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Simulated Transient Behavior of HT9 Cladding

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SIMULATED TRANSIENT BEHAVIOR OF IRRADIATED HT9 CLADDING

ABSTRACT: Simulated transient tests were performed on sections of HT9 fast-reactor fuel pin cladding irradiated to a fast fluence of nearly 16×10^{22} n/cm^2 at temperatures ranging from 370 to 620°C. After removing fuel, these specimens were internally pressurized and heated at one of several constant rates (0.56, 5.6, or 110°C/s) until specimen failure occurred.

A slight reduction of strength was observed in irradiated cladding, particularly at 110°C/s, when compared with transient results from unirradiated HT9 control specimens; however, this strength reduction did not correlate with either fluence or irradiation temperature. A small reduction of ductility was also observed for irradiated cladding failing at temperatures above 800°C at the lower heating rates (0.56 or 5.6°C/s); irradiated cladding was generally more ductile at 110°C/s than unirradiated HT9 cladding.

The HT9 cladding results were compared with similar transient data obtained previously from 20% Cold-Worked Type 316 Stainless Steel (316 SS) cladding. In the unirradiated state, this austenitic cladding is stronger and less ductile than HT9 cladding. However, the 316 SS cladding undergoes a significant loss of strength and ductility during irradiation when in contact with oxide fuel, by a mechanism labeled the fuel adjacency effect (FAE). The FAE is believed to be liquid metal embrittlement from fission products. The HT9 fuel pin cladding remained as strong or stronger than the 316 SS cladding when irradiated in contact with fuel, showing no evidence of the FAE up to the high fluences reported here. The ductility of the irradiated HT9 fuel pin cladding remained significantly greater than that of irradiated 316 SS cladding.

KEY WORDS: radiation, irradiation, nuclear fuel cladding, transient, mechanical properties, strength, ductility, HT9

Introduction

The martensitic stainless steel HT9 is being investigated for advanced cladding and duct applications in liquid metal reactors (LMR). It is a promising alloy because of its high resistance to swelling,[1] good resistance to irradiation creep,[1,2] and low thermal expansion coefficient. The nominal composition is 12Cr-1Mo-0.5Ni-0.6Mn-0.2Si-0.5W-0.3V-0.2C. The alloy must be used in a tempered condition. The tempering is usually conducted at 675 to 780°C in order to relieve the internal stresses in the martensite. Fuel pins constructed with HT9 cladding were irradiated to a peak burnup of 13.5 at% in the Experimental Breeder Reactor-II (EBR-II).[3] Postirradiation measurements in cladding sections confirmed the low swelling behavior of this alloy. The HT9 cladding is now being used for a number of subassemblies in the Fast Flux Test Facility (FFTF).

Design and safety analyses of LMRs require understanding fuel pin responses to a wide range of off-normal events. In a loss-of-flow or over-power transient, the temperature of the cladding is rapidly increased above its steady state service temperature. Modelling of the fuel pin transient behavior requires knowledge of the cladding mechanical behavior during the stress and thermal conditions encountered in transient events. Simulated transient tests have been performed on sections of fuel pin cladding and a large data base has been established for 20% Cold Worked 316 Stainless Steel (316 SS).[4-11] Results obtained for both unirradiated and irradiated HT9 cladding in a variety of simulated transients are presented in this paper.

Experimental Procedure

Simulated transient tests were conducted on cladding sections with the Fuel Cladding Transient Test (FCTT) system.[5] Tests were conducted by pressurizing the specimen to a predetermined value and then increasing the temperature at a fixed rate until rupture occurred. Specimen heating was achieved with an induction generator and the diameter of each specimen was monitored during the test with an extensometer.

Samples were obtained from fuel pins irradiated in EBR-II and FFTF and from cladding sections irradiated in a materials experiment in EBR-II. Control tests were also conducted in unirradiated samples. Irradiated specimens were prepared from 63 mm long sections cut from the fuel pins. Essentially all fuel was removed by mechanical drilling, which left only minor amounts of fuel on the cladding inner surface. A fused silica filler was placed inside the specimen to reduce the gas volume. End fittings were welded onto the specimen providing a gas inlet and specimen positioning capability. Table 1 summarizes the sample histories and geometries.

Tests were conducted at heating rates of 0.56, 5.6, or 110°C/s. Data obtained included failure temperature, strain, and hoop stress calculated from the internal pressure and the initial specimen geometry. Temperature, pressure, and specimen diameter were monitored throughout (several hundred times) a given test. A "uniform" failure strain was determined after the test was over by profiling the specimen diameter along its axial length in approximately 2.5-mm increments. The immediate burst region (although usually profiled) was excluded from the uniform strain determination.

Test Results

A plot of hoop stress versus failure temperature for both irradiated and unirradiated HT9 cladding tested at 0.56°C/s is shown in Figure 1. At this heating rate, only results from a single heat and geometry of HT9 cladding have been obtained. The irradiated cladding fluence range is from approximately 1 to 2.8 x 10^{22} n/cm^2, while irradiation temperatures were from 370 to 610°C. For comparison purposes, Figure 1 also contains 316 SS cladding trends at this lower heating rate.

A plot of hoop stress versus failure temperature for both irradiated and unirradiated HT9 cladding tested at 5.6°C/s is shown in Figure 2. At this heating rate, a variety of HT9 cladding specimen types were tested. The cladding irradiation fluences ranged from 1.2 and 15.6 x 10^{22} n/cm^2, and irradiation temperatures ranged from 370 to 615°C.

TABLE 1--HT9 cladding variables.

Heat	Cladding[a] Type	Anneal[b] Temperature °C (minutes)	OD, mm (mils)	ID, mm (mils)	Reactor	Experiment	Fluence, 10^{22} n/cm^2 T_i, °C Ranges
91354	1	1038 (1)	58.4 (230)	50.8 (200)	EBR-II[c]	P44	1-9.2 370-615
91354	1	1038 (1)	58.4 (230)	50.8 (200)	EBR-II	AAXV[d]	5 370-410
84425	2	W1049 (3.2)	68.6 (270)	57.4 (226)	N/A[e]	N/A	Unirradiated
84425	3	1074 (3)	68.6 (270)	57.9 (228)	FFTF[f]	AC01	5-15.7 370-574

[a]Cladding Type assigned arbitrarily according to heat and geometry; property differences between "types" were expected to be minor, as was shown.
[b]All tube lots were tempered at 760°C for 30 minutes.
[c]EBR-II = Experimental Breeder Reactor-II.
[d]Unfueled.
[e]N/A = not applicable.
[f]FFTF = Fast Flux Test Facility.

