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AN EMPIRICAL STUDY OF
SNAP REACTOR FUEL
IRRADIATION BEHAVIOR
(Title Unclassified)

AEC Research and Development Report

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SNAP REACTOR,
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M-3679 (50th Ed.)

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IRRADIATION BEHAVIOR
(Title Unclassified)

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by R.G. Chalker, Mgr. Eng. Sec. Mgt 5/3/73

by GO TIC, date 5/24/73

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ABSTRACT

The growth of zirconium - uranium hydride fuel during irradiation at elevated temperature is reviewed historically. Correlations of the early experimental data with temperature and burnup are compared with the more recent data, including that obtained from the SNAP 8 Experimental Reactor (S8ER). A new empirical correlation of all the available zirconium - uranium hydride irradiation data is presented. Fuel growth is associated with the presence of zirconium-hydride phase phenomena superimposed on burnup effects, and the effects of the carbon additive, temperature, and other parameters are considered, but appear to be secondary in importance to hydrogen composition. A brief review of the zirconium-hydride phase diagram system is included as a section of the report.

The results of the study are applied to the SNAP 8 Developmental Reactor (S8DR). Estimates of the fuel/clad contact probability based on the S8ER gap distribution and the proposed fuel correlation indicate that, for a minimum diametral gap of 6.4 mils, less than one element in the S8DR should be expected to have fuel/clad contact. Contact is defined as complete circumferential contact at any axial point. The estimate is based on 600 kwt/1300° F coolant outlet operation for 12,000 hr.

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I. INTRODUCTION

The SNAP (Systems for Nuclear Auxiliary Power) family of compact nuclear reactors has been developed by Atomics International under contract with the U.S. Atomic Energy Commission. Reactors which have been operated include the SNAP 2 Experimental Reactor (SER), the SNAP 2 Developmental Reactor (S2DR), the SNAP 10A Reactor (S10FS-3), and the SNAP 8 Experimental Reactor (S8ER). Data for each of these reactors are shown in Table 1.

TABLE 1
SNAP REACTOR IRRADIATION EXPERIMENTS

Experiment	Coolant Outlet Temperature (°F)	Peak Burnup (met. at. %)	Power (kw)	Energy Released (kwh)
SER	1200	0.02	50	2.25×10^5
S2DR	1200	0.02	65	2.73×10^5
S10FS-3	1000	0.025	38	3.8×10^5
S8ER	1300	0.25	600	5.1×10^6

Zirconium - uranium alloy hydride was used as the fuel-moderator material for each of these reactors. The unique ability of this material as a moderator at high fuel temperatures (1500°F) reduces the magnitude of the reactivity transient during startup. The hydrogen which serves as the moderator is bound to the zirconium and the core average hydrogen density is relatively invariant with temperature. The operation of the SER,⁽¹⁾ S2DR,⁽²⁾ and S10FS-3⁽³⁾ was highly successful and clearly demonstrated the feasibility of this design approach. No fuel element damage was detected in any of these reactors; however, because of the successful operation, only limited examinations were made. The planned operation of the S8ER at 600 kw/1300°F coolant outlet required quantitative fuel growth data. The proposed operation was at a higher burnup and fuel temperature than for any of the cores which had been operated. The NAA-53 Irradiation Experiments were run to evaluate the effects of burnup and temperature on fuel growth. Correlations of these data with burnup and temperature were presented in References 4, 5, 6, and 7. The various equations in the order of reference are given here:

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- 1) $\% \Delta V/V = 1.2 \cdot 10^6 b^2 e^{-22600/T} + b$
- 2) $\% \Delta V/V = T b^{1.25} (t \ln t)^{0.25} e^{(3.7 - 22500/T)}$
- 3) $\% \Delta V/V = [196b + 50.2 (bT)^{1.5}] e^{-24040/T}$
- 4) $\% \Delta V/V = b^{1.5} (T/1000)^{1.5} e^{(9.8 - 17400/T)}$

where

T = fuel temperature ($^{\circ}$ R)

b = burnup (met. at. %)

t = time (hr)

V = fuel volume.

