# Nuclear Technology
# − Supporting Foundations for a Stable Supply −

## Brief Overview

We steadily promoted nuclear technology that has central function for global warming measures and support foundation of stable power supply in the future.

For plant life management of light water reactors, we continuously contribute to preparing the road map for the industrial world on various technical subjects related to ageing degradation to promote development of prediction, assessment, and countermeasure technology for irradiation embrittlement, degradation caused by thermal hydraulics, and stress corrosion cracking (SCC). In particular for irradiation embrittlement, we investigated irradiation embrittlement mechanism for pressure vessels during high neutron irradiation through microstructural observation of surveillance test specimens for pressure vessels and prepared on the master curve fracture toughness test method for the domestic pressure vessel steels.

For backend project support researches, we developed high-, low-level radioactive wastes disposal technology and recycled fuel transport and storage technology to support projects by the national government and the power industry smoothly. Among them, for the margin depth disposal of low-level radioactive wastes, research accomplishments of long-term durability for engineered barrier materials in bentonite and cementituous materials were reflected in the technical reports of the Japan Society of Civil Engineers and private standards in the Atomic Energy Society of Japan.

## Achievements by Research Theme

**Plant life management research of light water reactors (PLM general project)**

●High accuracy prediction and its standardization of irradiation embrittlement
- Investigated irradiation embrittlement mechanism of pressure vessel during high neutron irradiation of $6 \times 10^{19}$ n/cm$^2$ by microstructural observation of surveillance test specimens and archive pressure vessel materials irradiated at test reactors.
- Prepared standard drafts of master curve fracture toughness test adaptable for domestic pressure vessel steel. [Q08025]

○Comprehensive measures of degradation caused by thermal hydraulics
- Improved assessment function of the Flow Accelerated Corrosion evaluation method by applying iron solubility and oxide film property model. [L08016]
- Confirmed adaptability of the Liquid Droplet Impingement Erosion evaluation method based on three-dimensional steam flow calculation to wall thinning location and shape prediction. [L08019]

○Advanced SCC assessment method
- Clarified the loading direction effect to crack propagation property at the weld boundary of stainless steel (Fig. 1). [Q08020]
- Clarified hydrogen peroxide influence to SCC sensitivity of stainless steel at BWR plant start-up. [Q08022]

○Plant life management measures
- Comprehended the effect of evaluation of oxidation progress and the residue of antioxidant of insulating materials for assessment of deterioration property of actual cable.
- Contributed to the FY 2009 road map for industrial world on plant life management research.

**Radiation safety**

○Assessment of low dose radiation effect
- Clarified necessity of radiation protection model considering dose rate by epidemiological study for inhabitants living at high natural radiation areas in India and China, and various experiments on animals or cells.

○Rationalization of ensuring radiation safety
- Expansion of CRIEPI's clearance level measuring devices for metal wastes to apply concrete wastes.
- Improved explanation of safety of sub-surface disposal with engineered barriers through risk-informed safety assessment analysis applying uncertainty quantification method developed by CRIEPI.

**Backend project supporting research**

●High-level radioactive waste disposal
- Concluded various element technologies from the viewpoint of area selection survey and site characteristic survey.
- Assessed adaptability of characteristic assessment method of nuclide migration in base rock by the underground water flow assessment method and tracer experiments.
- Collected material characteristics data of various low alkaline cements to assess adaptability as a disposal site construction material.

●Low-level radioactive water disposal
- Experimentally clarified eluviations of engineered barrier materials (cement and bentonite) at disposal site environmental conditions and alteration characters.
- Clarified permeability mechanism of gas generated from wastes to develop resultant stress combined analysis model.
- Reflected research achievements related to long-term durability of bentonite and cement engineered barrier materials to technical reports to Japan Society of Civil Engineers and private standards of Atomic Energy Society of Japan (Fig. 3). [N06028]

●Transpiration and storage of recycled fuel
- Monitored aged deterioration of concrete cask containing spent fuel stored for 15 years to ensure its integrality.
- Developed SCC assessment method and countermeasure technology for canister materials of concrete cask.
- Conducted the simulation test of airplane crash onto metal casks to confirm validity of confining function assessment method.