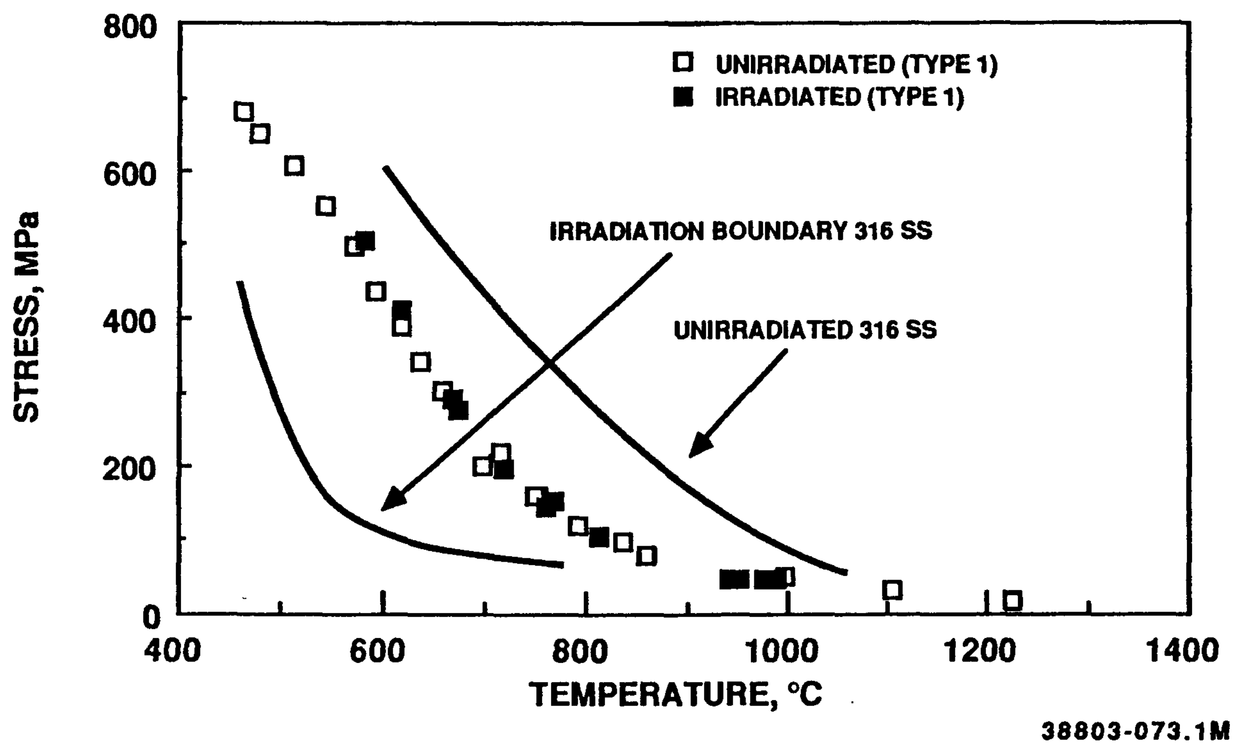

FIG. 1--HT9 0.56°C Transient Strength Results.

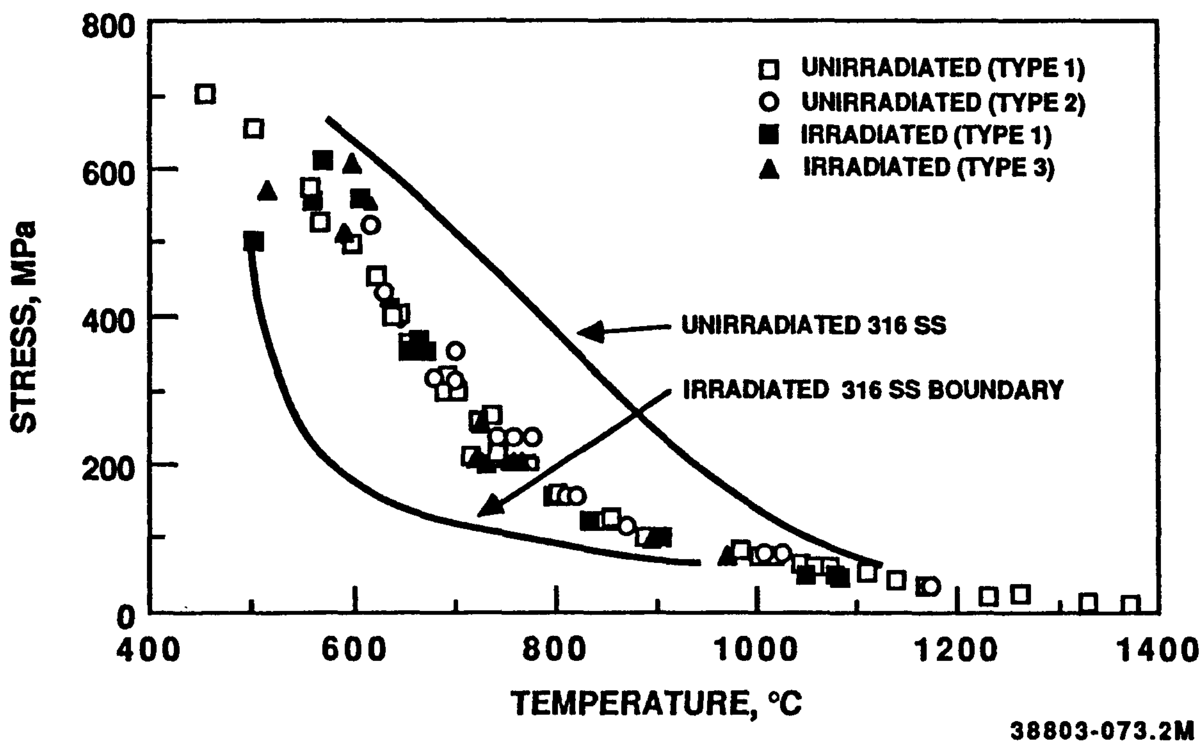

FIG. 2--HT9 5.6°C/s Transient Strength Results.