The data used to obtain these correlations are presented in Table 1. The fuel pieces were nominally 0.188-in. -diameter by 2.5-in. -long containing 10 wt % U, 1.65 wt % H, carbon additive, and the remainder Zr. The experiment was made at accelerated burnup rates to demonstrate the feasibility of clad zirconium-uranium hydride fuel elements operating at SNAP 8 temperature and burnup levels prior to the operation of the S8ER. A secondary objective was to gain additional information concerning the effects of radiation on the fuel at various temperatures and burnup levels. The overall program to define fuel growth during irradiation has been continued, and results have been obtained from the post-operational examination of the S8ER,⁽⁸⁾ the NAA-82-1 Irradiation Test,⁽⁹⁾ the NAA-115-1 Irradiation Test,⁽¹⁰⁾ the NAA-116 Irradiation Test,⁽¹¹⁾ and the NAA-117-1 Irradiation Test.

The SNAP 8 Experimental Reactor was shut down on April 15, 1965, after accumulating one year of operation⁽¹²⁾ in the power range of 400 to 600 kw with a NaK coolant outlet temperature of 1300 $^{\circ}$ F. On August 2, 1965, removal of fuel elements from the core was begun, and the initial screening examination, completed on September 17, 1965, showed that 167 of the elements had one or more cracks in the cladding. While the exact sequence of events which resulted in the cracking of the cladding on the S8ER fuel is still undetermined, it is certain that the fuel growth was a contributor. Detailed examination of 15 selected elements from the core clearly indicated that fuel growth had exceeded the predicted values. Subsequent data from the experiments listed earlier and from the S10FS-3 supported the findings from the S8ER.

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Data from the several irradiation experiments show effects due to burnup, temperature, additives, hydrogen content, and mechanical restraint.

A correlation of fuel growth for the parameters mentioned is needed in order to optimize reactor design and performance. Since the hydrogen moderator is effective only so long as it is retained in the fuel, growth of the fuel must not affect the hydrogen retention properties of the element. A ceramic oxide barrier is coated on the inner surface of the cladding to improve the hydrogen retention characteristics of the cladding at SNAP reactor operating temperatures (up to 1300°F coolant outlet). This barrier material can withstand little tensile strain and remain fully effective.

Design of the fuel element is basically fixed by the reactivity requirements for a particular application. Reactivity requirements during operation of a high temperature reactor (1300°F outlet) are dominated by the effect of hydrogen loss from the fuel. Sufficient radial and axial clearance must be allowed to guarantee that the hydrogen barrier will remain effective during the design lifetime. This problem cannot be solved by making the clearance between the cladding and fuel larger than that necessary to accommodate the expected fuel growth, because the thermal conductivity of hydrogen gas in the gap is sufficiently low to cause a significant temperature drop across the gap, directly proportional to the gap width. Thus, the hydrogen loss rate from the fuel, which is strongly temperature dependent, is affected by the gap size. The gap must accommodate fuel growth, but the reactivity loss considerations require that the gap size be minimized.

This report represents an extension of the fuel growth correlation for SNAP reactor fuel to account for the variables in addition to temperature and burnup. This involves developing a growth model consistent with the data obtained from the S8ER and the results are considered applicable for predicting the fuel growth for the SNAP 8 Developmental Reactor (S8DR).

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II. EMPIRICAL CORRELATION STUDIES

The results of the NAA-53 Irradiation Test were used to obtain fuel growth correlations to be used to predict the fuel growth for S8ER. The irradiation of NAA-53-1 was terminated on July 24, 1961, NAA-53-2 was removed from the MTR on November 27, 1961, and NAA-53-3 was removed from the MTR on May 6, 1963. Since that time the S8ER fuel has been examined, and the NAA-82-1, NAA-115-1, and the NAA-116 Irradiation Tests have been reported.^(8,9,10,11) Results from the NAA-117-1 and NAA-115-2 Irradiation Tests will be available in the near future. (Measurements of the cladding diameter changes for the NAA-117-1 samples have been obtained and the diametral fuel growth inferred.) The primary need for a new correlation which reflects the new data is to predict the fuel growth expected for S8DR.

