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20. ABSTRACT (Continue on reverse side if necessary and identify by block number)

The effects of neutron irradiation and high-temperature water-vapor environment on the fatigue behavior of Type 348 stainless steel were investigated. At 550°F (288°C), fatigue life and crack-growth rate comparisons were made of irradiated and unirradiated sheet specimens as a function of total strain amplitude. The tests were conducted in reverse bending at a constant amplitude of vibration in dry air, in water vapor, and in vacuum. Tests in water vapor were designed to simulate water-cooled reactor environments. Notched specimens were cycled at resonance, and crack-growth rates were determined by a method that relates changes in the

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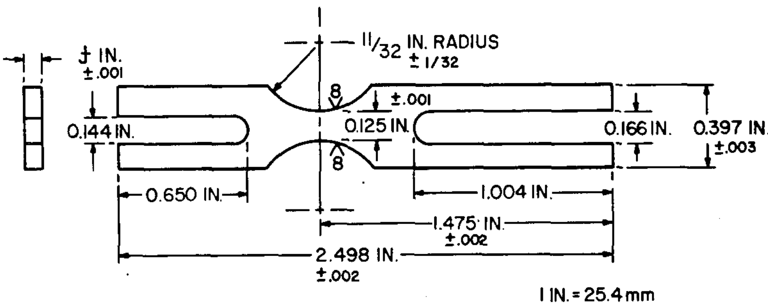

Fig. 1 — Dimensions of fatigue specimen

water reactor. Irradiation increased values of 0.2 percent yield strength from 42.3 ksi (292 MN/m^2) to 94.2 ksi (435 MN/m^2) and ultimate tensile strength from 63.1 ksi (435 MN/m^2) to 94.2 ksi (648 MN/m^2).

The fatigue machine used in this study was of the reverse-bending, constant-deflection type. It is described in detail in Ref. 16, and its general mechanical features are shown in Fig. 2. The sheet specimen is clamped in the pedestal and fitted with an extension rod which positions the permanent magnet away from the heat zone of the external split-tube furnace (not shown). As external electromagnets vibrate the specimen in reverse bending, the amplitude of vibration is measured by a traveling microscope. In the electronic control circuit there are two feedback loops: one to maintain a constant amplitude of vibration and the other to keep the vibration frequency of the specimen at resonance. The signal that is generated as the permanent magnet passes beneath the capacitor plates is used to trigger transistor switches which pass current to the electromagnets at the correct frequency. The growth of a crack causes the resonant frequency of vibration to decrease; after various frequency decreases, these changes are correlated to crack length as monitored by a microscope. This method, which has been described elsewhere (17), provides an indirect means of measuring crack depths.

Dynamic strains were measured in reverse bending for irradiated and unirradiated specimens at 550°F (288°C) by means of foil-type strain gages mounted on the gage section of unnotched fatigue specimens. The reported strain values represent the vibrational half-amplitude maximum total strain (percent) without correction for the notch. For the unirradiated material, the yield point, determined from both cyclic strain and tensile stress-strain measurements, occurred at 0.16 percent strain; therefore, strains above this value are comprised of both plastic and elastic components. For the irradiated material, strains were purely elastic at all strain amplitudes.

Fatigue evaluations were conducted in dry air, water vapor, and vacuum. For those conducted in air, the test chamber was first evacuated cold to approximately 10^{-5} torr ($\sim 10^{-3}$ N/m^2), then heated to 550°F (288°C) and held there until the pressure was again reduced to 10^{-5} torr. The vacuum system was then valved off and dry tank air (−57°F, −50°C dew point) was admitted until the gauge pressure was 2 lb (9 N). These procedures were intended to isolate any possible water vapor effects and to provide a reproducible test

