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Transient Failure Behavior of HT9

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Prepared for the U.S. Department of Energy
Office of Environmental Restoration and
Waste Management

Westinghouse
Hanford Company Richland, Washington

Hanford Operations and Engineering Contractor for the
U.S. Department of Energy under Contract DE-AC06-87RL10930

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Date Published
July 1994

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DISCLM-2.CHP (1-91)

TRANSIENT FAILURE BEHAVIOR OF HT9

ABSTRACT

Alloy HT9 has been chosen as candidate materials for fast and fusion reactor applications because the material exhibits excellent resistance to void swelling. However, ferritic alloys are known to undergo a ductile-brittle transition as the test temperature is decreased. This inherent problem has limited their applications to reactor component materials subjected to low neutron exposures. Despite the ductile-brittle transition problem, results show that the materials exhibit superior resistance to fracture under very high neutron fluences at irradiation temperatures above 380°C. Results also show that the transient behavior for HT9 cladding specimens taken from the fuel column region and cladding taken from outside the fuel column or unirradiated cladding are the same. HT9 cladding maintained its transient strength with irradiation to a fluence of 9×10^{22} n/cm² ($E > 0.1$ MeV).

1.0 INTRODUCTION

Fuel pellets are filled in long cylindrical cladding tubes in a nuclear reactor. These tubes are very thin with a thickness of 0.35 mm and an inner diameter of 4 to 5 mm. They are constantly subjected to severe mechanical loads and thermal environments during normal operation of reactors or transient heating conditions in off-normal events. The cladding will deform by the internal pressure at elevated temperatures, the expanding fuel pellets, and the rapid rise of temperature in loss-of-flow (LOF) and transient overpower (TOP) events. As a consequence, the mechanical properties of the irradiated cladding will be altered under these circumstances. Understanding the mechanical behavior of fuel cladding for a wide range of temperature and stress states is thus essential for the reliable prediction of fuel pin performance during normal operation and off-normal situations.

It is desirable to measure the postirradiation fracture resistance of fuel cladding for reactor safety analysis. However, because the tube wall thickness is only 0.35 mm, fracture toughness test techniques on such a thin and curved structure have not yet been developed. Alternatively, experimental results have shown that the cladding failure behavior at a constant temperature can be characterized by "stress" and "rupture time" reasonably well, while parameters such as "stress" and "failure temperature" suitably describe the tube life of cladding pin subjected to off-normal transient conditions. In the following sections, stress rupture testing performed at a constant temperature and fuel cladding transient testing (FCTT) at a constant temperature ramp rate are discussed.

2.0 STRESS RUPTURE TESTING

Postirradiation stress rupture tests were first performed on 20% CW 316 SS cladding specimens irradiated in the EBR-II to fluences of 5.1 and 9.1×10^{22} n/cm² ($E > 0.1$ MeV) at 649°C . This early work was conducted to investigate the effect of neutron irradiation on the cladding rupture life of the material used in the FFTF first core. Results showed that average rupture lives were reduced by a factor of 20 to 30 over the unirradiated values. The cladding rupture testing proved to be effective for material development and useful to reactor design and safety analysis.

The in-reactor rupture strength of HT9 was evaluated by performing stress rupture tests on tubing samples. Specimens 22 mm long were cut from cladding with 4.6 mm outside diameter (OD) by 4.2 mm inside diameter (ID). Other specimens with a length of 28 mm were cut from cladding having 5.8 mm OD by 5.1 mm ID. The tube specimens were pressurized and welded by electron beam welding, not the standard TIG welds. Argon was filled in the specimens as the pressurizing medium and cover gas. They were soaked in furnaces at various temperatures. The stress rupture times were determined when the specimen failed. Earlier stress rupture data from irradiations in the EBR-II were obtained from 5.84 mm (OD) by 5.08 mm (ID) cladding fabricated from HT9 (Heat No. 91353). The failure time, hoop stress, and nominal and maximum diametral strains are given in Table 1.

The FFTF MOTA experiment contained rupture specimens fabricated from three ferritic alloys: HT9 heats 91353 and 84425, and 9Cr-1Mo. The HT9 stress rupture data exhibit rupture times similar to those determined from thermal testing (Figure 1). The same MOTA experiment had rupture specimens of 20% CW 316 SS irradiated to a peak fluence of 14×10^{22} n/cm². The rupture times of irradiated specimens of 20% CW 316 SS were found again to be shorter than those of unirradiated specimens.