The data considered most applicable were obtained from the post-irradiation examination of the S8ER. Even though the reactor operated at various powers and temperatures and was subjected to thermal cycling during tests and scrams, the results represent the only large body of data for uranium-zirconium-hydride fuel which is directly applicable to reactor environments expected for the S8DR. The data from the various irradiation tests are generally for sub-dimensional fuel without axial redistribution of hydrogen, a situation which normally occurs during reactor operation. There is good agreement, on the whole, between the S8ER results and those obtained from the irradiation tests. This fact tends to support the continuation of the irradiation test program to cover a wider range of parameters.

Parameters considered as variables in the correlation were burnup, temperature, hydrogen content, hydrogen loss, carbon content, and time.

The attempt to correlate the data was begun to ascertain what had happened to the fuel irradiated in the S8ER, to find a correlation which fit all the irradiation experiment data, and to provide a basis for predicting the capability of the S8DR.

Uncertainties in each of the parameters are relatively large and make it necessary to use a statistical sample. An individual datum point has little weight, but a cluster of points can be used. Diameters are measured to an

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accuracy of ± 0.0005 in. ($\sim 0.1\%$), densities are measured to ± 0.006 gm/cc ($\sim 0.1\%$), hydrogen content is measured to ± 0.05 H/Zr, and the uncertainty on the remaining parameters is dependent on the method of measurement. Calculated values for the burnup and temperature have been used in this analysis.

The diameter change observed was used as the basis for this analysis. Variations in the ratio of volumetric change to diameter change were large enough to establish the importance of the diameter changes as an engineering variable. The final correlation is based on diameter-change data obtained from samples with intact cladding and samples which were not affected by β -phase material or large axial mechanical restraint. A second effort is continuing on the analysis of volumetric change and the method of analysis is discussed below.

A ceramic material is coated on the cladding tube to retain the hydrogen moderator. Since it is desirable for this material to suffer little tensile strain, diameter changes of the fuel are of prime importance. As discussed above, however, the diametral gap must be minimized to reduce the operating temperature of the fuel.

The data are in general agreement and both volumetric-change and diametral-change data are presented in this report. However, the design requirements are most sensitive to changes in the diameter of the fuel and this parameter was used in preparing an engineering design correlation for diametral gap selection usage.

A. BURNUP

This study began with a collection of data (Tables 2 through 9) from irradiation tests. The data are presented graphically in Figures 1 to 5 which show the volumetric change plotted against the fuel burnup. No separation of the data has been made and both β -phase growth and axial restraint affected data are included. The samples which were affected by axial restraint or β -phase material are not used to design S8DR, since the design requirements consider the presence of either of these as a design lifetime constraint. The S8ER data with the β -phase samples deleted are shown in Figure 6.

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TABLE 2
NAA-53 IRRADIATION TEST DATA

Sample Number	Burnup (met. at. %)	Fuel Temperature (°F)	Irradiation Time (hr)	$\% \frac{\Delta V}{V}$	Initial Hydrogen Content (H/Zr)	Final Hydrogen Content (H/Zr)	Carbon Content (wt %)
53-1U	0.68	1300	2076	1.3	-	-	0.02
53-1L	0.52	1472	2076	3.6	1.77 ± 0.02	1.58 ± 0.03	0.02
53-2U	0.97	1444	3876	8.8	-	-	0.02
53-2L	1.03	1213	3876	4.5	1.77 ± 0.02	1.63 ± 0.19	0.02
53-3U	0.36	1515	1385	2.7	1.77 ± 0.03	1.69 ± 0.18	0.5
53-3L	0.30	1411	1385	-	1.80 ± 0.05	1.42 ± 0.12	0.5

