Experimental Setup with Transient Behavior of Fuel Cladding of SFR

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1. Introduction

The tempered-martensitic steel such as HT9 was investigated for advanced cladding in sodium-cooled fast reactors. It is promising alloy because of its high resistance to swelling, good resistance to irradiation creep, and low thermal expansion coefficient [1,2]. Nowadays, in Korea, advanced cladding such as FC92 is developed and its transient behaviors are required for the safety analysis of SFR. Design and safety analyses of sodium-cooled fast reactor (SFR) require understanding fuel pin responses to a wide range of off-normal events. In a loss-of-flow (LOF) or transient over-power (TOP), the temperature of the cladding is rapidly increased above its steady-state service temperature. Transient tests have been performed in sections of fuel pin cladding and a large data base has been established for austenitic stainless steel such as 20% cold-worked 316 SS [3-6] and feriritc/martensitic steels such as HT9 [7-9].

The concept and general approach of transient testing of fuel cladding specimens was to define the failure stress and strain for fuel cladding in the temperature up to 1400℃ at transient heating rates of 0.56, 5.6 and 111℃/sec [7,8]. To demonstrate the transient environment, the experimental systems are introduced in this study. Results obtained for FC92 and HT9 cladding in a variety of simulated transients using the facility in KAERI will be further presented.

2. Technical Status

Simulated transient tests have been conducted over HT9 since 1980's for securing safety analyses in the U.S. Westinghouse has constructed fuel cladding transient test (FCTT) system to evaluate the strength of both irradiated and un-irradiated HT9 claddings.

In the FCTT system, the ramp test condition which simulates a typical TOP event and the ramp and hold test condition which simulates LOF event were introduced as shown in Fig. 1. The desired transient temperature was achieved by an induction coil system with feedback thermocouple spot welded to the cladding specimen. During the transient test, pressure was measured with a strain-gauge pressure transducer. The pressure transducer output recorded during each test.

The stress versus failure temperature results obtained from un-irradiated HT9 is shown in Fig. 2. The failure criterion developed on the basis of standard FCTT data has been used for fuel assembly design and safety to analyze LOF events in which the temperature is assumed to increase steadily. Also, a general description of the correlation for the ramp test has been developed.

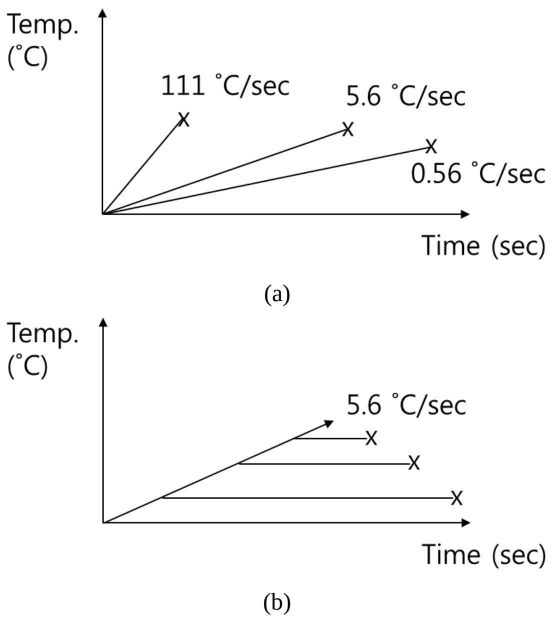

Fig. 1. Typical transient test conditions; (a) ramp test and (b) ramp and hold test.

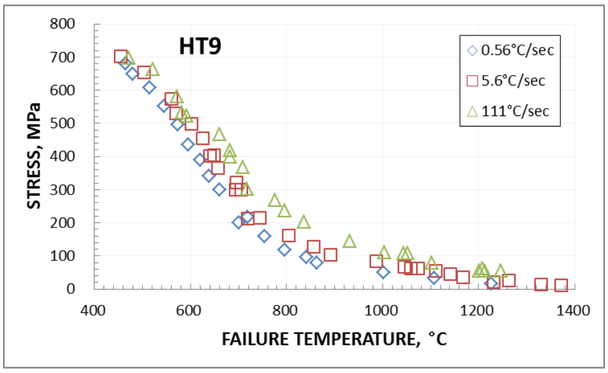

Fig. 2. Stress versus failure temperature results for un-irradiated HT9 cladding with 0.56, 5.6, and 111℃/sec transient heating rates of the ramp tests [7,8].

JAEA has operated the temperature-transient-to-burst tests on both irradiated and un-irradiated cladding PNC-FMS. The specimens were internally pressurized by pure argon gas and then heated linearly by direct electric current in order to study rupture strength during ramp heating [9,10]. A type R thermocouple was spot-welded at the axial center of each cladding. The heating rate was controlled at 5℃/sec, which is the hypothetical heating rate of operational transients in MONJU. The hoop stress conditions, which were estimated by thin walled tube approximately, were 120 and 196 MPa.