In Figure 3, FCTT hoop stress results obtained from HT9 cladding at 110°C/s are plotted as hoop stress versus failure temperature. Cladding irradiations were at temperatures between 370 to 615°C, and fluences ranged from 1.1 to 15.7 x 10^{22} n/cm^2.

Ductility data from HT9 cladding tested at 0.56°C/s are presented in Figure 4. Similarly, strain data from both unirradiated and irradiated HT9 cladding tested at 5.6°C/s are presented in Figures 5 and 6, respectively. Strain data from HT9 cladding tested at 110°C/s are presented in Figure 7. As with the previous strength data figures, 316 SS cladding strain trends and data are also presented in Figures 4 through 7 for comparison purposes.

Discussion

An examination of the HT9 cladding FCTT data was made to determine the effect of fluence, irradiation temperature, geometry, and heat on transient properties. A comparison of the HT9 cladding transient results with 316 SS properties will follow.

HT9 Transient Properties Dependence on Irradiation and Test Parameters

Figure 8 presents trend lines for the HT9 cladding tested at the three heating rates. The lines are an "average" of the strength results at each heating rate and are generally a good fit for results from unirradiated cladding. These trend lines together with low fluence (1-3 x 10^{22} n/cm^2) and high fluence (11-16 x 10^{22} n/cm^2) cladding strength data are plotted for 5.6 and 110°C/s in Figures 9 and 10, respectively. At both heating rates, differences between the fluence extremes and the trend lines appear minimal, which suggests only a minor dependence of strength on fluence.

Data for failure strength obtained at the lower heating rate of 0.56°C/s exhibit no obvious difference between the irradiated and unirradiated conditions, although fluences are limited to below 3 x 10^{22} n/cm^2.

Irradiation temperature effects should show up most clearly at the highest heating rate, as the least thermal recovery would take place at that rate. Indeed, the greatest strength deviation from the HT9 cladding trend

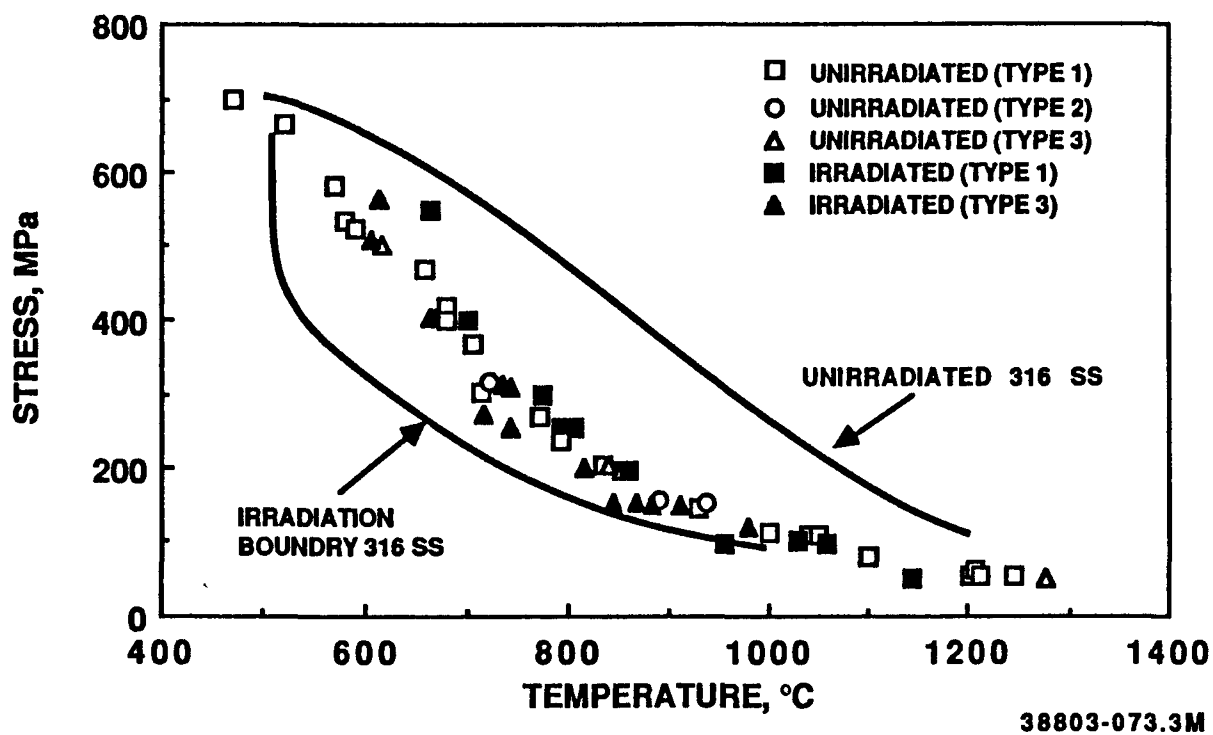

FIG. 3--HT9 110°C/s Transient Strength Results.