FATIGUE RESISTANCE OF TYPE 348 STAINLESS STEEL UNDER SIMULATED WATER REACTOR CONDITIONS

INTRODUCTION

The influence of neutron irradiation on the fatigue behavior of austenitic stainless steels has become a matter of considerable interest in recent years due to the use of these alloys in existing reactors and in designs for proposed reactors. Core components such as fuel-element supporting structures and cladding are subjected to cyclic and transient mechanical and thermal stresses which could cause them to fail due to fatigue. Previous studies (1-7) of neutron irradiation effects on fatigue properties of stainless steels have not produced information describing behavior under test and irradiation temperatures typical of conditions found in the cores of pressurized water reactors (PWR's). Since it has been shown (8, 9) that changes in mechanical properties of stainless alloys are directly affected by test and irradiation temperatures, the usefulness of these data for PWR evaluations is uncertain. Also, environmental studies of the effect of the water coolant on reactor material properties have been carried out only for ferritic steels (10, 11). Since it has been well established for titanium (12) and aluminum alloys (13, 14) and some steels (15) that water and water vapor environments substantially accelerate fatigue crack growth rate, knowledge of the effect of water on stainless steel is necessary for prediction of material performance in pressurized water reactor environments. In addition, some of these studies have employed relatively thick-section test specimens as compared to the thin structural core components or cladding normally used; therefore the results may not be applicable because of differences in stress state.

The purpose of this study was to determine the effect of irradiation on the fatigue properties of thin-section AISI Type 348 stainless steel under conditions of irradiation temperature, simulated coolant environments, and projected maximum fluence typical of core components in the MH-1A Army PWR. In addition, the material selected for this study was similar in composition to the cladding and structural material employed in this reactor.

EXPERIMENTAL PROCEDURE

The Type 348 stainless steel had the following chemical composition by weight percent: 0.06 C, 1.82 Mn, 0.52 Si, 0.002 P, .005 S, 18.40 Cr, 9.92 Ni, 0.05 Mo, 0.02 Cu, 0.01 Co, 0.77 Cb + Ta, 0.0005 B, and <.001 Ag. Specimen blanks were cut from a 3/4-in.-thick, hot-rolled, mill-annealed (1830°F, 999°C) plate and then machined to the dimensions shown in Fig. 1. In addition, the majority of the specimens incorporated a 90-degree (1.58 rads) face notch, 0.002-in. (0.051-mm) deep, with a 0.001-in. (0.025-mm) root radius to localize the formation of a single crack. After machining, the specimens were electropolished in an acetic-chromic acid solution. Specimens were irradiated to a fluence of 1.5×10^{21} n/cm^2 >0.1 MeV at approximately 550°F (288°C) in the Advanced Test Reactor, a pressurized

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in, but remained essentially the same as in vacuum; accordingly, in these experiments it was presumed that the partial pressure of oxygen was very low ($<10^{-7}$ torr). After the test temperature had stabilized, specimens were vibrated at resonance at a nominal frequency of 5 Hz. Changes in resonant frequency during test were automatically recorded, and the machine shut itself off after a preset value of frequency decrease had occurred.

RESULTS AND DISCUSSION

Air Environment

Pre- and postirradiation fatigue properties of Type 348 stainless steel were determined at 550°F (288°C) in dry air as a function of total strain amplitude. The results presented are mainly for notched specimens cycled in reverse bending at constant amplitudes. In the determinations of fatigue life, the criterion for failure was the development of a crack 0.018 in. (0.46 mm) long. At this point the crack was growing very fast, and complete failure was imminent. Fatigue life comparisons of unirradiated and irradiated Type 348 stainless steel in air at 550°F (288°C) are shown in Fig. 3. Irradiation increased the fatigue life by a factor of about 10 at high strains and somewhat less at intermediate strains. However, at low strains, where the postirradiation fatigue life appeared to approach an endurance limit, the increase in life became very large. Unnotched specimens that were irradiated were affected in a similar way by the radiation. Pre-irradiation comparisons of both types of specimens showed that at a strain of 0.149 percent the life of the smooth specimen was 4.6×10^5 cycles, 15 times that of the notched specimen. Comparisons of notched and smooth irradiated specimens showed an even greater difference in fatigue life. Whereas the notched specimen failed after 7.8×10^4 cycles, at 0.172 percent strain, cycling of the irradiated smooth specimen had to be discontinued after 4×10^6 cycles, with no evidence of cracking. The fatigue life for the smooth, unirradiated specimen of 4.6×10^5 cycles is much less than that which would be predicted from the Berling-Conway method of universal slopes (4), which usually produces fairly accurate estimates of low-cycle behavior at these temperatures. This result may have been due to the thin section specimens and reverse bending loading employed in these evaluations.

Information on the fatigue process was determined from changes in resonant frequency of vibration, which reflected microstructural changes and cracking occurring in the material. At low strains, the resonant frequency for the unirradiated material initially increased due to work hardening. However, for the irradiated material, no increase in frequency was observed because displacement damage had prehardened the material. The growth of the crack that developed at the notch caused the frequency to decrease. To relate frequency decrease to crack depth, a calibration curve was obtained from two unirradiated specimens; the resulting correlation between the percentage of frequency decrease and crack length is shown in Fig. 4. Crack depths were measured on the specimen edges at a magnification of 400. To check the agreement of the data for the two material conditions, the crack lengths at failure were also measured for a number of irradiated specimens and are presented as solid data points.