Table 1 Stress rupture data for HT9 irradiated in the EBR-II

Temp. (°C)	Hoop stress (MPa)	Fluence ($\times 10^{22}$ n/cm ²)	Time (Hr)	(ΔD)/D (%)	(ΔD)/D _{max} (%)
593	170	1.18	1879	3.9	9.6
	187	1.10	1750	2.8	4.4
	215	0.153	243	3.8	8.7
650	75	1.81	2873	3.4	6.5
	80	1.07	1694	3.7	8.3
	90	0.387	615	2.1	6.5
704	34	0.354	563	20.7	39.0
	38	0.284	450	12.1	35.0
	48	0.166	264	4.9	23.0

3.0 FUEL CLADDING TRANSIENT TESTING

A detailed knowledge of fuel cladding behavior during reactor transients is required for core design and the safety analysis of both LOF and TOP events. The FCTT system was developed to simulate reactor transient heating and loading conditions in the laboratory [2]. Transient test equipment and procedures were developed for both unirradiated and irradiated cladding. The data obtained from simulated transients are useful for understanding the changes in mechanical properties of the irradiated cladding during an actual overpower transient. These cladding properties are necessary for analysis of

Figure 1 Stress rupture data for HT9 and 9Cr-1Mo [1].

design base transients, for determination of post-transient operation performance and cumulative damage rules, and for definition of fuel pin integrity limits.

3.1 Experimental Procedures

Simulated transient tests were conducted by using the FCTT system to heat the pressurized cladding sample at a fixed rate until rupture occurred. The specimen was heated to a predetermined temperature with closed-loop induction

power control and the feedback thermocouple spot welded onto the specimen. An extensometer was used to monitor the diameter of the specimen during the test. Specimens were tested at heating rates of 0.56, 5.6, and 110°C/sec with different induction coils. The heats of HT9 used to fabricate material specimens for EBR-II and FFTF experiments and for fuel pin experiments along with the vendors are listed in Tables 2 and 3, respectively.

Table 2 HT9 Heats Used for Mechanical Properties Experiments.

Heat	Vendor	Experiment
91353 Bar	Carpenter Technology	AAXV, MOTA
91353 Duct	Carpenter Technology	AAXV
91354 Bar	Carpenter Technology	AAXV, MOTA
91354 Tubing	Superior Tube Company	AAXV
84425	Carpenter Technology	MOTA
XA3607	Combustion Engineering	AAXV, MOTA

Table 3 HT9 Heats Used for Fuel Pin Experiments.

Heat	Vendor	Production Method	Fuel Pin Experiments
91353	CarTech	Vacuum Induction Melt, Electroslag Remelt	-
91354	CarTech	Vacuum Induction Melt, Electroslag Remelt	P44
84425	CarTech	Vacuum Induction Melt, Vacuum Arc Remelt	ACO-1, ACO-3, ACO-4, FO-2
92235	CarTech	Vacuum Induction Melt, Vacuum Arc Remelt	CDE
CV0385	ORNL	Vacuum Remelt of 91354	-
XA3607	Combustion Engineering	Electroslag Remelt of an Air Induction Melt	-
R74075	Federal Mogul	Hot Isostatic Powder Compaction at 2050°F	-

Irradiated specimens were prepared from fuel pins after all fuel was removed by mechanical drilling. The specimen was first pressurized to about 80% of desired pressure at room temperature, then heated to the initial temperature. After reaching the initial pressure and temperature, the specimen was heated at a constant rate until it failed. The parameters measured are failure temperature and failure strain at the selected hoop stress which remains relatively constant throughout the test. The failure strain measurement is taken as close as possible to the failure. A cumulative damage failure correlation can be developed from these FCTT data. Failure is predicted on the basis of a cumulative damage function (CDF) in which the life fraction is calculated from the correlation.

3.2 Transient Failure Correlation

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. In reality, LOF type events may give rise to relatively constant temperatures for a period of time. In this case, ramp and hold tests provide a better simulation of the transient conditions to which the fuel cladding is exposed. A general description of the correlation for FCTT and ramp/hold tests is given as follows. The temperature dependence of stress rupture life is represented in terms of the Dorn parameter, θ , and the activation energy, Q , as follows:

$$t_r = \theta \exp\left(\frac{Q}{RT}\right) \quad (1)$$

where $R = 1.986 \text{ cal/mole}\cdot\text{K}$.