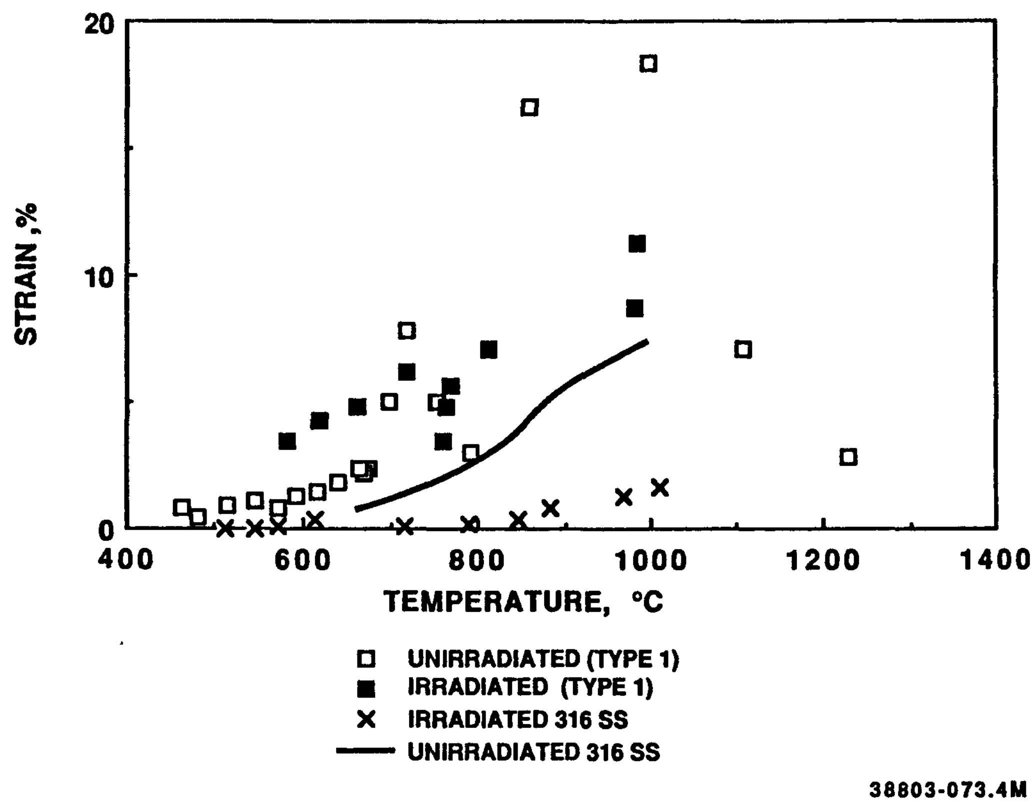

FIG. 4--HT9 0.56°C/s Transient Ductility Results.

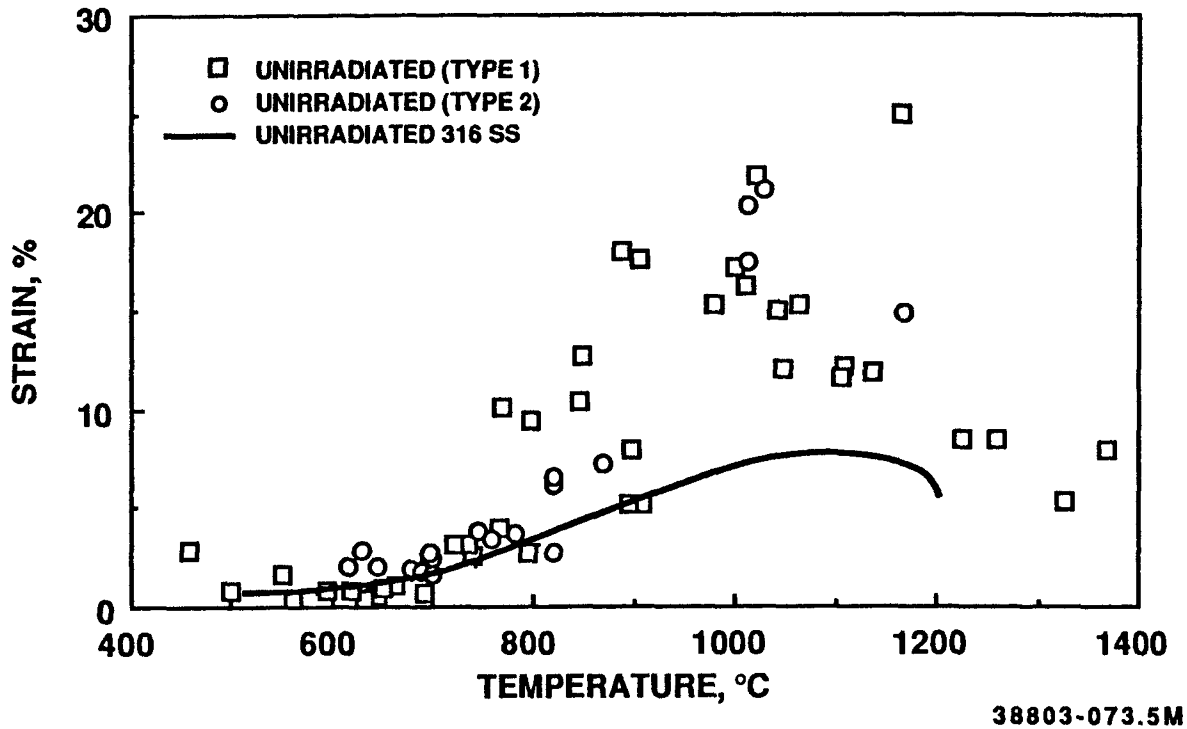

FIG. 5--HT9 5.6°C/s Transient Ductility Results for Unirradiated Cladding.

