. This scheme also contribute the achievement of the performance, such as Fig. 1.

Melting and solidification for any number of material components

JUPITER can mechanistically estimate behaviors of melt relocation and/or solidification of several materials. Then, by using JUPITER, detailed component distribution inside fuel debris can be evaluated. For example, corium spreading and accumulating behavior inside the pedestal can be simulated more accurate than before. (See Fig. 2).

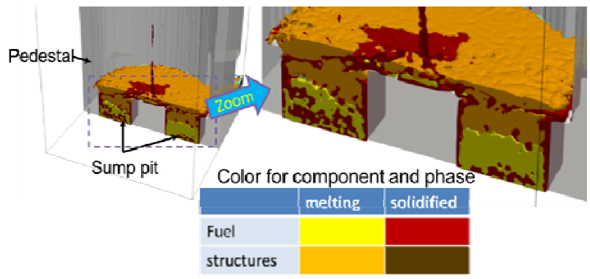


Fig.2 Spreading behavior of corium inside the pedestal

Application to other fluid flow phenomena

JUPITER can also apply not only melt relocation behaviors in severe accidents, but also other fluid flow problems such as the thermal-hydraulics around ADS beam window and milk crown analysis (see Fig. 3). Of course, JUPITER can be applied to any other multiphase flow problems.

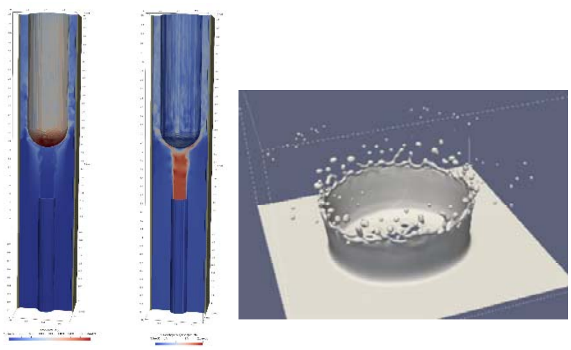


Fig.3 Thermal-hydraulics around the ADS beam window (left), milk crown (right)

Ready for big data visualization

Simulation data outputted by massively parallel computation become quite big. An efficient and powerful volume rendering visualization software named PBVR (Particle Based Volume Rendering) was implemented to JUPITER and then the visualization for big data has been easily performed.

■ Specifications

Language	C	Curvature/Surface tension force model	CLSVOF/CSF model
Parallelization	MPI, OpenMP	Phase change model	Temperature recovery method
Compiler	GNU C, Intel® C, PGI® C	Applicable Fluid	Water, air, molten material, etc. given by input data
Convective term discretization	WENO, Third order upwind, etc.	Surface capturing scheme	THINC, THINC/WLIC method
Time integration	Third order accuracy TVD Runge-Kutta method	SGS eddy viscosity model (LES)	Dynamic Smagorinsky, Coherent Structure, ILES

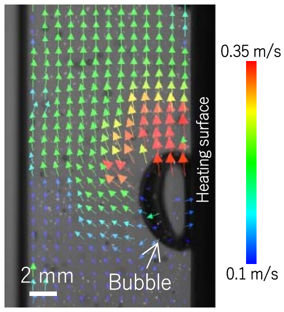
TH Experiment and Measurement Technique for TH Phenomena

Our research group contributes to predictive evaluation of the situation in core at severe accidents and performance evaluation of related equipment by supercomputer simulations such as TPFIT or JUPITER. In addition to developing these prediction evaluation methods, our research group performs various experiments to validate those simulations and construct basic physical models.

