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SLOW CYCLE STRAIN FATIGUE IN THIN WALL TUBING

Preliminary Report

By
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METALS, CERAMICS, AND MATERIALS

SLOW CYCLE STRAIN FATIGUE IN THIN WALL TUBING:

PRELIMINARY REPORT

by

M. B. Reynolds

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ABSTRACT

A technique has been developed for slow cycle strain fatigue testing using specimens of thin-wall tubing of the type under consideration for use as super heat fuel cladding. Data on type 304 stainless steel has been obtained under radiation and in the absence of radiation. The strain cycle fatigue life of this alloy at 1200-1300 F is decreased three fold by the presence of a neutron flux of $1.5 \times 10^{14} > 1$ Mev. Out-of-reactor data has been obtained on Inconel at 1300°F.

INTRODUCTION

Fuel cladding for nuclear superheat reactor application is subject to considerably more severe environmental and stress conditions than those encountered in normal water reactor applications. These include more severe corrosive environment, higher surface temperature (1000 to 1500 F), and higher thermal stresses resulting from the temperature fluctuations peculiar to this type of reactor. In addition, the properties of the fuel cladding alloy are changed by the fast neutron radiation.

The changes in hardness, tensile properties, and impact strength of metals under irradiation have been dealt with at some length in the technical literature.⁽¹⁾ This literature deals chiefly but not entirely with the post-irradiation properties of alloys irradiated at temperatures below 200 F. With the exception of the Hanford creep experiments⁽²⁾ and the Oak Ridge "tube burst" tests⁽³⁾ almost no experimental measurements of deformation under irradiation have been carried out. The ability of superheat fuel cladding tubing to withstand severe local deformation without loss of integrity is of considerable importance since it may be expected that such tubing will experience highly localized stresses as the result of thermal gradients, external coolant pressure, and fuel pellet discontinuities. Little is known about the effect of fast neutron radiation or deformation under stress. In the Oak Ridge experiments short lengths of tubing are pressurized during irradiation at elevated temperature and deformation occurs under the resulting hoop stress, time-to-rupture being determined as a function of stress. Of the materials tested, Inconel, type 304 stainless steel, and INOR-8, are of interest as possible superheat fuel cladding materials. The stress rupture life of Inconel at 1500 F was found to be decreased several-fold by radiation. In the case of type 304 stainless steel rupture life at 1300 F was not significantly reduced but deformation at failure was considerably reduced.

The importance of elevated temperature ductility under radiation as a requirement for superheat fuel cladding led directly to an experimental program at Vallecitos Atomic Laboratory devoted to determination of the

effect of fast neutron radiation on this property. It was felt that since cladding would be subjected to a number of stress-reversals in service, some type of slow-cycle strain fatigue test would be preferable to a simple creep or stress rupture test of the Oak Ridge or Hanford types. Initial efforts were directed toward development of a bellows-actuated device which would subject a tubular specimen to cyclic uniaxial strain. After expenditure of considerable effort this approach was abandoned because of the inability of vendors to fabricate bellows of adequate load capacity under the temperature and neutron flux conditions existing within a nuclear reactor. A simple test using standard thin-wall fuel cladding tubing was finally devised. In this test, gas pressure is used to alternately expand and compress a specimen of tubing between a pair of rigid concentric mandrels. The strain range is determined by the diameters of the inner and outer mandrels relative to the specimen diameter and wall thickness. The strains experienced by the specimen are multiaxial, particularly at the tube ends, and may be somewhat difficult to interpret, but this disadvantage is considerably out-weighed by experimental simplicity and the fact that deformations are of the type to be expected in service. The details of test assemblies for reactor installation are to be presented in a separate report. ⁽⁴⁾