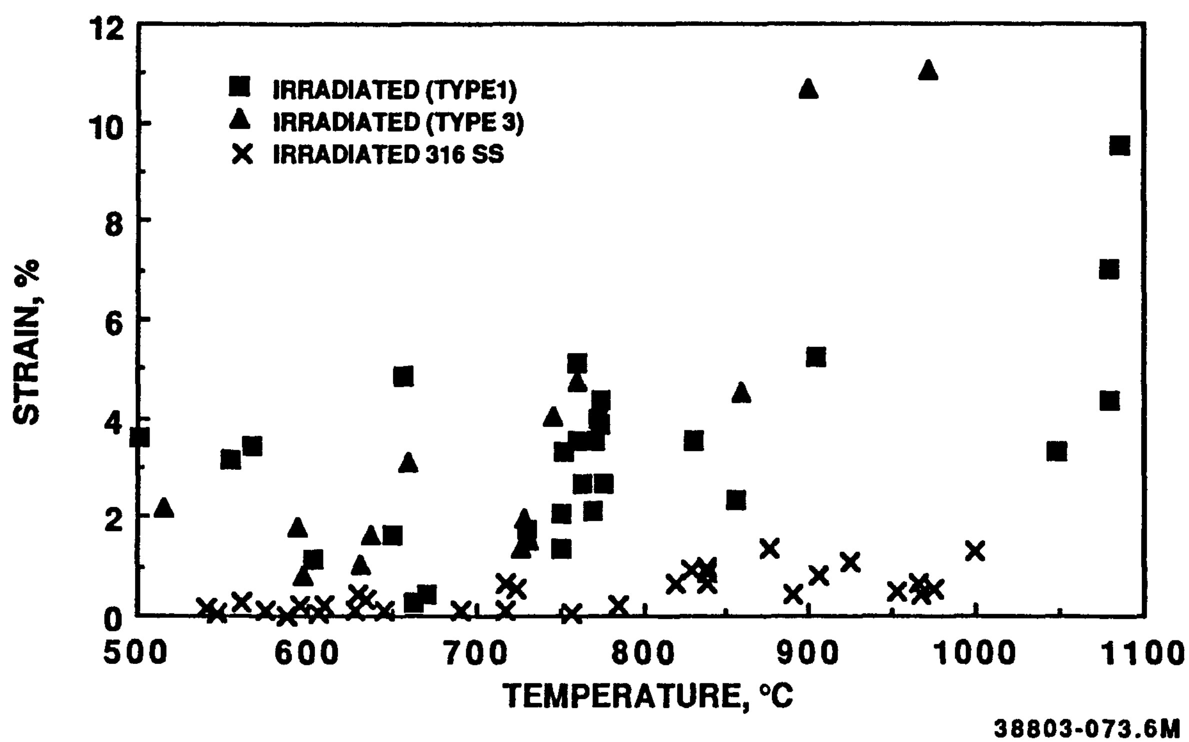

FIG. 6--HT9 5.6°C/s Transient Ductility Results for Irradiated Cladding.

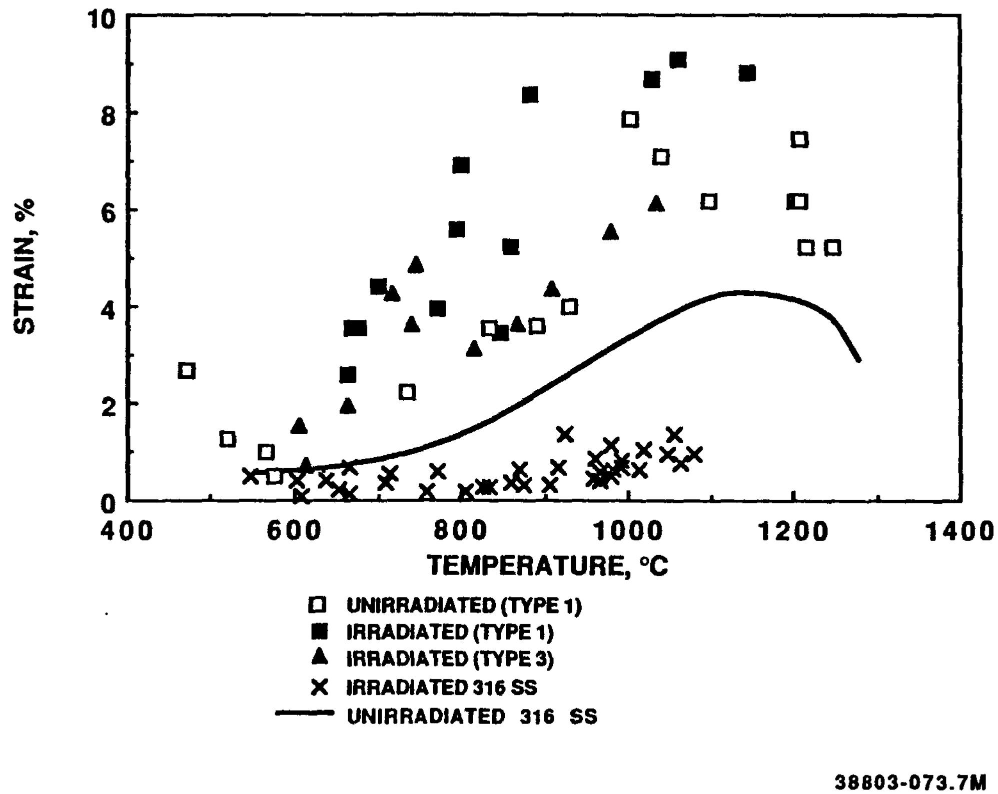

FIG. 7--HT9 110°C/s Transient Ductility Results.

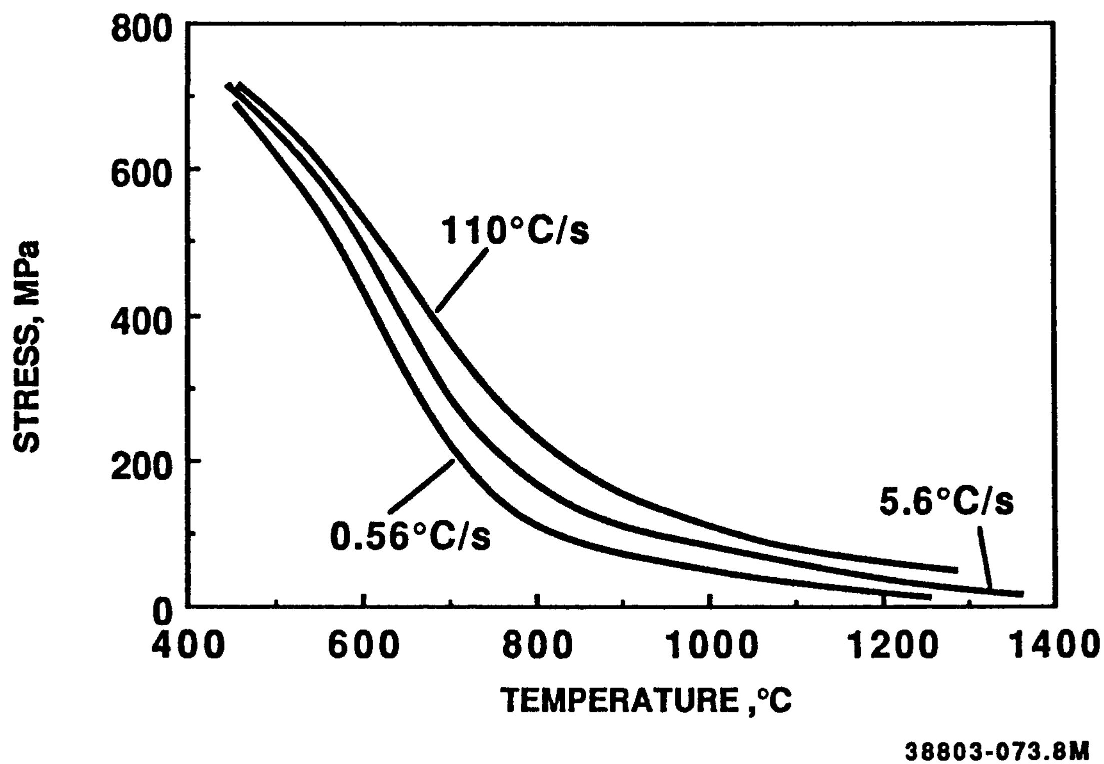

FIG. 8--HT9 Transient Strength Trend Curves.