By applying the crack-length correlation to the continuous record of resonant frequency change during testing, the investigators obtained an indication of cracking behavior. Examples of crack growth for irradiated and unirradiated conditions at approximately equal total strains

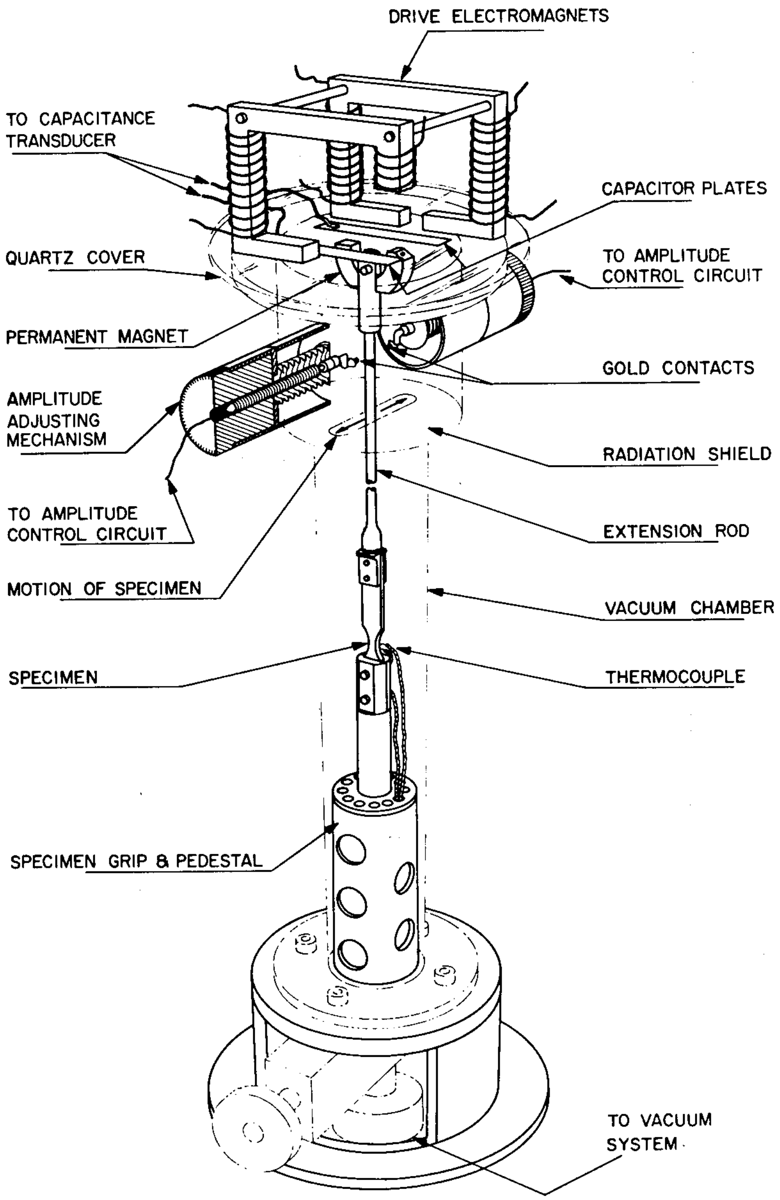

Fig. 2 — Mechanical details of the flexural-fatigue machine

environment free of atmospheric impurities, which even in small amounts can significantly affect fatigue life, particularly at low strains (18). The procedures were similar for the water-vapor tests; however, the vacuum system was throttled down before the distilled and degassed water was admitted to the test chamber through a variable conductance valve. The water-vapor pressure was controlled at 7 torr by adjustments of the throttle valve, variable conductance valve, and of the internal pressure of the water reservoir. In previous water-vapor experiments (19) the pressure of oxygen did not change when water vapor was leaked