Transient tests are being conducted on FC92 and HT9 claddings in KAERI. Tests were conducted by pressurizing the specimen to a predetermined value and then increasing the temperature at a fixed heating rate until rupture occurred. Specimen heating was achieved with a radiant heating until its failure. Vertical type of infrared heating system was adopted to give a stable temperature along the axial direction of cladding. This heating system gives temperature stability on cladding especially under the ramp and hold test. Tests were conducted at heating rates of 0.56, 5.6 and 20℃/sec, where 20℃/sec is reported to be enough to simulate fast ramp condition from the previous LWR operation [11]. The schematic of transient test facility is shown in Fig. 3. The diameter of claddings was measured using micro-meter at near the breach region after failure occurred. Data obtained included failure temperature, strain, and hoop stress calculated from the internal pressure and initial specimen geometry. After tests finishes, a general description of the correlation of FC92 for the ramp test will be developed. Detailed test conditions for ramp test are summarized in Table I.

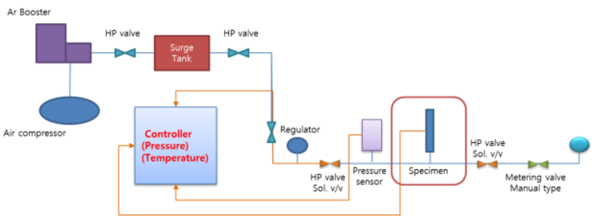

Fig. 3. Schematic of transient test facility in KAERI.

Table I: Ramp test conditions

Material	FC92B, FC92N, HT9
Heating rate	0.56°C/sec, 5.6°C/sec, 20°C/sec
Hoop stress	500 MPa to 50 MPa

3. Conclusions

This paper summarizes the technical status of transient testing facilities and their results. Previous researches showed the transient behaviors of HT9 cladding. For the safety analyses in SFR in Korea, simulated transient tests with newly developed FC92 as well as HT9 cladding are being carried out. A development of a general description of the correlation of FC92 for the ramp test has been also planned.

Acknowledgement

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REFERENCES

[1] D. S. Gelles, Effects of Irradiation of Ferritic Alloys and Implications for Fusion Reactor Application, Journal of Nuclear Materials, Vol.149, p. 192, 1987.

[2] R. J. Puigh and G. L. Wire, In-Reactor Creep Behavior of Selected Ferritic Alloys, AIME Proceedings of Topical Conference on Ferritic Alloys for Use in Nuclear Energy Technology, Jun.19-23, 1983, Snowbird, UT.

[3] M. L. Hamilton, G. D. Johnson, C. W. Hunter, and D. R. Duncan, Mechanical Behavior of Irradiated Fuel Pin Cladding Evaluated Under Transient Heating and Pressure Conditions, British Nuclear Energy Society Conference, Apr., 1983, Brighton, England.

[4] C. W. Hunter, R. L. Fish, and J. J. Holmes, Mechanical Properties of Unirradiated Fast Reactor Cladding During Simulated Overpower Transients, Nuclear Technology, Vol.27, p. 376, 1975.

[5] C. W. Hunter and G. D. Johnson, Mechanical Properties of Fast Reactor Fuel Cladding for Transient Analysis, ASTM STP 611, American Society for Testing and Materials, 1976, Philadelphia, PA.

[6] W. F. Brizes and G. D. Johnson, Analysis of the Mechanical Properties of Irradiated Fuel Pin Cladding Relative to Transient Performance Applications, British Nuclear Energy Society Conference, Apr., 1983, Brighton, England.

[7] N. S. Cannon, F. H. Huang, and M. L. Hamilton, Simulated Transient Behavior of HT9 Cladding, WHC-SA-0127-FP, 1988.

[8] F. H. Huang, Transient Failure Behavior of HT9, WHC-SA-2513-FP, 1994.

[9] I. Shibahara, T. Omori, Y. Sato, S. Onose, and S. Nomura, Mechanical Property Degradation of Fast Reactor Fuel cladding During Thermal Transients, ASTM STP 1175, American Society for Testing and Materials, 1993, Philadelphia, PA.

[10] Y. Yano, T. Yoshitake, S. Yamashita, N. Akasaka, S. Onose, S. Watanabe, and H. Takahashi, Effects of Fast Reactor Irradiation Conditions on Tensile and Burst Properties of Ferritic/Martensitic Steel Cladding, Journal of Nuclear Science and Technology, Vol.44, p. 1535, 2007.

[11] D. A. Powers and R. O. Meyer, Cladding Swelling and Rupture Models for LOCA Analysis, NUREG-0630, 1980.