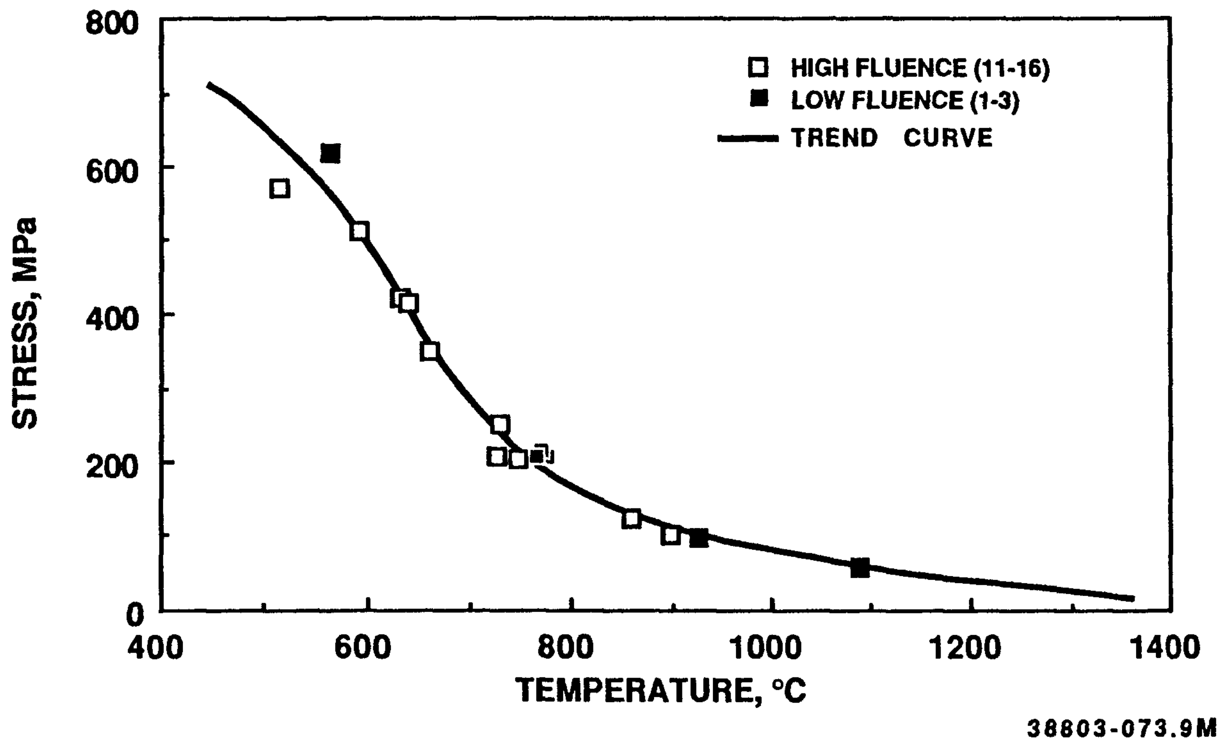

FIG. 9--HT9 Fluence Extreme Comparison at 5.6°C/s.

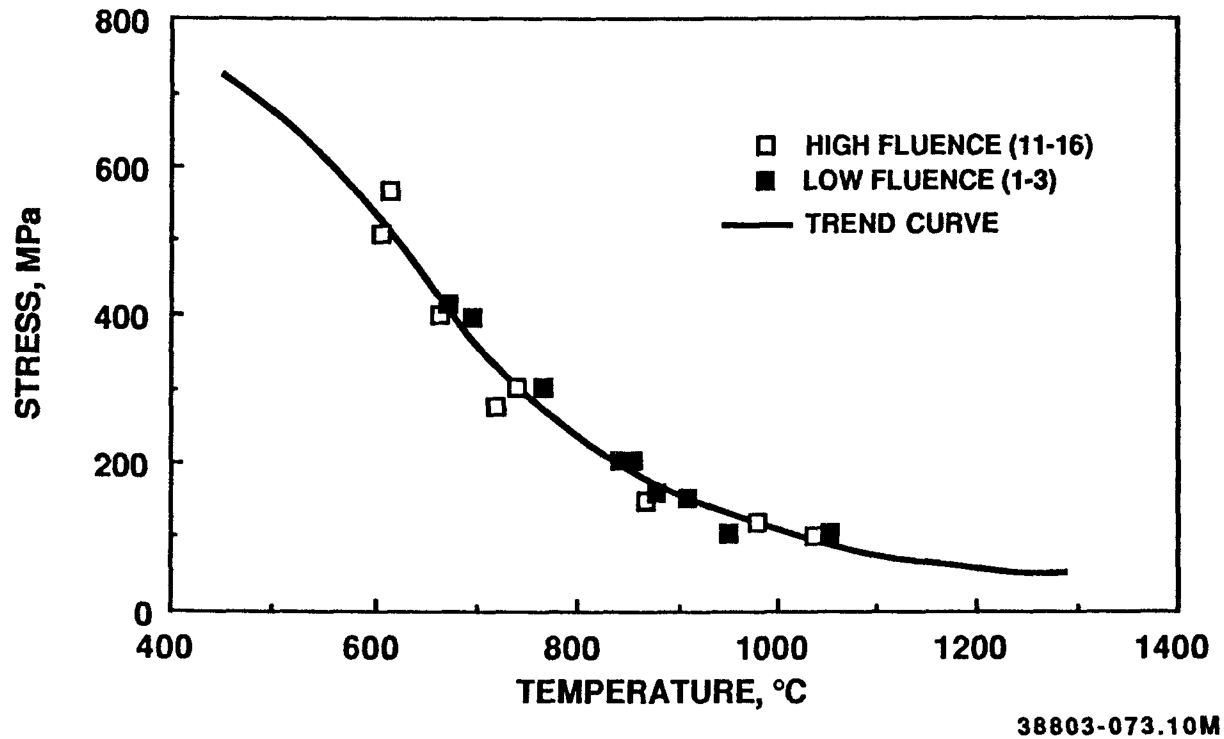

FIG. 10--HT9 Fluence Extremes Comparison at 110°C/s.

line was observed at 110°C/s. There appears to be a small trend toward irradiation softening for failure temperatures above 700°C, particularly for the large diameter (AC01) pins. There are also limited indications of irradiation hardening for specimens tested at stress levels above 550 MPa at both 5.6 and 110°C/s. Figure 11 presents 110°C/s strength data split by irradiation temperature at 550°C; three of the four largest deviations at this heating rate were for irradiation temperatures above 550°C. However, only a minor dependence of strength on irradiation temperature is observed.

