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**MECHANICAL PROPERTIES OF IRRADIATED
TYPE-304 AND TYPE-316
STAINLESS STEEL IN FAST REACTORS**

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BY J. H. Kahn
DATE 2/18/70

BY
F.A. COMPRELLI
H.J. BUSBOOM
M.F. GEBHARDT

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BREEDER REACTOR DEVELOPMENT OPERATION

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MECHANICAL PROPERTIES OF IRRADIATED TYPE-304
AND TYPE-316 STAINLESS STEEL IN FAST REACTORS

by
F. A. Comprelli
H. J. Busboom
M. F. Gebhardt

Approved:

W. E. Baily
W. E. Baily, Project Engineer
Fast Ceramic Reactor Program, PA-10

Approved:

E. L. Zebroski
E. L. Zebroski, Manager
Sodium Reactor Technology

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SUMMARY

All data published to November 1968 on the mechanical properties of irradiated Type-304 and Type-316 stainless steel have been compiled and analyzed, and property limits are suggested for current use by designers. Revision of these limits in the light of new data is to be expected.

For use at total fluences above 10^{22} nvt, the following property limits are suggested:

- Yield stress at 0.2% offset:
 - 44,000 psi at 500°C
 - 31,000 psi at 600°C

20,000 psi at 700°C

- Ultimate stress:
 - 58,000 psi at 500°C
 - 46,000 psi at 600°C
 - 30,000 psi at 700°C
- Uniform elongation: less than 1%
- Total elongation: less than 1%
- Minimum creep rate: unchanged to slightly increased by irradiation
- Time to rupture: reduced significantly by irradiation
- Creep strain to fracture: as low as 1%

1. INTRODUCTION

Loss in ductility of stainless alloys at high temperature as a result of neutron irradiation was recognized as a major problem since 1962 and a major effort was initiated to evaluate the performance limits. Neutron exposures which correspond to projected economic fuel burnups in fast reactors have been estimated about 3×10^{23} nvt. Thus far, the highest exposure achieved is 8×10^{22} nvt (≥ 1.0 MeV). The major portion of published data was gathered in experiments conducted in Dounreay Fast Reactor (DFR) and EBR-II. Very little effort has been made prior to this study to collect the results obtained and relate the residual mechanical properties of irradiated steels to fuel element designs.

The information reviewed in this study represents the published efforts of General Electric—BRDO, Pacific Northwest Laboratory, Oak Ridge National Laboratory, Argonne

National Laboratory, General Electric—NSP, UKAEA—Culchetch, Harwell, and Dounreay, and the Centre d'Entudes Nucleaires de Saclay, France. As such, the data were gathered under a broad range of irradiation conditions and post-irradiation test parameters. The interpretation of the data was focused on the design needs at the operating conditions of an LMFBFR (Liquid Metal Fast Breeder Reactor). Where applicable, uncertainties in existing data are noted.

This report deals with the analysis of information which has been published, judgment of the metallurgical trends, and the suggestions of performance limits for the materials of interest, Type-304 and Type-316 stainless steel for LMFBFR's. These suggested limits should be reviewed and modified as new data are gathered and understanding of the phenomenon has progressed.

2. DISCUSSION OF THE DATA

The effect of irradiation upon mechanical properties of Type-304 and Type-316 stainless steel was considered from three viewpoints: tensile data, burst properties and creep-rupture behavior, and suggested limits for use by designers were made for each.

For the most part, only data at fluences greater than 10^{22} nvt have been considered. Because no data are available at fluences greater than 10^{23} nvt, extrapolation to the design conditions of LMFBFR's must be considered tenuous. However, it is reasonable to assume that the minimum property levels, apparent at fluences less than 10^{23} nvt, will not be improved with continuing exposure.

2.1 TENSILE PROPERTIES

Available data (Ref. 1-8) are primarily from irradiations in EBR-II and DFR and all at either low temperatures during irradiation, relatively low fluence or both. The available data for Type-304 and Type-316 stainless steel in the annealed condition are plotted in Figures 1 through 4.

