Basic Technology Subjects

Nuclear Technology Research Laboratory

Brief Overview

The Nuclear Technology Research Laboratory aims at positively contributing to the solving/alleviation of energy and global environmental problems by means of developing nuclear technologies, including base technologies to support the safety and stable operation of LWRs as well as the recovery from the Fukushima Daiichi nuclear power plant accident, so that the use of nuclear energy may be accepted by society in a positive manner.

Achievements by Research Theme

Nuclear Reactor Systems Safety

Technologies to maintain and improve the reliability of reactor safety systems, along with the technologies of accident prevention and mitigation at severe accidents have become important to the enhancement of light water reactor safety. We aim to develop those fundamental technologies in connection with the field of thermal-hydraulics and risk assessment.

- As a collaborative study with USNRC, we have been enhancing the nuclear system analysis code, TRACE, which is developed by USNRC and used for worldwide regulatory research. The TRACE code was validated against our experiments which simulated transient events with a reduction in power and coolant flow, and was found to be capable of reproducing the flow redistribution between channels with different powers. The results conclude that TRACE code can reproduce transient events at a sufficiently high accuracy.
- To clarify the cooling characteristics of fuel rods during a water level decrease in reactor vessels, a boil-off experiment using a simulated fuel rod bundle was conducted (Fig. 1). Our experiments clarified the effect of flow rate, thermal bundle power and water temperature on the flow dynamics (void fraction, bubble velocity, bubble diameter, etc.) and fuel rod temperature under atmospheric pressure condition, making the accurate evaluation of time required for fuel rod damage possible[*1].
- Installation of a filtered containment venting system (FCVS) is required for severe accident countermeasures. For the FCVS operation, understanding of its performance in various accident situations and building a database on the decontamination factor under various conditions is necessary. For this purpose, aerosol test, iodine (I_2) test and organic iodine (CH_3I) test was performed under atmospheric pressure using a quasi-full-scale test facility. Moreover, a quasi-full-scale test facility which can test in high temperature and pressure condition was constructed (Fig. 2)[*1].
- For use in probabilistic risk assessments (PRAs) of Japanese nuclear power plants (NPPs), a common cause failure (CCF) database containing the analysis results of CCFs of domestic NPPs has been developed in the NUCIA system in JANSI. CCF is one of the dominant risk sources in NPPs. Moreover, candidate solutions have been proposed for the technical problem that a portion of the Monte Carlo calculations for failure rate estimation of components with rare failures are not sufficiently converged; a problem revealed in the application of JANSI generic failure rates to the PRA in domestic NPPs.

Nuclear Fuel and Reactor Core

Research aiming for enhancement of light water reactor safety has been conducted to obtain an understanding of cladding degradation mechanisms, clarification of chemical properties, thermal performance and mechanical behavior of nuclear fuel in accidental conditions, and upgrading computer tools for reactor core burnup performance analysis. Evaluation of molten fuel characteristics and development of subcriticality measurement methodology for fuel debris have also been promoted for contributing to decommissioning of Fukushima-Daiichi nuclear power plant.

- For securely maintaining subcriticality of a degraded reactor core during a severe accident, the concept of "accident-tolerant control rod"[*2] was proposed, which would be intact before fuel meltdown, coexist with fuel material after fuel meltdown, and be operational in a similar manner to conventional control rods during normal reactor condition. The candidates of appropriate neutron absorber materials and the fundamental structures of accident-tolerant control rod were obtained as a result of reactor performance analyses and material tests. (L13005)
- Toward the preparation for fuel debris removal in Fukushima-Daiichi nuclear power plant, an ultra-high temperature furnace for manufacturing simulant fuel debris was upgraded so as to attain a temperature higher than 2700 degree-C, which is necessary to melt fuel. This furnace was successfully used to fabricate ziconia-calcia mixed oxide melt (ZrO_2 - CaO) having the melting point near to that of the fuel debris.[*3]

Nuclear Fuel Cycle

For the early commencement of commercial operation at Rokkasho reprocessing plant, we conduct experiments necessary for the new safety regulation standard. Also, the development of contaminated water treatment technology is carried out for preventing the spreading of radioactive contamination in the case of a severe nuclear accident. Applying the pyrochemical technology, "debris" fuel is estimated in order to develop a treatment method for the damaged fuel generated in a core meltdown accident. In this way, CRIEPI can maintain the technical level of pyrochemical process.