Ductility data were presented in Figures 4 through 7, and exhibit a fair amount of scatter. The generally high HT9 failure strains exhibited in these plots makes transgranular fracture the most likely mode of failure. A very limited amount of metallography was performed, and in all cases failures were transgranular.

The HT9 ductility scatter observed is considered to be more a property of uncontrolled strain-rate during the test than of variations in the test material. Under the constant stress conditions of the test, strain-rate has been modeled as exponentially dependent on specimen temperature [12]. The high HT9 cladding ductility observed produced generally high strain-rates (to 200%/s based on extensometer data) prior to failure. Qualitatively, minor variations in the specimen "micro-flaws" that initiate specimen failure could cause a more major fluctuation in failure strain because of the high strain-rates involved. However, there is much less cladding strength scatter because the exponential-like flow dependence on temperature produces flow initiation and failure over a narrow temperature span.

A peak in ductility was generally observed at about 1000°C for all HT9 cladding conditions and heating rates. For unirradiated cladding the peak failure strains were about 20 to 25% for the lower two heating rates, but dropped to about 8% at 110°C/s. For irradiated HT9 cladding, the peak heights were about 8 to 10%, and essentially independent of heating rate. Thus, irradiation generally reduced HT9 cladding ductility, although absolute ductility levels remained high at all three heating rates. The irradiation induced reduction of ductility without a significant change in cladding strength may be due to basic changes in the "micro-flaws" that initiate cracks and specimen failure; however, determination of the precise nature of these changes is beyond the scope of this work.

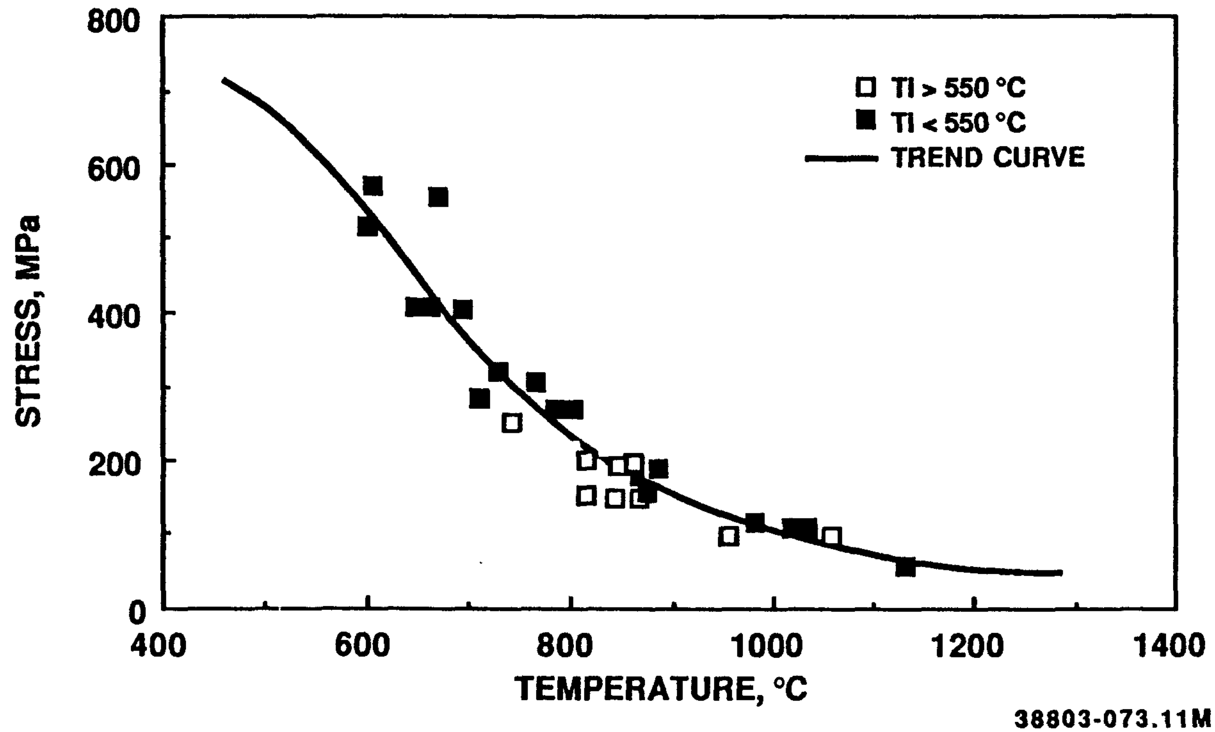

FIG. 11--HT9 High Irradiation Temperature Transient Strength Comparison at 110°C/s..

Some minor trends in the ductility data were noted, and will be discussed; however, the scatter in the data limits the conclusions that can be drawn from these trends. At 0.56°C/s, there appears to be a trend in which irradiation increases HT9 cladding ductility for failure temperatures below about 700°C, and decreases the ductility at higher failure temperatures.

At 5.6°C/s, irradiated HT9 cladding ductility was clearly less than unirradiated HT9 cladding for failure temperatures above 800°C. Irradiated HT9 cladding with failure temperatures below 700°C was slightly more ductile, in general, than in the unirradiated case.

At 110°C/s, irradiated HT9 cladding was generally more ductile than unirradiated HT9 cladding, but the maximum strain observed was reduced to less than 10% at this heating rate from about 20% at the lower heating rates.

Examination of Figures 1 through 7 does not reveal any consistent variation of either strength or ductility based on HT9 cladding heat or geometry.

HT9 Cladding Comparison with 316 SS

A comparison of HT9 cladding failure strength with that of 316 SS resulted in the same trends at all three heating rates (Fig. 1 to 3). The unirradiated 316 SS cladding was stronger than either unirradiated or irradiated HT9 cladding. However, this situation was reversed when irradiated 316 SS was compared with HT9 cladding, the unirradiated and irradiated HT9 cladding were both stronger than irradiated 316 SS.