TABLE 3
NAA-115-1 IRRADIATION TEST DATA

Sample Number	Burnup (met. at. %)	Fuel Temperature (°F)	$\% \frac{\Delta V}{V}$ (maximum)	Initial Hydrogen Content (H/Zr)	Final Hydrogen Content (H/Zr)	Carbon Content (wt %)
1	0.39	1525-1575	2.75	1.68	1.47	0.1
2	0.41	1500-1550	2.90	1.59	1.50	0.1
3	0.41	1425-1475	3.73	1.65	1.61	0.02
4	0.41	1460-1510	4.30	1.71	1.65	0.4
5	0.41	1560-1610	4.01	1.63	1.62	0.02
6	0.43	1440-1490	2.86	1.69	1.55	0.1
7	0.41	1590-1640	4.15	1.81	1.50	0.1
8	0.43	1475-1525	3.51	1.81	1.44	0.1
9	0.43	1590-1640	1.91	1.69	1.62	0.1
10	0.42	1550-1600	1.65	1.69	1.63	0.1
11	0.39	1375-1425	1.94	1.67	1.67	0.1
12	0.37	1275-1325	2.41	1.69	1.68	0.1

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TABLE 4
NAA-43 IRRADIATION TEST DATA⁽¹³⁾

Sample* Number	Burnup (Mass Spec) (met. at. %)	Fuel Tempera- ture (°F)	Irradi- ation Time (hr)	% $\frac{\Delta V}{V}$	Initial Hydrogen Content (wt %)	Final Hydrogen Content (wt %)
43-5U	2.05	768	9652	7.1	1.0	0.96
43-5L	1.89	-	9652	7.1	1.0	1.0
43-6	3.35	730	13370	7.3	1.0	0.78

*Fuel pieces were unmodified 90 Zr-10U

TABLE 5
NAA-75 IRRADIATION TEST DATA⁽¹⁴⁾

Sample* Number	Burnup (met. at. %)	Fuel Tempera- ture (°F)	Irradi- ation Time (hr)	% $\frac{\Delta V}{V}$	Initial Hydrogen Content (wt %)	Final Hydrogen Content (wt %)	Carbon Content (wt %)
75-1U	1.05	1075	5253	3.77	1.443	1.23	0.2
75-1L	2.15	1011	5253	7.11	1.455	1.48	0.2
75-2U	0.50	1059	2955	4.94	1.411	1.33	0.2
75-2L	1.26	914	2955	4.11	1.400	1.52	0.2

*Each fuel specimen was composed of Zr-15 U-6.5 Hf-0.2 C.

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TABLE 6
NAA-82-1 IRRADIATION TEST DATA

Sample Number	Axial Position (in.)	% $\frac{\Delta V}{V}$	Axial Position (in.)	Burnup (met. at. %)	Axial Position (in.)	Initial Hydrogen Content (H/Zr)	Final Hydrogen Content (H/Zr)	Fuel Temperature (°F)
1	3/8 to 1-3/8	0.98	1-3/4 to 2-1/8	0.07	0 to 3/8	1.84	1.89	1050
2	2-1/8 to 3-1/8	0.62	5-1/2 to 5-7/8	0.14	1-3/8 to 1-3/4	1.84	1.85	1250
3	3-5/8 to 5-1/2	1.02	8-5/8 to 9-1/4	0.06	3-1/8 to 3-5/8	1.84	1.77	1425
4	6-1/4 to 7-1/4	1.79	-	-	5-7/8 to 6-1/4	1.84	1.70	1490
5	7-5/8 to 8-5/8	1.30	-	-	7-1/4 to 7-5/8	1.84	1.70	1490
6	9-5/8 to 10-5/8	0.98	-	-	9-1/4 to 9-5/8	1.84	1.71	1425
7	11-1/8 to 11-7/8	0.46	-	-	10-5/8 to 11-1/8	1.84	1.81	1325
8	-	-	-	-	11-7/8 to 12-1/4	1.84	1.81	1150

Fuel specimen was composed of Zr-10U-0.19C-0.21 impurities
Irradiation time was 8760 hr

TABLE 7
NAA-116-1 IRRADIATION TEST DATA

Sample Number	Uranium Content (wt %)	Carbon Content (wt %)	Initial Hydrogen Content (H/Zr)	Final Hydrogen Content (H/Zr)	% $\frac{\Delta V}{V}$	-% $\frac{\Delta \rho}{\rho}$
14	10.04	0.12	1.27	1.24	1.30	1.26
15	10.11	0.09	1.86	1.66	1.31	1.22
16	10.00	0.02	1.80	1.76	1.60	1.15
17	9.89	0.08	1.68	1.67	1.59	1.23
18	9.89	0.08	1.68	1.66	1.08	1.15
19	19.00	0.13	1.75	1.76	1.56	1.47
20	9.98	0.40	1.42	1.39	2.23	1.53
21	9.95	0.43	1.76	-	1.85	1.42
22	19.00	0.13	1.75	1.72	2.56	2.82
23	9.89	0.08	1.68	-	-	-