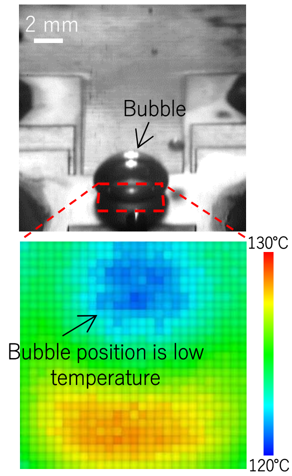
■ CONTENTS:

Seawater heat transfer experiment

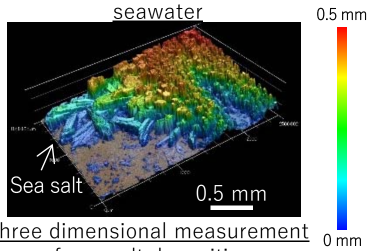
In the Fukushima Daiichi Nuclear Power Plant (1F) accident, seawater was injected into the reactor core instead of pure water. The injection of seawater into the core was unprecedented, the influence of seawater on the cooling performance is not sufficiently understood. TH experiments with seawater using test sections simulated nuclear reactor cores and thin film heat transfer surfaces were performed to understand the influence of seawater on the cooling performance including boiling phenomena.



Measurement of boiling behavior and velocity distribution in seawater



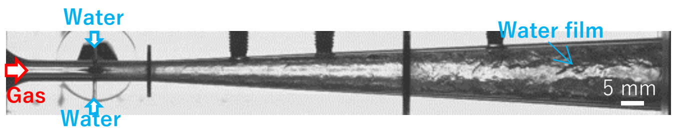
Measurement of Boiling Behavior and Heat Transfer Surface Temperature Distribution



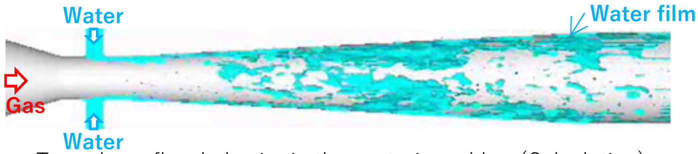
Three dimensional measurement of sea salt deposition

Two-phase flow behavior of Venturi scrubber

In order to prevent the damage of the containment vessel at severe accident, high-pressure steam in the containment vessel must be released to the outside (so called venting). At this time, fission products (FPs) contained in steam must be removed to suppress spread out of FPs to the environment. Therefore, light water reactors in Japan are mandatory to install facilities (filtered venting systems) that removes FPs at the venting. Although the overall performance (decontamination factor) of filtered venting systems was confirmed, the detailed decontamination mechanism, that is required to improve performance and consider the most suitable operation, is not clear. To understand the detailed decontamination mechanism, both experimental and numerical works has been performed.



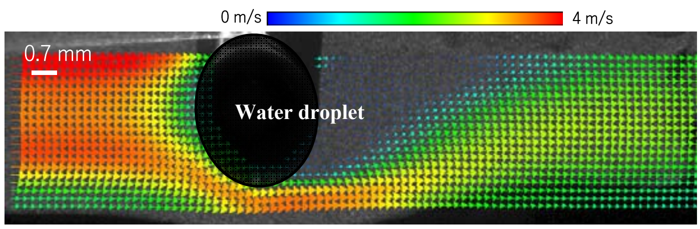
Two-phase flow behavior in the venturi scrubber (Experiment)



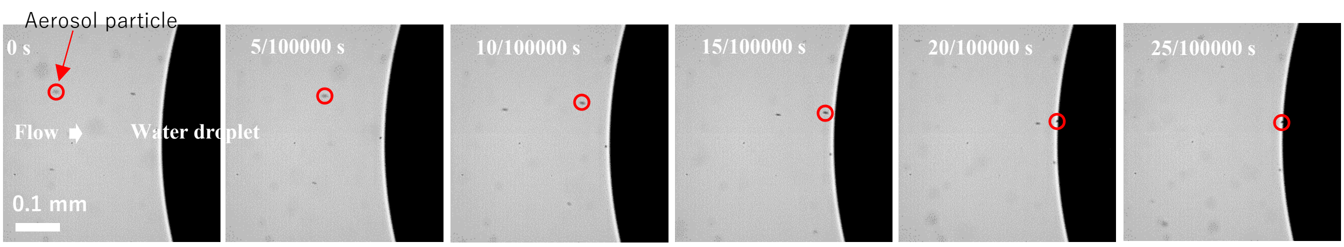
Two-phase flow behavior in the venturi scrubber (Calculation)

Elucidation of the removal mechanism of radioactive aerosol by water droplets

As one of physical mechanisms of removal radioactive aerosol by the above Venturi scrubber, an inertial impaction of aerosol particles to a liquid droplet is considered. However, since the aerosol particles has very small diameter of about 1/1000 mm and moves at a very high speed, the capturing behavior of aerosol particles by droplets has not been observed directly. The direct observation method of the capturing behavior was developed, and observed the capturing behavior of the particles to discuss the removal mechanisms.