These figures show the variation of yield stress, ultimate tensile stress, uniform elongation* and total elongation** as a function of test temperature. On these curves, the data are limited to Type-304 and Type-316 stainless steel which had been irradiated to total fluences*** greater than 10^{22} nvt at irradiation temperatures from 250°C (482°F) to 670°C (1238°F). In Figures 5 through 8, the primary tensile properties are plotted as a function of total fluence at test temperatures of 600°C (1112°F) and 700°C (1292°F).

* Uniform elongation is defined as the strain to the onset of necking or plastic instability.

** Total elongation is defined as the strain at rupture.

*** All EBR-II total fluence values in this report are based upon Argonne National Laboratory physics calculations (power normalization) reported in the EBR-II Operations Quarterly Reports. Current estimates indicate that these reported values should be corrected by 1/1.2. This correction has not been made in the EBR-II data reported here.

SYMBOL	MATERIAL	TOTAL FLUENCE ($\times 10^{22}$)	REFERENCE	IRRADIATION TEMPERATURE (°C)	STRAIN RATE (min^{-1})
○	304	2.0	(1)	538	0.02
●	304	1.7	(2)	590 ± 80	0.002
⊗	304	2.0	(3)	380	0.02
▽	304	3.4	(4)	590 ± 80	0.02
△	316	3.6	(7)	250-350	0.095
▲	316L	3.4	(4)	590 ± 80	0.02
304 AND 316 UNIRRADIATED STANDARDS

Figure 1. 0.2% Yield Stress Versus Test Temperature

SYMBOL	MATERIAL	TOTAL FLUENCE ($\times 10^{22}$)	REFERENCE	IRRADIATION TEMPERATURE ($^{\circ}\text{C}$)	STRAIN RATE (min^{-1})
○	304	2.0	(1)	538	0.02
●	304	1.7	(2)	590 ± 80	0.002
⊗	304	2.0	(3)	380	0.02
▽	304	3.4	(4)	590 ± 80	0.02
△	316	3.6	(7)	250-350	0.095
▲	316L	3.4	(4)	590 ± 80	0.02
▨	304 AND 316 UNIRRADIATED STANDARDS

Figure 2. Ultimate Tensile Stress Versus Test Temperature

SYMBOL	MATERIAL	TOTAL FLUENCE ($\times 10^{22}$)	REFERENCE	IRRADIATION TEMPERATURE ($^{\circ}\text{C}$)	STRAIN RATE (min^{-1})
○	304	2.0	(1)	538	0.02
●	304	1.7	(2)	590 ± 80	0.002
⊗	304	2.0	(3)	380	0.02
▽	304	3.4	(4)	590 ± 80	0.02
△	316	3.6	(7)	250-350	0.095
▲	316L	3.4	(4)	590 ± 80	0.02

Figure 3. Uniform Elongation Versus Test Temperature

SYMBOL	MATERIAL	TOTAL FLUENCE ($\times 10^{22}$)	REFERENCE	IRRADIATION TEMPERATURE ($^{\circ}\text{C}$)	STRAIN RATE (min^{-1})
○	304	2.0	(1)	538	0.02
●	304	1.7	(2)	590 ± 80	0.002
⊗	304	2.0	(3)	380	0.02
▽	304	3.4	(4)	590 ± 80	0.02
△	316	3.6	(7)	250-350	0.095
▲	316L	3.4	(4)	590 ± 80	0.02
304 AND 316 UNIRRADIATED STANDARDS

Figure 4. Total Elongation Versus Test Temperature

SYMBOL	MATERIAL	TOTAL FLUENCE ($\times 10^{22}$)	REFERENCE	IRRADIATION TEMPERATURE (°C)	STRAIN RATE (min^{-1})
○	304	2.0	(1)	538	0.02
●	304	1.7	(2)	590±80	0.002
△	304	3.4	(4)	590±80	0.02
▽	316	0.8	(6)	482	0.002
□	316	3.6	(7)	250 - 350	0.095
▼	M316	0.1	(8)	250 - 350	0.095
■	316	0.2	(5)	525	N. A.
△	316L	3.4	(4)	590±80	0.02
▨	316L	0.5 - 0.7	(8)	250 - 350	0.095
▩	M316	0.1 - 10.0	(7)	250 - 350	0.095