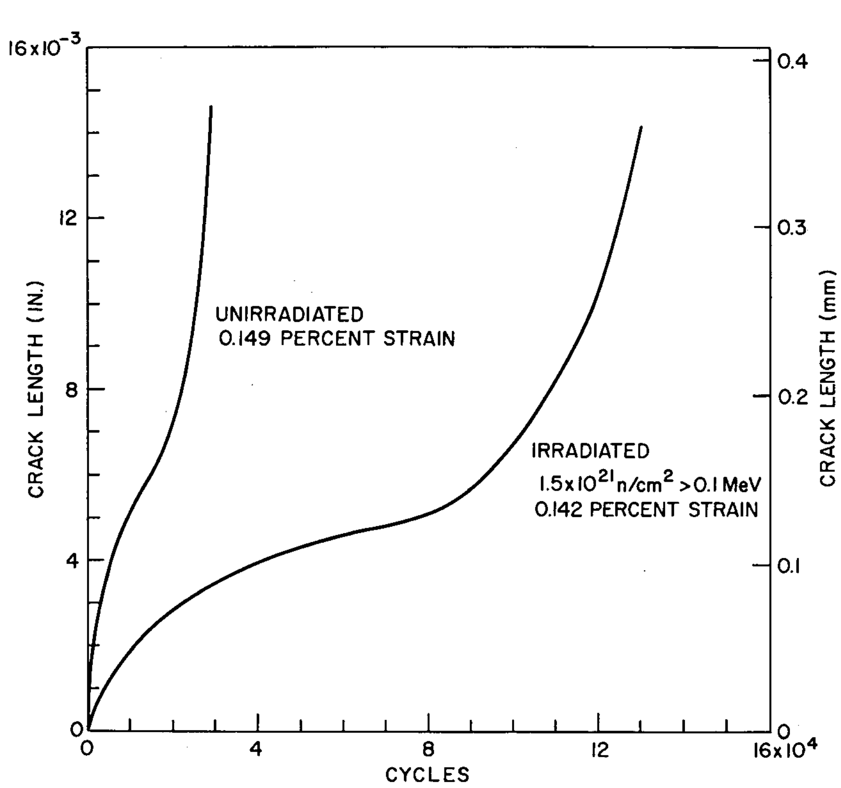

Fig. 5 — Crack growth in irradiated and unirradiated Type 348 stainless steel during fatigue at 550° F (288° C)

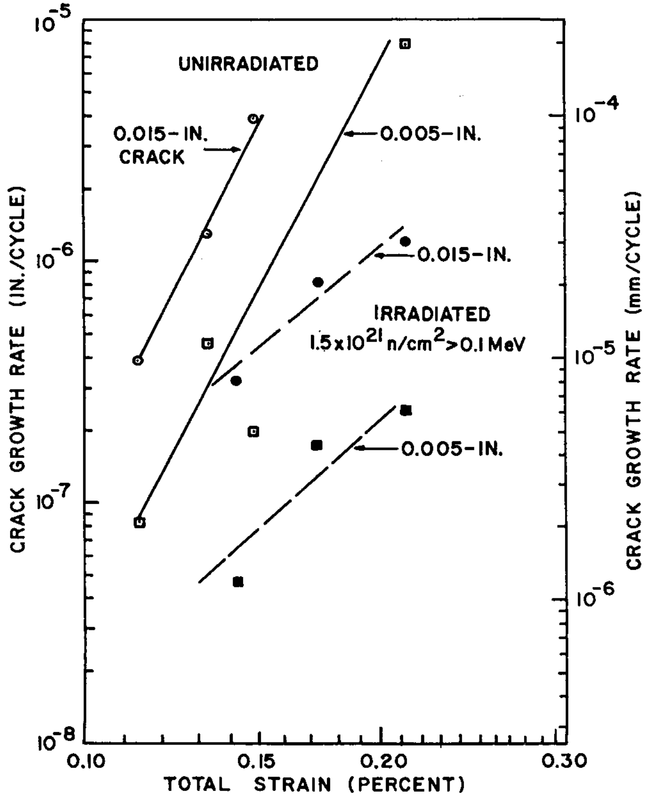

Fig. 6 — Effect of irradiation on crack-growth rate in Type 348 stainless steel for crack lengths of 0.005 in. (0.13 mm) and 0.015 in. (0.38 mm) as a function of total strain. Dashed lines represent data for irradiated specimens.