■It was necessary to study the radiation elements release behavior during dry out of the high level liquid waste concentrated solution (HLLW) in the event of a severe accident at a reprocessing plant. Release rates of Ru element, which is major elements for radioactive risk, were measured using actual HLLW. Also the Ru release rate from the residue (Mo, Ru, Rh, Pd and Te metallic particles remaining in the fuel dissolution process) was estimated under the high temperature condition during a severe accident. Results obtained in the experiment using the simulated residue showed that very little Ru was released.

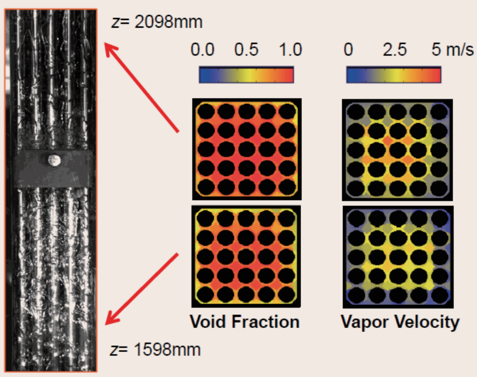
■Electrochemical properties of simulated debris in molten salt were estimated and the reduction behavior of the debris was also studied in order to develop a pyrochemical technology which can be applied to the debris fuel generated during a core meltdown accident. Additionally, the reduction tests using actual debris generated in an TMI accident were carried out. Sufficient data were obtained to evaluate the applicability of pyrochemical technology to the debris fuel treatment.

Human Factors

In order to contribute to building an organization that exhibits good performance without any human error during both normal operation and emergencies, we will develop measures toward preventing human error and fostering a safety culture by bringing out the features of individuals, teams, and organizations.

■In order to promote embedding and observance of safety behaviors which are essential regardless of normal operation and emergencies, we clarified work conditions (work characteristics and structures such as being put under time constraint and troublesome tasks) that affect the psychological process of safety behaviors*4 by literature review and subjective experiments. It is expected that introducing safety rules in consideration of the effects of these conditions will promote self-active safety behaviors by workers. (L13001, L13004)

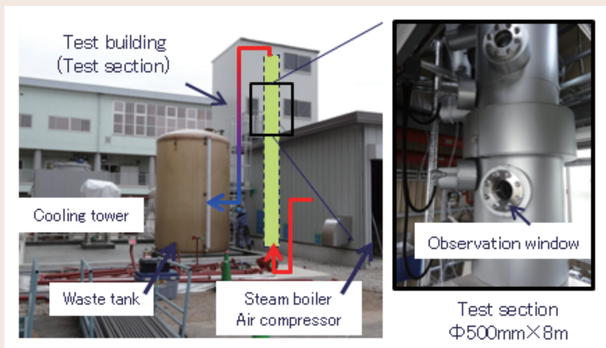
*1 Part of this research was conducted as the Infrastructure Development Project for Enhancement of Safety Measures at Nuclear Power Plants sponsored by the Ministry of Economy, Trade and Industry.
*2 Accident-tolerant control rod is included in the framework of the Civil Nuclear Energy R&D WG between the Japanese and US governments.
*3 entrusted by Japan Atomic Energy Agency.
*4 A process that a worker decides to perform a safety behavior through "a subjective assessment of hazard", "a supposition of the positive result of the behavior (ability to avoid risk, feeling of satisfaction)", and "a supposition of the negative result of the behavior (decrease in work efficiency, increase in workload)". (I.e. safety behavior that is supposed to avoid high risk and to be low workload will be performed even if the hazard is invisible.)



Side view of a rod bundle

Cross-sectional distribution of flow parameters

Fig. 1: Visualization of boiling two-phase flow in 5×5 rod bundle(inlet flow rate 0.3m/s, inlet subcooling 1 K, thermal bundle power 37.5 kW)



Overview of a quasi-full-scale FCVS test facility

Enlarged view of the test section

Fig. 2: Quasi-full-scale FCVS test facility. Specifications: test section (inner-diameter: 500 mm, hight: 8 m, design pressure: 1.6 MPa), steam boiler (mass flow: 800 kg/h, pressure: 0.8 MPa), air compressor (flow rate: 4.0 m^3/min).