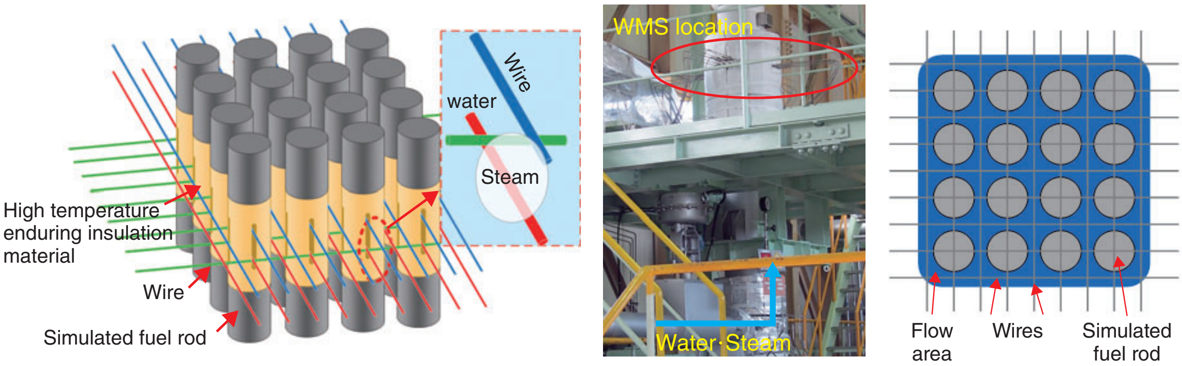
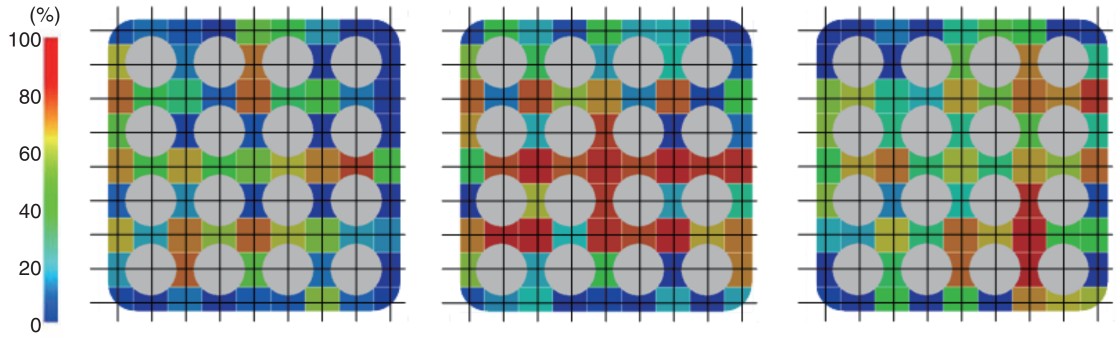
4–9 Visualization of Water–Steam Fractions under High-Pressure and High-Temperature Conditions in Reactor Cores

— Development of Technology for the Measurement of Steam Volumetric Fractions in Fuel Assemblies with Wire-Mesh Sensors —



(a) Structure of the WMS (b) Outline of the test apparatus (c) Cross-sectional view of the WMS location

Fig.4-19 Test apparatus and principles for the measurement of steam volumetric fractions in fuel assemblies
WMS measures the electrical current flowing through the small space between two wires by applying electrical voltages over these wires. If steam exists in the small space between them, electricity will have difficulty flowing, resulting in a lower current. For example, the current is low between the red and green wires in (a); however, it is high between the green and blue wires. To enhance the measuring technology for application to two-phase flow under high-pressure and -temperature conditions, materials with high temperature endurances and electrical insulation were adopted for the setting of the wires during manufacture of the test apparatus.



(d) At measurement starting time (e) 0.012 s after start (f) 0.024 s after start

Fig.4-20 Changes in the measured steam volumetric-fraction distribution with time
These figures show the measurement results at pressure = 2.6 MPa, temperature = 226 °C, water velocity = 0.58 m/s, and steam velocity = 1.74 m/s. The high-steam-volumetric-fraction region appears in the central part of the test channel (e); however, it becomes low (f). These figures show that the steam volumetric-fraction distribution changes significantly over a short time.

In boiling-water reactor cores, the coolant is heated by fission energy, resulting in boiling. Water flow (liquid phase) mixed with steam (vapor phase) is called water–steam two–phase flow (abbreviated to two-phase flow in the following). The temperature distributions in the reactor cores are affected significantly by the two-phase-flow conditions, especially the steam volumetric fraction. Therefore, when designing a reactor core or analyzing an accident, it is very important to evaluate the steam volumetric fraction inside the cores. However, the measurement of this fraction inside a reactor core under high-pressure and high-temperature conditions is very difficult because the flow area of the coolant is very small owing to the existence of fuel rods. This is the reason that most experiments have been performed under low-pressure and low-temperature conditions. To evaluate the steam volumetric fraction in the reactor cores, we developed a measurement technology that can be applied to high-pressure, high-temperature test apparatuses that simulate reactor cores as based on wire-mesh sensor (WMS) technology.

Fig.4-19(a) shows the structure of the constructed WMS and the basic principle of its operation. Metals with high electrical conductivity affect the WMS measurement, which is based on the measurement of electrical current. Therefore, as shown in Fig.4-19(a), an electricity insulation material capable of enduring very high temperatures was inserted between the upper and lower metal tubes that simulated fuel rods, with very small holes drilled in the material for the setting of wires. Fig.4-19(b) shows the outer appearance of the test apparatus. To simulate the reactor cores, the test section has a height of approximately 4 m and comprises four rods in both the vertical and horizontal directions, meaning a total of 16 rods (Fig.4-19(c)). As an example of the test results, Fig.4-20 shows how the measured steam volumetric fraction changes with time. The red part (representing a high fraction of steam) and the blue part (representing a high fraction of water) change complicatedly over a short time interval. In the future, to contribute to evaluations of the temperature distribution and boiling condition inside real reactor cores, we will obtain more experimental data to validate simulation codes, and to evaluate and improve experimental correlations.

Reference

Liu, W. et al., Measurement of Void Fraction Distribution in Steam–Water Two–Phase Flow in a 4×4 Bundle at 2 MPa, Transactions of the American Nuclear Society, vol.114, 2016, p.875-878.