Samples were 0.29 in. in diameter by 0.75 in. long and operated at ~500°F fuel centerline temperature

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TABLE 8
NAA-116-2 IRRADIATION TEST DATA

Sample Number	Uranium Content (wt %)	Carbon Content (wt %)	Initial Hydrogen Content (H/Zr)	Final Hydrogen Content (H/Zr)	% $\frac{\Delta V}{V}$	-% $\frac{\Delta \rho}{\rho}$
24	10.00	0.02	1.75	1.68	3.13	4.03
25	10.11	0.13	1.26	-	3.71	3.25
26	9.89	0.08	1.68	1.64	3.48	4.15
27	10.04	0.12	1.27	-	-	-
28	10.11	0.09	1.86	1.85	3.79	4.97
29	9.95	0.43	1.76	-	3.47	4.36
30	9.95	0.43	1.88	-	3.37	4.51
31	9.95	0.43	1.85	-	-	-
32	5.00	0.10	1.65	1.65	1.80	1.61
33	5.00	0.10	1.66	-	2.11	-

Operated at ~500°F fuel centerline temperature

TABLE 9
NAA-117-1 IRRADIATION TEST DATA⁽¹⁵⁾

Sample Number	Burnup (met. at. %)	Fuel Centerline Temperature (°F)	% $\frac{\Delta V}{V}$ (maximum)	Average Hydrogen Content (H/Zr)		Carbon Content (wt %)
				Initial	Final	
1	0.77	1496	2.85	1.69	1.62	0.10
2	0.77	1539	2.82	1.69	1.53	0.10
3	0.77	1499	4.56	1.70	1.57	0.05
5	0.77	1492	5.06	1.55	1.35	0.05
6	0.77	1534	4.20	1.60	1.45	0.09
7	0.77	1542	3.59	1.64	1.58	0.40
9	0.51	1350	1.95	1.70	1.62	0.05
10	0.51	1417	2.42	1.69	1.62	0.10
11	0.51	1389	2.35	1.70	1.67	0.40
12	0.37	1278	1.34	1.67	1.63	0.10
13	0.31	1511	1.06	1.60	1.61	0.40
14	0.31	1449	0.91	1.70	1.44	0.40
15	0.37	1327	4.22	1.83	1.66	0.11
16	0.37	1287	1.07	1.85	1.65	0.40
17	0.69	1340	2.17	1.81	1.62	0.05
18	0.69	1361	2.76	1.79	1.71	0.11

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Figure 1. Fuel Growth vs Burnup With NAA-53, NAA-115-1, and Empirical Correlations

Figure 2. Fuel Growth vs Burnup With NAA-53 and NAA-75 Data

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Figure 3. Fuel Growth vs Burnup With NAA-82-1 and NAA-116-1 and -2 Data

Figure 4. Fuel Growth vs Burnup With NAA-43 and NAA-75 Data

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Figure 5. S8ER Fuel Growth Data

Figure 6. Diametral Fuel Growth vs Burnup
With NAA-82-1, NAA-115-1, and
S8ER Data

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At this point the importance of some variable (or variables) other than burnup is apparent. This additional phenomenon gives a projected offset at zero burnup. Burnup, of course, is a major variable, theoretically amounting to a minimum of 1.2 to 2.3% $\Delta V/V$ per metal atom percent burnup, but cannot be considered as representing the whole effect.

Diameter-change data are shown, plotted against burnup in Figure 7 after having been corrected for hydrogen loss. These data were obtained only from samples with intact cladding. The data were also separated according to their hydrogen content during irradiation, the data being restricted to samples operating in the δ - and ϵ -phase domain.