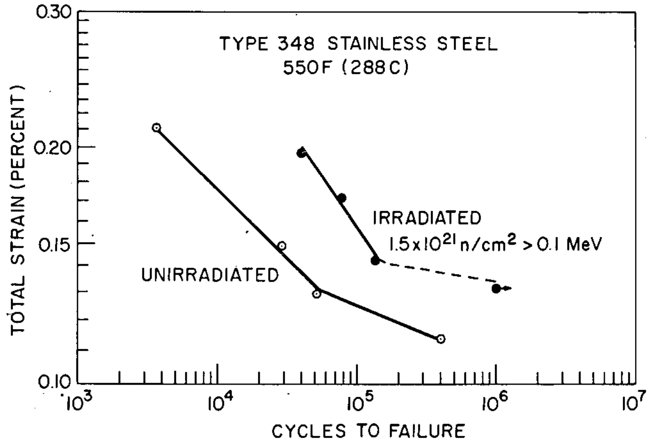

Fig. 3 — The effect of total strain per half cycle on the fatigue life of irradiated and unirradiated Type 348 stainless steel at 550° F (288° C) in dry air

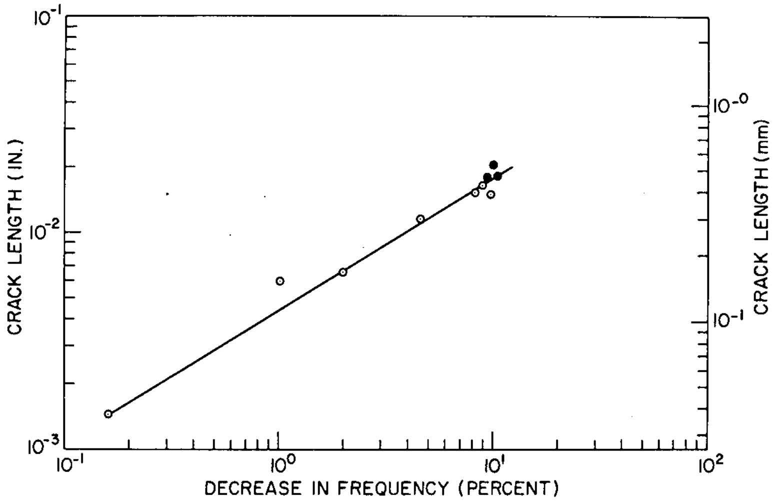

Fig. 4 — Correlation of crack length and percent decrease in frequency during fatigue of Type 348 stainless steel at 550° F (288° C). The three black circles indicate failure points for irradiated specimens.

of 0.142 and 0.149 percent are shown in Fig. 5. The plot of crack length *versus* cycles indicates that the time to crack initiation is extremely short and that crack lengths increased rapidly initially and also at the terminal stages. In all cases, irradiation increased the number of cycles required for a crack to grow to a particular length.

The effect of irradiation on the crack-growth rates of 0.005-in. (0.13-mm) and 0.015-in. (0.38-mm) cracks as a function of strain is shown in Fig. 6. Irradiation has decreased crack-growth rates at both crack lengths. Also, the increase in crack-growth rate with increase in strain

Water-Vapor Environment

Comparisons have also been made between irradiated and unirradiated notched specimens to determine the effect of water vapor on fatigue life. Fatigue life data for these steels in vacuum and water vapor are plotted along with the previously described air results. In Fig. 8 it can be seen that irradiation has produced an increase in fatigue life in vacuum ($<1 \times 10^{-5}$ torr) of about a factor of six. In the unirradiated condition, water vapor (at 7 torr) reduces fatigue life, but not as much as air. After irradiation, the steel's fatigue life is about the same in water vapor as it is in vacuum. On the basis of the test results in vacuum, it is apparent that the response of the material to water vapor has been altered by irradiation. The relative effects of air and water vapor on the lives of unirradiated and irradiated Type 348 stainless steel specimens are shown in Table 1, where the ratio of the fatigue life in vacuum to that in gas has been tabulated. Irradiation appears to have decreased the sensitivity of the material to water vapor and also, although somewhat less, to air.