Figure 5. 0.2% Yield Stress Versus Total Fluence at 600° and 700°C

Figure 6. Ultimate Tensile Stress Versus Total Fluence at 600° and 700°C

SYMBOL	MATERIAL	TOTAL FLUENCE (×10 ²²)	REFERENCE	IRRADIATION TEMPERATURE (°C)	STRAIN RATE (min ⁻¹)
○	304	2.0	(1)	538	0.02
●	304	1.7	(2)	590±80	0.002
△	304	3.4	(4)	590±80	0.02
□	316	0.8	(6)	482	0.002
▲	316	3.6	(7)	250 - 350	0.095
■	316	0.2	(5)	525	N. A.
▲	316L	3.4	(4)	590±80	0.02

Figure 7. Uniform Elongation Versus Total Fluence at 600° and 700°C

Figure 8. Total Elongation Versus Total Fluence at 600°C and 700°C

These figures show that significant increases in yield strength can be expected at test temperatures up to 760°C (1400°F) for all conditions of irradiation reported. However, this strengthening should not become an essential requirement for effective design. Ultimate tensile strengths were not found to increase in the same manner, and at the higher temperatures were lower than unirradiated values, which is a reflection of the sharp reduction in post-irradiation elongation. The elongation data are widely scattered and reflect the sensitivity of this measurement both to irradiation conditions (temperature, fluence, and neutron spectrum) and post-irradiation test parameters (temperature and strain-rate). However, all elongations were reduced by irradiation. Elongation values of less than 10% were common, with a considerable amount of the uniform elongation data falling below 2%. A trend of decreasing elongation with increasing fluence is apparent.

The effect of strain rate on uniaxial tensile properties was also examined (Ref. 4) for Type-304 and Type-316 stainless steel. The results are shown in Figure 9 for tests at 1300°F after irradiation in EBR-II to 3.4×10^{22} nvt total. At the slower strain rates, strength is appreciably reduced,

probably associated with a change in deformation mechanisms. The total elongation does not seem to be significantly affected over the range investigated. However, extrapolation of this behavior to lower strain rates and biaxial loading patterns may be misleading as discussed in the later section on creep properties.

For comparative purposes, tensile data for cold-worked material are listed in Table I. The PNL studies (Ref. 2) were on Type-304 stainless steel which had been cold-rolled 25% and then irradiated to 1.7×10^{22} nvt total in EBR-II at $590 \pm 80^\circ\text{C}$. The UK data (Ref. 7) were generated on M316* which had been prestrained 17% (probably in tension) prior to irradiation in DFR at 250-350°C to $\sim 5 \times 10^{22}$ nvt total. The testing strain rates were 0.002/min for PNL and 0.095/min for the UK, which accounts for some of the difference in the data.

* M316 has the nominal composition: 0.04C, 1.0 Mn, 0.2 to 0.5 Si, 15.5-17 Cr, 14-15.5 Ni, 1.5-1.9 Mo, 0.002 B, 0.005 Al, 0.03V.

Figure 9. Strain Rate Dependence of Tensile Properties for Type-304 and Type-316 at 1300°F After Irradiation to 3.4×10^{22} nvt Total

Table 1
DATA ON COLD-WORKED MATERIAL

Test Temperature [°C(F)]	PNL (Type 304)*				UK (Type M316)**		
	0.2% Yield Stress (ksi)	Ultimate Tensile Strength (ksi)	Uniform Elongation (%)	Total Elongation (%)	0.2% Yield Stress (ksi)	Ultimate Tensile Strength (ksi)	Total Elongation (%)
500 (932)	79.2	87.2	3.7	5.9	94	96	11
600 (1112)					56	64	19
650 (1202)	54.9	62.3	1.7	3.3	44	54	18
700 (1292)					40	47	11
750 (1382)	27.7	35.1	1.3	2.5			