Velocity distribution of aerosol particles around the water droplet



Continuous images of capturing behavior of an aerosol particle near the droplet surface

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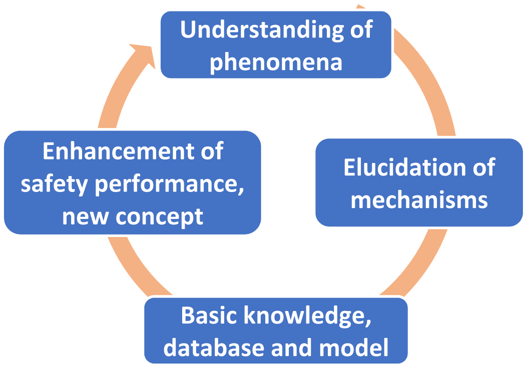
■ Activity of Development Group for LWR Advanced technology

Our final goal is to contribute to improved/enhanced safety performance of LWR. For that, we have proceeded a fundamental research on LWR advanced technology to provide fundamental knowledge for improvement of scientific rationality for regulation standard, safety margin enlargement for design, and decommissioning and dismantling of Fukushima Dai-ichi Nuclear Power Station. We have investigated the fission product, fuel/material behaviors under conditions from normal operation to severe accident for the construction of their database including physical/chemical model towards the improvement of evaluation method such as a severe accident analysis code.



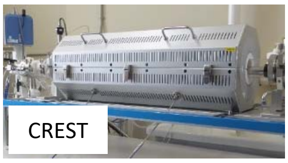
■ Overview of fundamental research on fission product behavior

Our group has been proceeding a fundamental research on FP behavior during transport from the reactor core to the Primary Containment Vessel (PCV) and Reactor Building (RB) under a LWR severe accident. We have conducted systematic experiments and analyses in order to construct "ECUME", a database for FP chemistry and each elemental process. ECUME will become a fundamental basis for improvement of source term evaluation technology.

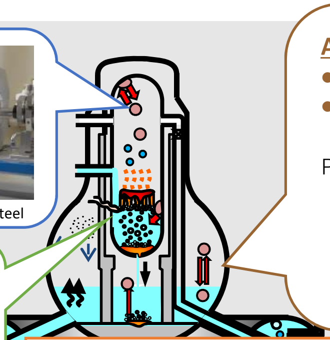
Severe accident phenomena covered by ECUME

Chemical reaction with structural materials

- Cs chemisorption onto SS*
- Chemical reaction of control rod material boron (B) vapor species with Cs and I deposit

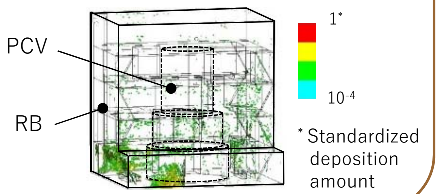


*SS: stainless steel



Aerosol behavior

- Interaction with droplets
- Deposition in RB (CFD analysis)



*Standardized deposition amount

Release behavior from core

- Release behavior of B and molybdenum
- Release behavior of ruthenium and strontium

Reproduce of phenomena during accident

Release and transport behavior

- Release from core, chemical reaction, aerosol formation/ growth/deposition

Each elemental model, chemical reaction kinetics, and thermodynamic properties which are essential for evaluating FP behavior will be included incorporated in the ECUME.

■ Research & development of Accident Tolerant Fuels (ATFs) for practical use *

For the reduction of the accident risk and the improvement of further safety of the existing LWRs, ATFs with higher reliability are being developed with a focus on enhanced tolerance to high temperature steam under severe accidents in the collaboration among industry-university-government. The technical basis to introduce ATFs into LWRs will be established and it's reflected to the commercial use by the industries.