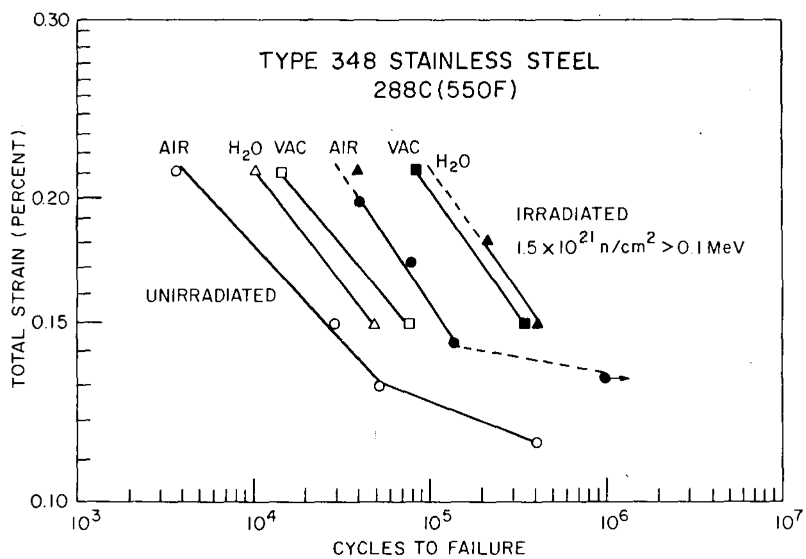

Fig. 8 — The effect of total strain per half cycle on the fatigue life of irradiated and unirridiated Type 348 stainless steel at 550°F (288°C) in dry air (2-lb gauge), water vapor (7 torr), and vacuum ($<1 \times 10^{-5}$ torr)

Usually, for the assessment of structural performance of reactor materials, property evaluations are conducted in air, the easiest environment for testing. Obviously, the properties should be evaluated as nearly as possible under the appropriate service conditions of is less after irradiation. These effects are reflected in the fatigue life data shown in Fig. 3. It therefore appears that increases in irradiated fatigue life are due primarily to lower crack-growth rates since times to crack initiation were insignificant under the conditions of this study.

Failure of a nuclear structural component will probably originate in an undetected defect which grows under cyclic loading to the critical size for castastrophic failure. Therefore, information on the rate at which this small flaw may grow is necessary for designing a structure and assuring its safe operation. By plotting a series of strain-*versus*-cycles curves, each representing a particular crack length, a designer can predict the number of cycles to which a material containing a flaw of a particular size can be subjected before failure. An example of this approach is shown in Fig. 7. In terms of the engineering parameter of total strain, the plot should provide a means for assessing the performance of a structure containing a crack of 0.002-in. (0.05-mm) or larger. The plot shows that at comparable values of strain, more cycles are required to extend a crack a particular length increment in the irradiated material than in the unirradiated.

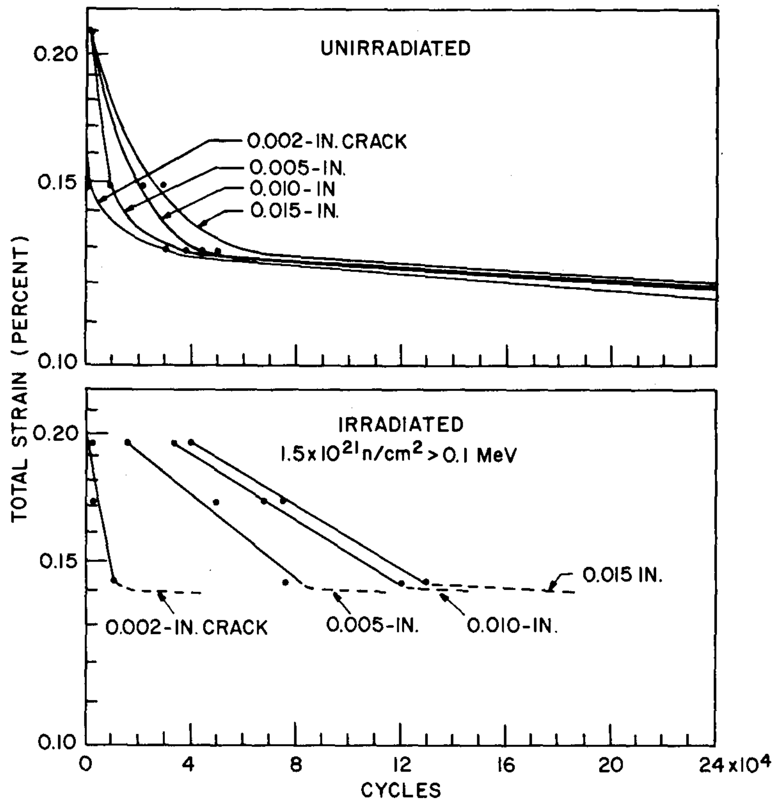

Fig. 7 — Cycles required to grow cracks of various sizes as a function of strain for irradiated and unirradiated Type 348 stainless steel