**Next generation nuclear technology**

●Metallic fuel cycle
- Demonstrated complete recovery of long lived transuranics elements by dry separation technology from real high level liquid waste generated from PUREX reprocessing of spent LWR fuel. [L08011]
- Derived mass balanced data at the main process based on Pu and U tests relating to dry reprocess technology.
- Developed structures and crucible materials with less actinide loss for cathode processor of engineering scale. [L08005]

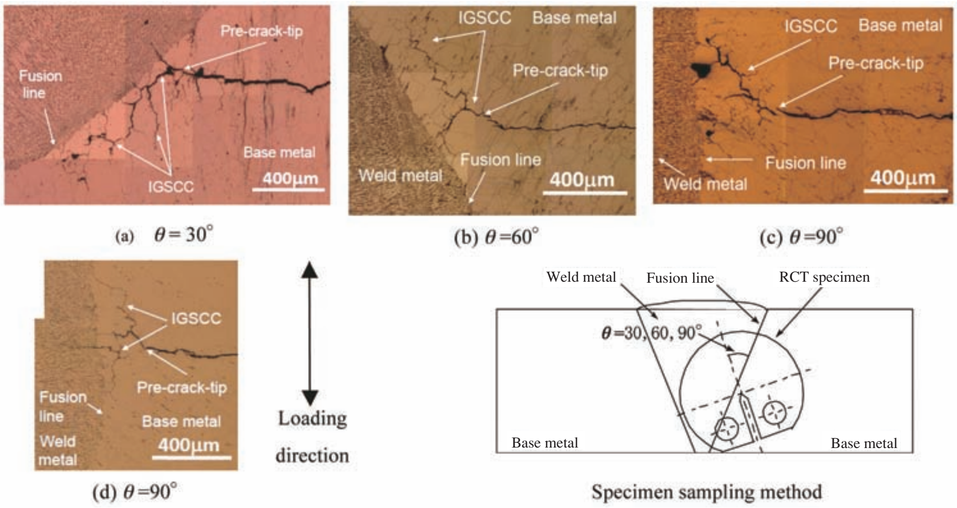



**Fig.1** Examples of cracking morphologies near weld fusion line.

**Fig.2** Verification test for surface contamination measurement at the Kashiwazaki-Kariwa Nuclear Power Station of Tokyo Electric Power Company

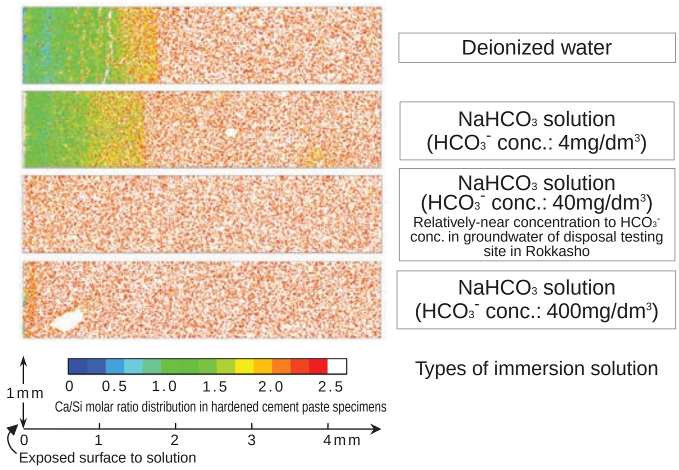
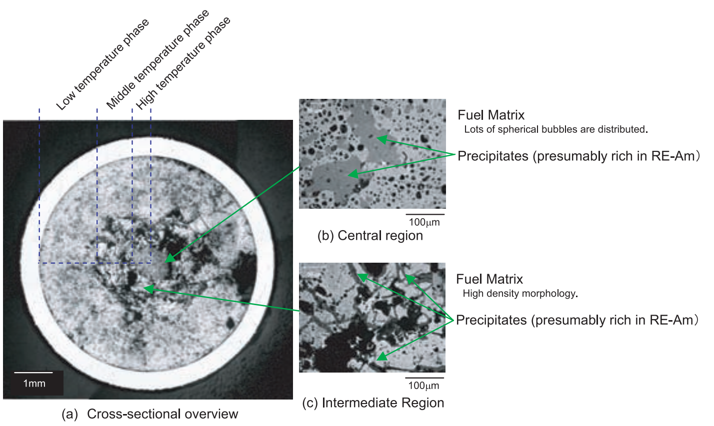



**Fig.3** Ca/Si molar ratio distribution in hardened cement paste specimens exposed to solutions with various levels of $HCO_3^-$ concentration

The retardation effect of leaching of hardened cement paste significantly depends on $HCO_3^-$ concentration of immersion solution.




- Three distinct concentric zones are visible corresponding to the radial temperature distribution.
- The characteristics of fuel matrix morphology in each region were similar to those of MA-free U-Pu-Zr fuels.
- Precipitates as large as ~200 $\mu$m appeared in the high-temperature central region.

**Fig.4** Optical metallography of irradiated U-Pu-Zr alloy containing 5%MA and 5%RE.