B. CARBON AND TIME

The carbon additive in uranium-zirconium hydride fuel improves the grain structure of the extruded elements. The effect of the amount of carbon additive on diametral fuel growth is shown in Figure 8. Both the NAA-115-1 data and S8ER data have been included, and there is an indication that the amount of carbon additive has an effect on fuel growth of about 0.4% $\Delta D/D$. The effect is relatively small in the range 0.05 to 0.20 wt % carbon (approximately 0.2% $\Delta D/D$). The uncertainty on individual measurements is $\pm 0.13\%$ $\Delta D/D$ and the points shown in Figure 8 represent the midpoint of a cluster of points at each value of carbon additive. Thus the effect of carbon additive in the range below 0.2 wt % is considered a secondary variable. The diameter-change data are shown in Figure 9, after corrections for hydrogen loss and carbon content have been made.

The effect of time was established by separating the data into four arbitrary time units:

- 1) The end-of-life (EOL) hydrogen content overlaps the beginning-of-life (BOL) calculated hydrogen profile after redistribution.
- 2) The EOL hydrogen content is just less than the BOL prediction.
- 3) The EOL hydrogen content is within one uncertainty band (calculated for the BOL prediction) of the BOL prediction.
- 4) The EOL hydrogen content is greater than one uncertainty band below the BOL prediction.

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Figure 7. Diametral Fuel Growth, Hydrogen Corrected

Figure 8. Diametral Fuel Growth vs Carbon

Figure 9. Diametral Fuel Growth, Hydrogen and Carbon Corrected

Figure 10. Diametral Fuel Growth, Hydrogen, Carbon, and Time Corrected

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The correction term was obtained by plotting the data from Figure 9 against the arbitrary time units. Midpoint values were used to establish the magnitude of the effect, as in the case of the carbon additive. The variation was about 0.15% $\Delta D/D$; i. e., approximately equal to the uncertainty on the individual data points. Thus, time is also considered to be a secondary effect. The time-corrected data are shown in Figure 10.

C. TEMPERATURE

The effect of temperature on non- β -phase material radial growth is illustrated in Figure 11, and the temperature-corrected data are shown in Figure 12. The uncertainty in the individual data points is nearly equal to the spread of the data. Thus, temperature has been considered important but as secondary to phase. The activation value on these data is about 12,500°R.

Previous correlations, based on the NAA-53 irradiation test data, indicated that the activation temperature is about 22,600°R. This value is not supported by the observed radial fuel growth; however, the density-change data supports a multiplier nearer to that calculated using the NAA-53 data. Plots of density change vs axial position were prepared for each of the S8ER elements on which measurements were available. A few of the points were obviously in error, but the majority appeared consistent.

The concept of the $\delta/\delta + \epsilon/\epsilon$ family was introduced earlier for the diameter change data. Density-change measurements from elements or portions of elements which were assigned to this family were plotted against reciprocal temperature. The results are shown in Figure 13, and no attempt has been made to correct these data beyond the correction for loss of hydrogen. The activation temperature for this data would be about 25,000°R. If the data from axially unrestrained samples with intact cladding are taken alone, the value is about 14,000°R, as shown in Figure 14.

D. HYDROGEN - BETA PHASE

Even though β -phase material is not considered for the present correlation, its effects on growth are interesting. Several samples from the irradiation tests and reactor experiments were affected by β -phase material growth. Data are presented in Table 10.

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Figure 11. Diametral Fuel Growth vs Temperature - $\delta/\delta+\epsilon$ Group

Figure 12. Diametral Fuel Growth, Hydrogen, Carbon, Time, and Temperature Corrected

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Figure 13. Volumetric Fuel Growth

Figure 14. Volumetric Fuel Growth - Unrestrained Axially

TABLE 10
ZIRCONIUM HYDRIDE FUEL GROWTH - BETA-PHASE MATERIAL

Sample	Temperature (°F)	Burnup (met. at. %)	$-\% \Delta \rho / \rho$ (as measured)
NAA-53-2-U	1450	0.97	8.8
NAA-53-3-L	1410	0.3	4.5
NAA-117-1-3	1540	0.73	4.6
NAA-117-1-5	1390	0.73	5.1
NAA-117-1-6	1400	0.73	4.2
S8ER-309	1400	0.2	5.8
S8ER-398	1380	0.2	3.7

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The effect of β -phase on material growth is also shown by annealing test results. Samples of irradiated fuel were annealed at temperatures such that β -phase material was present, and the results are shown in Table 11.

