The preliminary study for safety design of JAEA's Radioactive Material Analysis and Research Facility "Laboratory-2" dedicated to fuel debris analysis at TEPCO's Fukushima Daiichi Nuclear Power Station site

Akinori Sato, Ichiro Kosaka and Naoya Kaji

Japan Atomic Energy Agency, Fukushima, Japan
Corresponding author: Akinori Sato <sato.akinori@jaea.go.jp>

According to "Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station" (Roadmap) established by the Inter-Ministerial Council for Contaminated Water and Decommissioning Issues, prospects of a processing/disposal method and technology related to the safety of radioactive material should be made clear by around FY2021. Regarding the fuel debris, it is noted that a processing and disposal method of fuel debris will be decided after the start of retrieving. The analysis and research facility to receive vision for processing and disposal method of fuel debris is now being designed.

The Japan Atomic Energy Agency (JAEA) is now constructing Radioactive Material Analysis and Research Facility. It is built near the Tokyo Electric Power Company Holdings, Incorporated (TEPCO) Fukushima Daiichi Nuclear Power Station (1F) site in order to perform and collaborate with TEPCO's activities in 1F. The facility consists of three buildings: the administration building, Laboratory-1 and Laboratory-2 shown in Figure P6.

The administration building started operation in March 2018 provides office space, meeting rooms for researchers, and apparatus mock-up. Laboratory-1 is now under construction for radioactive analysis of low and medium level radioactive rubbles and secondary wastes. Laboratory-2 is planned as dedicated hot laboratory for radioactive analysis, and mechanical and chemical characterization of fuel debris. Therefore, some specific issues, such as shielding and criticality safety should be investigated as preliminary studies.

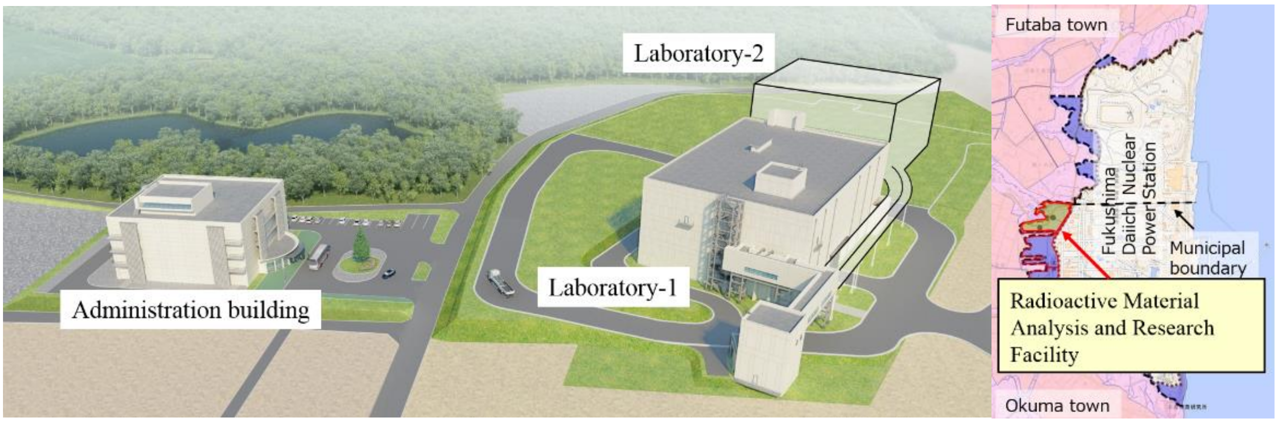

Figure P6: Completion image (left) and location (right) of Radioactive Material Analysis and Research Facility

Description of Laboratory-2

Analytical items and Fuel debris treatment process. In order to characterize chemical and mechanical property of fuel debris in Laboratory-2, candidates of analytical items are shown in Table P1. The amount of fuel debris transported using conventional fuel cask from the 1F site is scheduled to be 12 times/year and set less than 60 kg/year (less than 5 kg per each transportation). The analysis flow of fuel debris is shown in Figure P7. After pre-treatment such as cutting, polishing and dissolution performed in concrete cells, samples (processed fuel debris) are transferred to glove boxes after chemical separation in steel cells. The samples are analyzed in glove boxes using several kinds of equipment. After analysis, samples including not analysed (residual) fuel debris are stored in the concrete cell pit, and then will be returned to the 1F facility.

Table P1. The expected analytical items in Laboratory-2

	Laboratory -2 (TBD)
Main Analysis items	Radioactivity Elemental analysis Organic matter Surface analysis Chemical analysis Hydrogen gas Mechanical characteristics Specific surface area / Particle size distribution Density Thermal property Calorific/Heating value Others (High-temp. properties, etc.)
Main Preparation	Cutting machine Polishing machine Electric discharge machine
Main Pre-treatment	Alkali dissolution units Acid dissolution units
Others	X-ray computed tomography scanner Pneumatic carrier system