Main working subjects

- Refinement of technology readiness level and clarification of attribute guide
- Basic data acquisition and detail analysis for fuel design
- Evaluation of influence of the ATFs installation
- Choice of ATF type with high priority
- Consideration of the standard necessary for practical use

Whole process of ATFs project

2015 2018 2021 2022 → **Commercial use in LWRs**

Proof-of-principle stage ⇨ Proof-of-engineering stage ⇨ Practical use development stage

Proof-of-principle stage:
- Base data acquisition and detail analysis for fuel design
- Fuel pin irradiation test in a research reactor
- Consideration of standards and criteria

Proof-of-engineering stage:
- Establishment of the design parameter
- Production of prototype fuel and component
- Lead test assembly irradiation

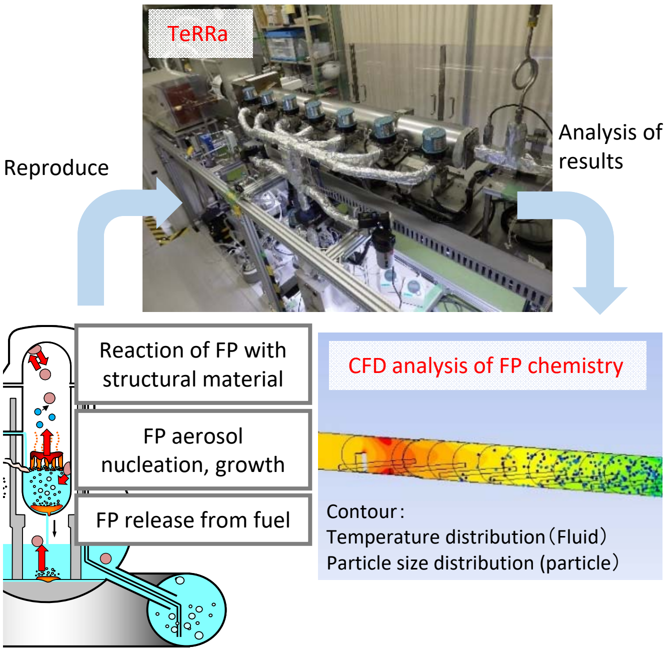
Practical use development stage:
- Irradiation test of production units for industrial scale
- Safety assessments and licensing review by the regulation authority

SiC fiber /SiC matrix composite
(Trial product of fuel cladding tube)

*This study is the result of 'Development of Technical Basis for Introducing Advanced Fuels Contributing to Safety Improvement of Current Light Water Reactors' carried out under the Project on Development of Technical Basis for Safety Improvement at Nuclear Power Plants by Ministry of Economy, Trade and Industry(METI) of Japan.

TeRRa (**Te**st bench for FP **R**elease and t**Ra**nsoprt)

■ What kind of reaction do fission products experience in a reactor ?



Fission product (FP) chemistry in a reactor

Chemistry of FPs, in particular Cs and I, is of importance for the estimation of radiation source distribution in Fukushima Dai-ichi Nuclear Power Station and their release behavior into environment. This is because FP chemistry dominates the FP behavior during release from fuel and transport in a reactor.

The FP chemistry can be affected by various factors such as aerosol formation and growth. This complexity makes it difficult to precisely evaluate the FP chemistry during transport in a reactor. Therefore, for the construction of FP chemistry ECUME database, which is our final goal, we employed the reproductive experiments for a series of FP release and transport, together with the analyses of the results considering chemical kinetics and aerosol behavior. Thus, we developed reproductive test set-up named TeRRa (**Te**st bench for FP **R**elease and t**Ra**nsoprt).