decreases life (3-7). This suggests that fatigue behavior may also be controlled by the mechanisms of hardening and helium embrittlement. It is therefore reasoned that at the low temperatures of this study, irradiation-induced increases in yield strength may reduce the size of the plastic zone at the tip of the crack and consequently lower the crack-growth rates. It has been shown for an alloy steel (21) that the smaller the plastic zone the lower the resulting crack-growth rates. The results of this study are in agreement with results of studies at room temperature which show an increase in fatigue life due to irradiation. However, at 550° F (288° C) the magnitude of the increase in fatigue life appears to be generally larger than that observed by others at room temperature.

CONCLUSIONS

From the findings of this study the following conclusions may be drawn concerning Type 348 stainless steel:

1. When this steel is irradiated to a fluence of 1.5×10^{21} n/cm^2 >0.1 MeV at 550° F (288° C) and tested at this temperature, its fatigue life is increased.

2. The effect of irradiation on life in air is greatest at low total strains, where the curves of total strain *versus* cycles to failure tend to flatten out sooner for the irradiated material. These data indicate that in the high-cycle fatigue range, irradiation would have a very great effect on life, and thus the fatigue performance of nuclear structures would be expected to be greatly enhanced under the temperature conditions of this study.

3. The fatigue life of this steel, when irradiated, was longest in vacuum and water vapor, and least in air. Accordingly, properties in air should give a conservative prediction of material performance in actual service.

4. The effect of air and water vapor on the steel appears to be reduced after irradiation.

5. Crack-growth rates are lower in the irradiated than in the unirradiated Type 348 stainless steel. Under the conditions of this study, time to crack initiation was extremely short and was not measurably affected by irradiation.

6. The increase in fatigue life of irradiated specimens is due primarily to the effect of irradiation on the growth rates of short cracks (less than 20 percent of specimen thickness).

7. The dependency of crack-growth rate on total strain is decreased after irradiation.

8. At 550° F (288° C) the irradiated fatigue behavior appears to be dominated by increases in strength caused by atom displacement damage.

ACKNOWLEDGMENTS

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Table 1
The Relative Effects of Air and Water Vapor on the Pre- and Postirradiation Fatigue Life of Type 348 Stainless Steel at 550°F (288°C) and 0.18% Total Strain

Gaseous Species	$\frac{N_{vac}}{N_{gas}}$	
	Preirradiation	Postirradiation
air	3.3	2.5
H_2O	1.5	1.0

N — cycles to failure

temperature, stress, irradiation, and coolant environment. To what extent the water-vapor data represent the high-temperature, pressurized-water environment found in reactors is not certain; however, data in the literature do indicate a correspondence between vapor and liquid. In a study (20) of a high-strength steel, H-11, tested in a series of relative humidities and in liquid water at room temperature, equal crack-growth rates were observed for both physical states when the relative humidity of the water vapor was above 60%. It was reasoned that this equivalence of crack-growth rate was due to water vapor condensation at the crack tip. In the present study, evaluations have been conducted in water vapor at a partial pressure of water sufficient to saturate crack surfaces at all crack-growth rates.

In view of the indicated evidence, this approach should produce material behavior similar to that which would be found for high-temperature pressurized water. If this is the case, then results from this study indicate that the fatigue performance of reactor components of Type 348 stainless steel should not be reduced by water-cooled reactor environments. The water environment has already been shown to increase fatigue crack-growth rates in ferritic steels at low test frequencies (11); accordingly, the possibility of frequency effects due to the application of strain at different cyclic rates should be considered.

Studies in the literature of the effect of irradiation on the tensile properties of austenitic stainless steels relate ductility losses to hardening and embrittlement processes in the material. It has been shown that at low test and irradiation temperatures, irradiation-produced point-defect clusters harden the metal matrix and reduce ductility. However, above about $T_m/2$ (~1040°F, ~560°C for 348 stainless), displacement damage anneals out, and ductility is generally thought to be controlled by helium embrittlement. (Helium is produced at high fluences by transmutation reactions in elements such as nickel.) Results of previous fatigue studies of austenitic stainless steels at fluences of $<10^{22}$ n/cm^2 can be grouped in approximately those same temperature ranges. At low temperatures, irradiation generally increases fatigue life (1, 2, 7) whereas at and above 932°F (500°C), irradiation

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