The cumulative damage fraction (CDF) is calculated by integration:

$$\text{CDF} = \int_0^t \frac{dt}{t_r(\sigma, T)} \quad (2)$$

Since damage is accumulated in the cladding whether the temperature is ramped or held constant, the calculated CDF value should reflect the entire temperature history encountered:

$$\text{CDF} = \int_0^t \frac{dt}{t_r} + \sum_0^{t_i} \frac{t}{t_r} \quad (3)$$

where the first term reflects the damage accumulated during temperature ramps, and the second term reflects the damage accumulated at constant temperature. For the ramp and hold test, where the initial temperature is T_i , the heating rate is dT/dt , the hold temperature is T_H , and the hold time is t_H , Equation (3) becomes

$$CDF = \int_{T_i}^{T_H} \frac{dT}{\dot{T} \theta \exp\left(\frac{Q}{RT}\right)} + \frac{t_H}{\theta \exp\left(\frac{Q}{RT_H}\right)} \quad (4)$$

The first term in Equation (4) was computed previously [3] using the approximation

$$\int_{T_i}^{T_H} \frac{dT}{\exp\left(\frac{Q}{RT}\right)} \sim \frac{RT_H^2}{Q} \exp\left(\frac{-Q}{RT_H}\right)$$

and thus

$$CDF = \frac{1}{\theta} \left(\frac{RT_H^2}{\dot{T}Q} + t_H \right) \exp\left(\frac{-Q}{RT_H}\right). \quad (5)$$

The Dorn parameter was calculated for ramp and hold tests using Equation 5 and assuming the validity of the life fraction rule, which predicts that failure will occur when the CDF is equal to 1:

$$\theta = \left(\frac{RT_H^2}{\dot{T}Q} + t_H \right) \exp\left(\frac{-Q}{RT_H}\right). \quad (6)$$

For standard FCTT tests, the hold time t_H is zero, and $T_H = T_f$, where T_f is the failure temperature. The Dorn parameter is therefore expressed by

$$\theta = \left(\frac{RT_f^2}{\dot{T}Q} \right) \exp\left(\frac{-Q}{RT_f}\right) \quad (7)$$

The relationship between the Dorn parameter and stress is often described for transient data by an expression of the form

$$\ln \theta = A + B \ln \left[\ln \left(\frac{\sigma^*}{\sigma} \right) \right] \quad (8)$$

where σ^* is the hardness of the material. The constants A and B can be determined from the experimental FCTT or ramp/hold data. The appropriate value of θ can then be calculated for each given stress σ from Equation 8. Data from the FCTT transient tests at heating rates of 0.56, 5.6, and 111°C/s were used to develop the correlation for HT9 cladding (Figure 2). The

Figure 2 CDF correlation for the transient test on HT9 cladding.

results were fit to Equation 8 to determine constants A and B. A was found to depend on hoop stress and heating rate, while B was dependent of stress only. For a given stress, the value of θ is obtained from Equation 8 and the failure temperature is calculated from Equation 7. The stress versus failure temperature trend curves for unirradiated HT9 cladding are shown in Figure 3.

Figure 3 Stress versus failure temperature trend curves for unirradiated HT9 cladding [4].

3.3 Ramp and Hold Failure Correlation.

All specimens were initially heated from 370°C at 5.6°C/s until a preselected hold temperature was reached. Results showed that the measured failure times did not consistently agree with those predicted using the life fraction approach, with either the existing FCTT failure correlation or with the stress rupture correlation for unirradiated cladding. Each correlation did a reasonable job over only a portion of the temperature range of interest. A new correlation was therefore required to describe the ramp and hold test data adequately.

Ramp and hold test results for unirradiated HT9 cladding are given in Table 4. The values of $\ln \theta$ were calculated from Equation 6 for all hold temperatures with a Q of 70,170 cal/mole, hold temperature, hold time, and heat rate listed in Table 2. In Figure 4, the values of $\ln \theta$ are plotted as a function of $\ln [\ln (\sigma^*/\sigma)]$, where $\sigma^* = 730$ MPa. As shown in the figure, the data at each hold temperature appear to be linear, and all of the constant temperature lines are parallel. The data for all hold temperatures ranging from 500 to 1050°C were fitted in Equation 8 to determine the constants A and B for each hold temperature. The value of B was determined first to be about the same for all the hold temperature lines ($B = 12.47$). The values of A were then obtained by a least squares fit to the data with Equation 8. Figure 5 shows the temperature dependence of A , which can be described by the following equations:

Table 4 Ramp and hold data on unirradiated HT9 cladding.
Heating rate = 5.6°C/sec.