Experimental

Slow-cycle strain fatigue tests based on this principle have been carried out both out-of-reactor and in-reactor. Specimen size has been standardized at 1.250 inch nominal outside diameter and 0.016 inch wall thickness with an active length of 3 inches. The separate parts and a completed specimen-mandrel assembly, as used in out-of-reactor tests are shown in Figure 1. The specimen is cycled by application of argon pressure alternately to the inside and outside of the specimen. Operation of the cycling mechanism may be described by reference to Figure 2. Assume that valve V_{11} is open and valve V_{10} is closed, i.e., gas pressure is applied to the inner side only of the specimen. Operation of the timer will then close valve V_{11} and simultaneously open valves V_{21} and V_{10}

relieving pressure on the inside of the specimen and applying pressure to the outside of it. After a few seconds have passed the time delay switch D_2 opens allowing valve V_{10} to close. The next operation of the timer closes valve V_{21} and opens valves V_{11} and V_{20} . After a few seconds valve V_{20} is closed by time delay D_1 . The number of complete cycles is recorded by magnetic counter R. It will be noted that outlet valves V_{10} and V_{20} are normally closed, each opening briefly once per cycle to relieve pressure on one or the other side of the specimen. With these valves closed either leakage in an inlet valve or specimen failure will result in simultaneous appearance of pressure on both sides of the specimen. When this occurs, parallel-connected pressure switches P_1 and P_2 both open, releasing a holding relay and stopping the cycling mechanism and cycle counter. P_3 serves as a low limit switch, stopping the timer when gas source pressure falls below a preset minimum value.

Activation pressures in excess of 1000 psi have been used throughout the series of experiments resulting in specimen hoop stresses of approximately 40 kpsi compared with yield strength of less than 30 kpsi for specimen material.

Results

1. Type 304 Stainless Steel

Initial tests were carried out on type 304 stainless steel tubing. Out-of-reactor tests were carried out at 1300 F with the exception of two tests carried out at 1200 F. Total strain ranges are computed from the relation

$$\Delta \epsilon = \frac{D_o - D_i - 2t}{D_s}$$

in which

- D_o = inside diameter of outer mandrel
- D_i = outside diameter of inner mandrel
- D_s = average diameter of specimen
- t = specimen wall thickness

Plastic strain range may be estimated by subtracting from the total strain range twice the elastic strain the specimen material will support, i.e. twice the elastic limit at the test temperature divided by the Young's modulus at this temperature. For type 304 stainless steel this correction is approximately 0.002.*

The in-reactor tests in the General Electric Test Reactor (GETR) were carried out in four assemblies each containing three specimens. The specimens in two of these were to have been cycled to failure at temperature during irradiation, those in the other two were to have been irradiated at temperature and subsequently cycled to failure at temperature in the GETR canal. Numerous experimental and capsule design difficulties prevented the successful completion of the reactor tests. Two specimens in the first capsule were cycled to failure, although at a temperature of approximately 1200 F rather than at the desired temperature of 1300 F. In the fourth capsule irradiated, a specimen (total strain range, 0.0014) was subjected to 660 cycles when the test was terminated by reactor shutdown. At the date of this report an effort is being made to cycle this specimen and the remaining specimens in a second capsule to destruction in the GETR canal.

The results of both in- and out-of-reactor tests are summarized in Figure 3. Neutron flux was estimated to be $1.5 \times 10^{14} > 1$ Mev. The mode of failure depended somewhat upon the strain range to which the specimen was subjected. For total strain ranges above approximately 0.01, post-test examinations disclosed longitudinal wrinkling of the type illustrated in Figure 4. In such specimens failure was by longitudinal cracking of the type shown in Figure 5. The test was terminated when the first of these cracks penetrated a specimen wall. In each case cracking appeared to be intergranular as will be seen

* Based on an estimated elastic limit of 20 kpsi and a value of 20×10^6 psi for the Young's modulus at 1300 F.

from Figure 6 and 7, and originated on the outside surface of the specimen. In some instances (see Figure 8) a corrosion film was in evidence on the specimen surface although there is no evidence that this was directly related to specimen failure -- at least fracture appearance did not differ from specimens on which the film did not appear, as seen in Figure 9. There was evidence of the discontinuous grain boundary carbide precipitation believed to materially weaken austenitic stainless steels.⁽⁵⁾ In specimens subject to cycling over smaller strain ranges cracks were of shorter length, of which Figure 10 is typical.

In the case of one specimen of type 304 stainless steel cycled over a total strain range of 1% out-of-reactor a non-uniform deformation was observed. Instead of wrinkling, the failed specimen exhibited longitudinal bands in which "necking" or thinning occurred. Alternating with the "necked" bands were bands of greater thickness than the original specimen wall. This effect is illustrated in Figure 11.

Specimens cycled under radiation apparently differ from non-irradiated specimens only in failure rate since photomicrographs of failures in an irradiated specimen do not differ appreciably from those of an un-irradiated specimen cycled over a similar strain range.