■ Overview of reproductive test of FP release and transport using TeRRa

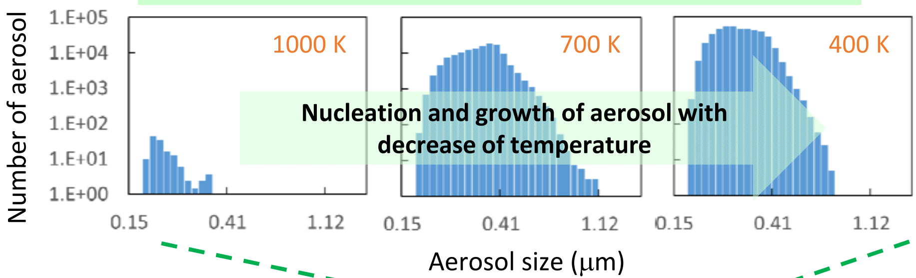
Reproduction of FP release and transport

TeRRa can simulate temperature profile and atmosphere during FP release and transport in a severe accident. In particular, TeRRa can achieve the maximum temperature of 2500 K, and can reproduce not only transport in a reactor but also release from fuel.

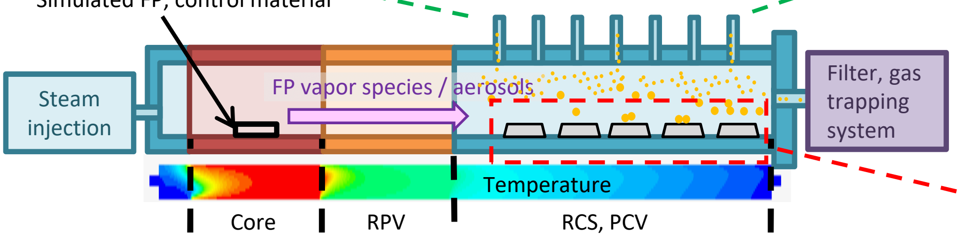
Variety of analysis technique for FP chemistry

TeRRa can provide various data for FP release and transport behavior. TeRRa can extract airborne aerosols during transport. These extracted airborne aerosols are subjected to the on-line measurement of size distribution and off-line characterization.

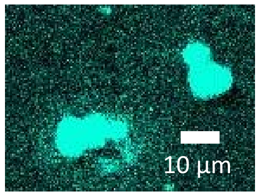
On-line measurement of size distribution of airborne aerosols



Specification of furnace	
Max. temperature	2500 K
Thermal gradient tube	1000 – 400 K
Atmosphere Flow rate Dew point	 Max. 5 L/min Max. 80°C



Off-line analysis of deposits
(e.g. SEM-EDS)



Examples of achievements from TeRRa test

The followings were newly-obtained from the TeRRa test results

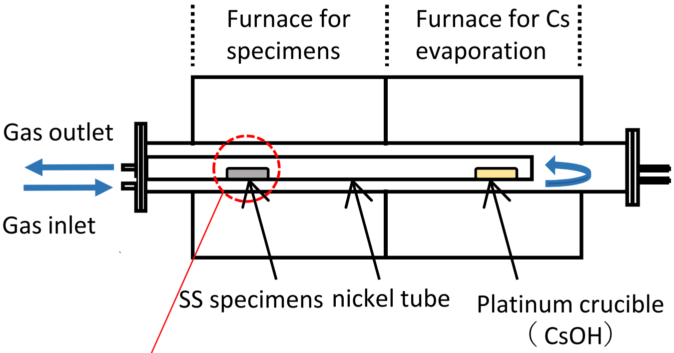
① B vapor species would react with SS above 1000 K, which could affect the chemical reaction of Cs with SS.

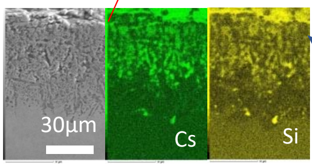
② B vapor species would enhance the formation of gaseous iodine by the reaction with Cs and I deposit

Chemisorption test apparatus onto structural material surface

CREST (Chemical REaction with STeel)

■ How does Cs exist in Fukushima reactors?