Stress (MPa)	Hold temp.(°C)	Hold time to failure (s)	Post-test Maximum strain(%)
517.7	500	1860	3.6
541.5	500	1418	3.1
486.7	500	15180	5.7
505.5	500	8820	4.0
315.7	600	653	5.6
364.4	600	123	4.5
260.4	600	9800	8.5
39.5	1050	56	41.5
24.7	1050	523	34.6
18.8	1050	1110	24.7

Figure 4 Stress dependence of Dorn parameter for various hold temperatures.

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Figure 5 Value of constant A as a function of hold temperature.

$$A = 24.942 - 0.153T_H + 9.488 \times 10^{-5}T_H^2 \quad T_H < 871^\circ\text{C}$$

(9)

$$A = -36.1 + 1.5 \tanh[(T_H - 871)/80] \quad 871^\circ\text{C} < T_H < 1050^\circ\text{C}$$

where T_H is the hold temperature in degrees centigrade. The value of A is assumed to saturate for hold temperatures above 1050°C.

The time to failure at a constant temperature for a given stress can now be predicted from Equations 6, 8, and 9. Hold time versus stress data are compared to the predicted values (solid lines) in Figure 6. The data and the predictions of the equations are in good agreement. A better fit of the A values in terms of T_H would result in even better agreement between predictions of hold time and the experimental data.

Figure 6. The hold time to failure and CDF failure predictions for various hold temperatures.

The failure correlation developed on the basis of ramp and hold test results can be used to predict not only the hold time to failure at a given hold temperature, but also the maximum hold temperature for a given hold time to failure. Clearly, calculation of the CDF should include the life fraction accumulated during ramp and hold periods to avoid overestimating cladding lifetime.

Examination of the data in Figure 4 revealed that there is an endpoint for each hold temperature line as the hold time for each temperature approaches zero. The dotted line in the figure connects the endpoints of all the hold temperature lines. This dotted line is the same line as the bilinear curve describing the relationship between the Dorn parameter and stress shown for standard FCTT tests at 5.6°C/s in Figure 2. It is clear that the standard FCTT condition is in fact the limiting case of the ramp and hold event, where the hold time is equal to zero, and that the failure correlation based on ramp and hold results should therefore be applicable to the prediction of failure temperatures for standard FCTT tests. Figure 7 shows the stress dependence of the failure temperature in standard FCTT tests conducted at 5.6°C/s. The prediction of the new failure correlation based on ramp and hold data is also plotted in the figure as a solid line for comparison. They are in reasonable agreement for a first iteration.

Figure 7 FCTT control data and failure correlation based on ramp and hold data.

Although analysis of ramp and hold tests can provide the information normally obtained from standard FCTT tests, more ramp and hold tests are required in order to firmly establish the linearity of the $\ln\theta - \ln\ln(\sigma^*/\sigma)$ lines at multiple hold temperatures. Generally, if hold time data are of interest, ramp and hold tests should be performed. A mixture of standard FCTT and ramp/hold tests should be performed to verify the failure temperature for cladding subjected to heating until failure.

3.4 Results and Discussion

In this section FCTT results obtained from irradiated cladding samples are presented. The tests were performed to evaluate cladding transient damage due to irradiation. Simulated transient tests were performed on HT9 cladding irradiated in the P44 experiment, structural specimens in the AAXV experiment, and cladding specimens cut from ACO-1 pins. All P44 specimens were tested at 5.6°C/sec, the AAXV specimens were tested at 111°C/sec, and the ACO-1 specimens were tested at the heating rates of 5.6°C/sec and 111°C/sec. Test results for the P44 specimens are shown in Figure 8. The behavior of unirradiated HT9 cladding is also plotted in Figure 8. As seen in this figure, the transient

Figure 8 Failure temperature of transient tests at 5.6°C/sec for P44 specimens of HT9.

failure temperatures and stresses for HT9 cladding specimens taken from the fuel column region and cladding taken from outside the fuel column or unirradiated cladding are the same. Results also showed that HT9 cladding maintained its transient strength with irradiation to a fluence of $9 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$). The strain in terms of failure temperature is plotted in Figure 9. Increasing temperature increases the strain of HT9 cladding although the data scatter is large.

6.0 CONCLUSIONS

FCTT testing shows that the transient behavior for HT9 cladding specimens taken from the fuel column region and cladding taken from outside the fuel column or unirradiated cladding are the same. HT9 cladding maintained its transient strength with irradiation to a fluence of $9 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$). The excellent resistance to cladding-fuel interaction and void swelling has led to choosing ferritic alloys as candidate materials for fast and fusion reactor applications.

Figure 9 Strain of irradiated HT9 cladding tested at 5.6°C/sec.

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