It appears logical that, neglecting possible effects on the kinetics of carbide precipitation,* radiation should strengthen the grains relative to grain boundaries, making transgranular failure less likely to occur under radiation than in its absence.

Due to wrinkling the actual maximum fiber strains experienced by the specimens were larger than indicated by the mandrel dimensions. The magnitude of these strains can be determined by measurement of

* Such an effect appears unlikely in view of the similarity in microstructures in irradiated and unirradiated material.

the radius of curvature in wrinkles, in failed specimens. These maximum strains, while undoubtedly significant to specimen failure, be assumed to represent the cyclic strain range. Rather it is most likely that the wrinkles resulting in high fiber strains are produced by the first application of pressure to the outside of the specimen and that subsequent cycles do not reverse these wrinkles but result in positive and negative strain excursions about the wrinkled configuration. Figure 12 illustrates a typical wrinkle configuration in an assembly sized to give a total strain range, based on mandrel dimensions, of 0.12. Fiber strains based on curvature are indicated by numbers and arrows. It would appear, then, that wrinkling introduces a correction of unknown magnitude, the effect of which is to raise the high strain end of the $\Delta\epsilon$ vs N curve.

2. Inconel

Out-of-reactor tests have been initiated on Inconel and Incoloy tubing of the same dimensions as the stainless steel. Results of cycling tests to date are shown in Figure 13. No examinations of failures have yet been carried out, but will be described in future reports when completed.

SUMMARY

A Technique has been developed for carrying out slow cycle strain fatigue tests on thin wall tubing at elevated temperature both in the laboratory and in a nuclear reactor. Laboratory tests have been carried out on Type 304 stainless steel and Inconel, and a limited amount of testing on Type 304 stainless steel has been carried out under radiation. The behavior of Inconel in the absence of radiation is similar to that of Type 304 stainless steel but appears somewhat inferior at lower strain ranges. At higher strain ranges the results obtained with tubing compare favorably with data of Swindeman and Douglas⁽⁶⁾ for uniaxial strain, but life at lower strain ranges is considerably lower than would be predicted from their data. The threefold decrease in fatigue life of Type 304 stainless steel under

radiation is consistent with the Oak Ridge tube burst data, but additional fatigue data is badly needed.

The slow cycle fatigue data obtained to date show no difference between the behavior of Inconel and Type 304 stainless steel which could not be explained on a basis of variations in tube wall thickness and other experimental error.

Figure 1. MANDRELS, SPECIMENS AND ASSEMBLY

V₁₁ Inlet Valve, Inner
V₂₁ Inlet Valve, Outer
V₁₀ Outlet Valve, Inner
V₂₀ Outlet Valve, Outer
P₁ Pressure Switch, Inner

P₂ Pressure Switch, Outer
P₃ Pressure Switch, Low Limit
R Cycle Counter
D₁, D₂ Time Delay Switches

Figure 2. CYCLING MECHANISM SCHEMATIC

Figure 3. STRAIN-CYCLE TESTS, TYPE 304 STAINLESS STEEL AT 1300 F

V-3165

1.5X

Figure 4. FAILED SPECIMEN - LONGITUDINAL WRINKLES.

V-3170

7.5X

**Figure 5. LONGITUDINAL CRACK IN FAILED SPECIMEN.
TOTAL STRAIN RANGE 0.12**

V-3692

500X

**Figure 6. CRACK IN FAILED SPECIMEN . TOTAL
STRAIN RANGE RANGE 0.04. UNIRRADIATED.**

V-8518-6,7,8,9

500X

Figure 7. CRACK IN FAILED SPECIMEN. TOTAL STRAIN RANGE
0.023. IRRADIATED.

V-4079

500X

Figure 8. CORROSION FILM ON SPECIMEN SURFACE

V-3353

100X

Figure 9. CRACK IN SPECIMEN WITH SURFACE CORROSION FILM

V-3402

16X

Figure 10. FAILURE CRACK IN SPECIMEN. TOTAL STRAIN RANGE 0.01

V-4157

4.25X

Figure 11. THINNING IN FAILED SPECIMEN

V-4449

2.125X

Figure 12. MAXIMUM FIBER STRAINS IN FAILED SPECIMEN

Figure 13. STRAIN-CYCLE TESTS INCONEL, 1300 F.

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