Formation of Cs-Si-(Fe)-O compounds by Cs chemisorption

Cs and Si distribution on the cross section of Cs chemisorbed SS specimen

Elucidating Cs adhesion mechanism onto structural materials

Elucidating the Cs adhesion mechanism onto the reactor structural materials is one of the important challenges for the decommissioning of Fukushima-Daiichi Nuclear Power Station because radioactive Cs can be the main radiation source. Relatively high temperature could have been attained at the upper part of RPV in the accident and Cs could react chemically (chemisorption) with the SS at such high temperature. We are studying the Cs chemisorption behavior from the aspect of its mechanism by using the CREST apparatus which can reproduce the Cs chemisorption onto the SS surface.

Examples of achievements from CREST test

① It was found that Cs chemisorption amount was influenced by not only temperature (previously known) but also the other chemical conditions, such as atmosphere, Si content in SS, CsOH concentration in the gas phase and so on.

② The influence of such chemical conditions was modelized in order to improve Cs distribution analysis technology in severe accident (SA) analysis code. The accuracy of our model was improved by 1 order of magnitude compared with the existing.

Development of Accident Tolerant Fuels (ATFs)

■ For further improved safety of Light Water Reactor *

ATFs candidates

1. FeCrAl-ODS stainless steel
- ATF cladding for BWR

2. SiC/SiC composite
- ATF cladding and channel box for BWR
- ATF cladding for PWR

R&D organization for practical use

FeCrAl-ODS Group
- Nippon Nuclear Fuel Development CO., Ltd.
- Hitachi-GE Nuclear Energy, Ltd.
- Global Nuclear Fuel Japan Co, Ltd.
- Hokkaido University
- Kyoto University
- Waseda University
- Japan Atomic Energy Agency

SiC/SiC composite Group
- Toshiba Energy Systems & Solutions CO.(BWR)
- Hitachi-GE Nuclear Energy, Ltd.(BWR)
- Mitsubishi Nuclear Fuel Co., Ltd. (PWR)
- Mitsubishi Heavy Industries, Ltd. (PWR)
- Quantum and Radiological Science and Technology
- Japan Atomic Energy Agency

Trial products of fuel cladding tube and channel box

9x9 type

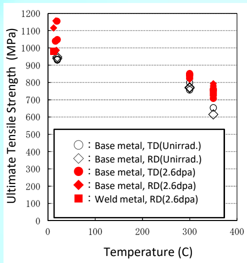
10x10 type

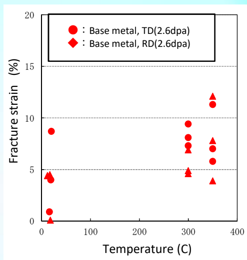
Fuel cladding tube of FeCrAl-ODS stainless steel (Length:>1m, Thickness:0.35mm)

Mini channel box of SiC/SiC composite

＊Press Release by Toshiba, 2014.7.3

Performance at severe accident
① FeCrAl-ODS

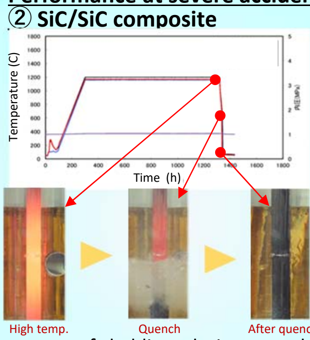




Post irradiation examination results (2.6dpa) of the base and the weld materials, temperature dependency of ultimate tensile strength (left) and fracture strain(right)

- The UTS was increased by irradiation. There is little difference between UTS of the base and the weld materials by irradiation.
- The irradiated materials maintained about 5 % of fracture strain at the temperature of more than 300 ℃.

Performance at severe accident
② SiC/SiC composite



Appearance of claddings during quenching test

- The SiC/SiC composite didn't collapse even if it quenched from more than 1200 ℃.
- The SiC/SiC composite maintained the shape at the simulated condition of LOCA.

*This study is the result of ‘Development of Technical Basis for Introducing Advanced Fuels Contributing to Safety Improvement of Current Light Water Reactors’ carried out under the Project on Development of Technical Basis for Safety Improvement at Nuclear Power Plants by Ministry of Economy, Trade and Industry(METI) of Japan.

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