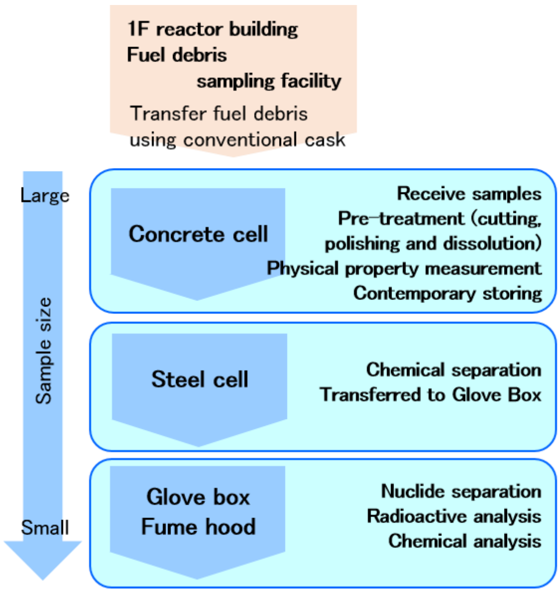

Figure P7: The analysis flow of fuel debris

Preliminary evaluation of safety design condition

The difference between Laboratory-2 and other hot laboratories is the fact that there is no reliable information on fuel debris to use as safety design conditions. For Three Mile Island (TMI) accident and Chernobyl accident, there have been many reports on the characterization of fuel debris. On the other hand, different factors, such as fuel composition and fuel type (MOX fuel) should be considered in the 1F reactors. Thus, it is necessary to consider the existence of fuel debris as various compositions depending on the location as is shown in Figure P8. Therefore, practically conservative assumption is requested for the safety design condition of Laboratory-2, especially concerning the radiation shielding and criticality safety parameters.

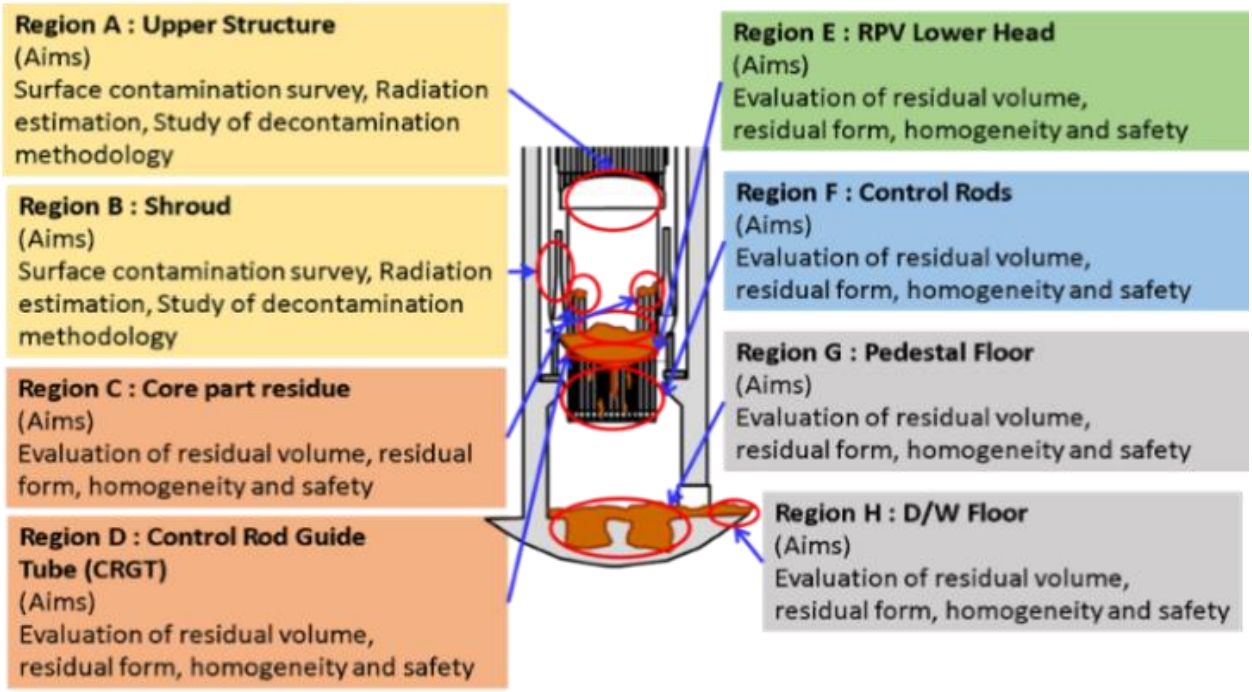

Figure P8: Image of 1F core

For evaluations of radiation shielding to the public at site boundary and workers in Laboratory-2, dose rates have to be evaluated. As a first step, it is necessary to estimate the source intensity for shielding evaluation by calculation. The calculation of source intensity was performed using estimated conservative parameters based on nominal data of reactor core before the 1F accident. Burn-up, uranium enrichment and nuclide composition of MOX fuel were adopted as parameters. The effect of activation of structural material was also included. As shielding calculation system under conservative condition, the radiation source was set as void to exclude self-shielding, because there are no data on parameters of fuel debris that affect the self-shielding such as density and composition.

As a preliminary evaluation of criticality safety, the critical mass limit was evaluated under conservative conditions such as fresh fuel and 30 cm water reflection, considering both uranium fuel and MOX fuel.

Knowledge from results / Summary

Through preliminary calculations, it is found that shielding design of Laboratory-2 is achievable even if calculations are performed under conservative conditions. Through preliminary calculation of critical mass limit, handling of 5 kg fuel debris is concluded to be feasible, considering both uranium fuel and MOX fuel.

As a result of the preliminary evaluation by using conservative condition and nominal 1F fuel data, it is implied that the condition as applied to post irradiation examination facility of irradiated fuel can be adopted to start safety design for our Laboratory-2.