The HT9 cladding was significantly more ductile than 316 SS at all three heating rates. At the rates of 0.56°C/s and 110°C/s, both unirradiated and irradiated HT9 cladding were more ductile than unirradiated 316 SS (Fig. 4 and 7), and ductility decreased significantly for irradiated 316 SS. At 5.6°C/s, there was some overlap of HT9 cladding and 316 SS ductility for unirradiated cladding below 850°C (Fig. 5); for irradiated cladding at this heating rate, HT9 cladding was clearly more ductile (Fig. 6).

The "irradiation boundary" given in the strength plots (Fig. 1 to 3) indicates worst-case results from 316 SS cladding affected by the fuel adjacency effect (FAE) at fluences of 5×10^{22} n/cm^2 or greater. Although these fluences were higher than for the low-rate HT9 cladding, a comparison remained of interest because a characteristic of the FAE is a significant reduction of strength and ductility results obtained for cladding over a wide range of fluence.

The FAE is further characterized by intergranular failure, low failure strain, and little or no macro corrosion. Failure strengths can vary widely for cladding irradiated and tested under essentially the same conditions.

Evidence indicates that the FAE is caused by liquid metal embrittlement of the 316 SS cladding by fission products, specifically cesium and tellurium. Transient tests performed by Duncan, et al.[13] demonstrated that cleaning a specimen with a nitric acid solution eliminated the FAE. Auger electron spectroscopy analysis of an uncleaned specimen indicated appreciable quantities of the fission products cesium and tellurium on the cladding inner surface, while these elements were not detected on an acid-cleaned specimen.

Fuel removal and cleaning for all the HT9 specimens tested was consistent with techniques used for 316 SS specimens, and would have left a thin layer of fuel and fission products coating the specimen inner wall.

A chemical screening process for reactor cladding was described by Adamson, et al.[14] with specific application for determining FAE-like behavior. The process included coating various cladding materials with a cesium-tellurium solution and testing for liquid metal embrittlement at elevated temperatures. It was found that face-centered cubic austenitic alloys (including 316 SS) were susceptible to embrittlement, whereas body-centered cubic ferritic alloys (including HT9 cladding) were not. This lack of embrittlement of HT9 cladding by the cesium-tellurium solution is consistent with the lack of an observed FAE in the HT9 cladding FCTT results.

Conclusions

Under simulated transient conditions at heating rates from 0.56°C/s to 110°C/s, irradiated HT9 fuel pin cladding significantly outperformed irradiated 20% Cold Worked Type 316 Stainless Steel (316 SS) cladding in both strength and ductility. The irradiated HT9 cladding exhibited no evidence of the fuel adjacency effect often observed in irradiated 316 SS fuel pin cladding.

There was not a significant correlation of HT9 cladding transient failure strength with either fluence (to nearly 16 x 10^{22} n/cm^2) or irradiation temperature (370 to 615°C). Indications of minor irradiation hardening at high stress and softening at failure temperatures above 700°C at 5.6 and 110°C/s were observed.

A peak in ductility was generally observed at about 1000°C for all HT9 cladding conditions and heating rates. For irradiated HT9 cladding, the peak heights were about 8 to 10%, and essentially independent of heating rate. Neutron irradiation generally reduced HT9 cladding ductility, although absolute ductility levels remained high at all three heating rates.

References

[1] Gelles, D. S., "Effects of Irradiation on Ferritic Alloys and Implications for Fusion Reactor Applications," Journal of Nucl. Matls., Vol. 149, 1987, pp. 192-199.

[2] Puigh, R. J., and Wire, G. L., "In-Reactor Creep Behavior of Selected Ferritic Alloys," AIME Proc. of Topical Conference on Ferritic Alloys for Use in Nuclear Energy Technology, 1984, p. 601.

[3] Makenas, B. J. and Lawrence, L. A., "Performance of Ferritic Alloy Cladding in LMR Mixed-Oxide Fuel Pins in EBR-II," ANS Transactions, November 1986, pp. 221-222.

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

[5] Hunter, C. W., Fish, R. L., and Holmes, J. J., Nuclear Technology, Vol. 27, November 1975, pp. 376-388.

[6] Hunter, C. W. and Johnson, G. D., "Mechanical Properties of Fast Reactor Fuel Cladding for Transient Analysis," ASTM STP 611, American Society for Testing and Materials, Philadelphia, PA, 1976, pp. 101-118.

[7] Hunter, C. W. and Johnson, G. D., "Fuel Adjacency Effects on Fast Reactor Cladding Mechanical Properties," International Conference on Fast Reactor Fuel Performance, ANS, Monterey, CA, March 1979, pp. 478-488.

[8] Duncan, D. R., Johnson, G. D., Hunter, C. W., and Hanson, J. E., "Measurement of Cladding Strain During Simulated Transient Tests," ANS Proc. of the International Meeting on Fast Reactor Safety Technology, Seattle, WA, Vol. 3, August 1979, pp. 483-492.

[9] Cannon, N. S. and Duncan, D. R.,"Effects of Irradiation Temperature, Fluence, and Heating Rate on Postirradiation Flow Properties of Cladding Under Simulated Temperature Transient Heating and Deformation Conditions," ASTM STP 683, American Society for Testing and Materials, Philadelphia, PA, 1979, pp. 557-566.

[10] Duncan, D. R. and Hunter, C. W., "Postirradiation Cladding Strength Under Biaxial Loading with an Increasing Temperature Ramp," ASTM STP 725, American Society for Testing and Materials, Philadelphia, PA, 1981, pp. 443-451.

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

[12] Wire, G. L., Cannon, N. S., and Johnson, G. D., "Prediction of Transient Mechanical Response of Type 316 SS Cladding Using an Equation of State Approach," Journal of Nucl. Matls., Vol. 62, 1979, pp. 317-328.

[13] Duncan, D. R., Panayotou, N. F., and Wood, E. L., "Chemical Degradation Mechanisms of Fast Reactor Fuel Cladding Mechanical Properties," ANS Trans. Vol. 38, 1981, p. 265.

[14] Adamson, M. G., Reineking, W. H., Kangilaski, M., and Vaidyanathan, S., "Screening Alternative LMFBR Cladding Alloys for Fission Product-Induced LME and FCCI," ANS International Conference on Reliable Fuels for Liquid Metal Reactors, Tucson, AZ, September 1986.