2.2 BURST PROPERTIES

The available data, shown in Figure 10, were gathered at ANL (Ref. 9) on Type-304L stainless steel driver fuel cladding from EBR-II, and at GE-BRDO (Ref. 10) on Type-304 capsule tubing used for fuel tests in EBR-II. Both tubes were unstressed during irradiation. During the post-irradiation testing, typical rupture times were ~ 2 minutes for the ANL specimens to as high as 40 minutes for the GE specimens. This difference in pressurizing rate did not significantly affect the consistency of results.

As can be seen in Figure 10, irradiation in EBR-II to total fluence levels as low as 0.5 to 2.5×10^{22} nvt does significantly affect the burst properties of Type-304 tubing. Although data are not available, similar behavior can be expected for Type-316 tubing. Significant increases in strength were observed for the irradiated material at temperatures below 1300°F, with corresponding reductions in diametral fracture strain. At test temperatures above 800°F, fracture strains of less than 2% were typical, as compared to unirradiated values of 10% to 20%.

2.3 CREEP BEHAVIOR

There has been a fair amount of creep work done, both in-pile and post-irradiation, in conjunction with the behavior of Type-304 stainless steel in thermal reactors. (Ref. 11) The data from these studies are presented parametrically in Figures 11 and 12. The general conclusions reached in these studies were: (1) little, if any, significant difference exists between the secondary or mini-

um creep rate of irradiated and unirradiated specimens, and (2) a significant effect of neutron irradiation is observed in the reduction of creep rupture life and, correspondingly, fracture ductility.

Comparable information for a fast reactor environment is limited and therefore, a sweeping analysis is not possible. However, the fast reactor data available for both Type-304 and Type-316 stainless steel can be evaluated against the trends previously established for irradiation of Type-304 stainless steel in thermal reactors.

The effect of fast reactor exposure on the minimum or secondary creep rate is shown in Figure 13. (Ref. 12-14) Except for the highly-stressed specimens, all values have fallen within the trend band established by the thermal irradiation data. In general, fast neutron bombardment had no effect or caused minor perturbations from thermal irradiation data. There are deviations from this trend, such as the French data (Ref. 12) and the high-stress PNL data, (Ref. 13) which showed an increase in the secondary creep rate after irradiation. All of the irradiated data were determined from relatively short-lifetime tests, and therefore the extrapolation of these data to lower strain rates is tenuous as discussed later.

For comparative purposes, UK data (Ref. 14) on 20% cold-worked Type-316 stainless steel are included in Figure 13. In addition to being considerably stronger than the annealed material, the cold-worked alloy showed an increase in secondary creep rate after irradiation in DFR.

The effect of fast neutron bombardment on rupture life (Ref. 12-14) is shown in Figure 14. As can be seen, a reduction in rupture life was observed for all annealed and cold-worked specimens. Although all data for annealed material do not fall within the thermal irradiation trend band, a comparable and consistent trend is evident for the fast reactor data.

* PNL—Pacific Northwest Laboratories. Cold-rolled to 25% reduction, then irradiated to 1.7×10^{22} nvt total in EBR-II at $590 \pm 80^\circ\text{C}$. Testing strain rate: 0.002/min in EBR-II.

** UK—United Kingdom. Prestrained 17%, then irradiated to $\sim 5 \times 10^{22}$ nvt total in DFR at 350°C . Testing strain rate: 0.095/min.

Figure 10. Tubing Burst Properties of 304 as a Function of Test Temperature After Irradiation in EBR-II

Figure 11. Parametric Relationship Between Applied Stress and Temperature - Compensated Minimum Creep Rate (Thermal Reactor Data) - Type-304

Figure 12. Parametric Relationship Between Applied Stress and Temperature - Compensated Time-to-Rupture (Thermal Reactor Data) - Type-304

SYMBOL		MATERIAL	TOTAL FLUENCE ($\times 10^{22}$)	TEST TEMPERATURE ($^{\circ}\text{C}$)	REFERENCE
IRRADIATED	UNIRRADIATED				
▲	No Data	316	1.0-2.2	650	(14)
■	No Data	316	0.4-1.6	650	(14)
▼	▽	316 C.W.	3.0	650	(14)
●	○	304	1.4	677-760	(13)
●	○	316	1.6	700	(12)

NOTE: C. W. - Cold Worked

Figure 13. Parametric Relationship Between Applied Stress and Temperature - Compensated Minimum Creep Rate for Fast Reactor Data

SYMBOL		MATERIAL	TOTAL FLUENCE ($\times 10^{22}$)	TEST TEMPERATURE (°C)	REFERENCE
IRRADIATED	UNIRRADIATED				
▲	△	316	1.0-2.2	650	(14)
■	□	316	0.4-1.6	650	(14)
▼	▽	316 C.W.	3.0	650	(14)
●	○	304	1.4	677-760	(13)
●	○	316	1.6	700	(12)

NOTE: C. W. - Cold Worked

Figure 14. Parametric Relationship Between Applied Stress and Temperature Compensated Time-to-Rupture for Fast Reactor Data

Data on rupture strain for tests at low strain rates are also uncommon for materials irradiated in fast reactors. Uniaxial tests for annealed Type-304 (Ref. 13) and Type-316 (Ref. 12) both showed rupture strains of 6% to 10% and higher at 1250° to 1400°F after irradiation to $\sim 1.5 \times 10^{22}$ nvt total. However, biaxial tests on annealed tubing irradiated in DFR showed significantly lower rupture strains at 1200°F (650°C). Type-316L tubing dropped from unirradiated values of 5% to 17% and 2% to 5% after 1.35×10^{22} nvt (Ref. 15) and leveled off at 2% after 3.5×10^{22} nvt (Ref. 16). Type-316 tubing (Ref. 14) has shown reductions from 11% to 3% after 0.4 to 1.6×10^{22} nvt and from 6% to 1% or 2% after 1 to 2.2×10^{22} nvt. The rupture

strains of the 20% cold-worked Type-316 tubing ranged from 0.9% to 2.5% at 550°C (1022°F) after 3×10^{22} nvt, which were comparable to unirradiated data for this material. These observations are consistent with a similar analysis (Ref. 17) made on annealed Type-304 and Type-316 stainless steel irradiated in thermal reactors, in which rupture strains predicted from uniaxial tests were an order of magnitude greater than extrapolations based on biaxial data (9% to 12% versus 0.8% to 1.2%).

On the basis of these observations, it seems advisable that a strain limit of 1% be established for very low strain-rate conditions.

3. SUGGESTED LIMITS

On the basis of these data, reasonable limits can be suggested for first round design analysis. The values in Figures 15 and 16 represent the probable minimum tensile properties for annealed Type-304 and Type-316 stainless steel which can be expected after irradiation in a fast reactor environment as a function of test temperature and total fluence. The effect of temperature during irradiation could not be clearly ascertained from the pattern of data,

but there are indications that lower irradiation temperatures are associated with greater increases in strength.

Parametric design curves for minimum creep rate and rupture lifetime are suggested in Figure 17 for Type-304 and Type-316 stainless steel after irradiation to total fluences greater than 10^{22} nvt. These curves represent the minimum level of performance as indicated by data published to date.

Figure 15. Recommended Tensile Properties Above Total Fluences of 10^{22} nvt as A Function of Test Temperature - Type-304 and -316 Stainless Steel

Figure 16. Recommended Tensile Properties as a Function of Total Fluence at 600° and 700°C - Type-304 and Type-316 Stainless Steel

Figure 17. Parametric Design Curves for Minimum Creep Rate and Time-to-Rupture in Fast Reactor Environment at Fluences Above 10^{22} nvt, Type-304 and Type-316 Stainless Steel

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