The results clearly show that β -phase material growth is dependent on both temperature and burnup. Creep properties of β -phase material relative to δ - and $\delta+\epsilon$ -phase material are shown in Figure 15. The creep rate for a given stress and temperature is several orders of magnitude higher for β -phase material than for the δ - or $\delta+\epsilon$ -phase material. For a given bubble agglomeration, the resistance to fission gas bubble growth may be ranked: $\delta+\epsilon$ -phase material greatest, δ - or ϵ -phase material next, and β -phase material lowest.⁽¹⁶⁾

The transformation from δ - to β -phase material begins with the nucleation of β -phase material in the δ -phase material. As the amount of β -phase material increases it tends to concentrate in the region of highest temperature. Boundary diffusion results from the increase in amount of β -phase material and the boundary sweeps the fission gas with it.⁽¹⁷⁾ Once fission gas bubbles have formed, they are free to diffuse along the phase boundary.

TABLE 11
POST-IRRADIATION ANNEALING RESULTS — BETA-PHASE
MATERIAL

Sample	Burnup (met. at. %)	β (%)	Time (hr)	Temperature (°F)	$-\% \Delta \rho / \rho$
116-2/251F2	0.83	85	75	1600	24.0
116-2/253F1	0.83	85	75	1600	16.2
116-2/273F4-2	0.83	85	75	1600	15.9
116-2/273F4-1	0.83	70	75	1500	6.8
116-2/14-2	0.18	50	90	1300	-0.5
S8ER/519C	0.2	5	4	1400	0.1
S8ER/476A1	0.1	100	75	1700	12.1

Thus, the effects of β -phase material growth on the fuel size can be extreme, and its presence in an operating core is considered as a limit. Since its presence has been eliminated by design, it was not considered further in the analysis presented here.

E. HYDROGEN - DELTA AND EPSILON ZONES

The initial hydrogen content for SNAP reactor experiments has been ≤ 1.8 . Thus, after thermal-gradient-induced hydrogen redistribution, the δ - and ϵ -phase materials are encountered at the beginning-of-life. After the S8ER fuel was examined, it was apparent that the fuel growth had exceeded that predicted. The data were plotted against end-of-life H/Zr as shown in Figure 16. There appeared to be an increase in fuel growth between 1.7 H/Zr and 1.6 H/Zr. Further investigation led to the conclusion that the presence of certain H/Zr material during irradiation effected fuel growth. Data obtained from the S8ER are shown in Figures 17 and 18. The movement of hydrogen through each of the elements is indicated by the arrows for H/Zr values of 1.75, 1.65, and 1.4 (β). Fuel growth appears to be associated with the presence of these H/Zr values and/or their passage. Similar effects were observed for Element 529 from S10FS-3, and for the NAA-82-1 irradiation test sample. The data are shown in Figures 19 and 20 for these samples. Of these data, those from the NAA-82-1 are the best indicators, as the magnitude of the observed diametral distortions exceeds the 0.06% $\Delta D/D$ measurement uncertainty. At this point, it appears that the data are separable into groups that were swept by some range of H/Zr. The conclusion is that there are at most three distinct H/Zr values which strongly affect growth, and these are arbitrarily referred to as β , $\delta/\delta+\epsilon$, and $\delta+\epsilon/\epsilon$ boundaries. Since β -phase is excluded by design, this leaves only two postulated boundary-affected "families". They will be approximated for this analysis by H/Zr values of 1.65 and 1.75 for the postulated $\delta/\delta+\epsilon$ and $\delta+\epsilon/\epsilon$ boundary-affected families, respectively. These two families were found to be similar, and were condensed into one $\delta/\delta+\epsilon/\epsilon$ family.

It is apparent that the data presented in Figures 7 through 12 and Table 12 are representative of the $\delta/\delta+\epsilon/\epsilon$ family. The data were obtained from samples with intact cladding, the presence of particular hydrogen content has been required, and they have been adjusted for the effects of the carbon additive, the

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Figure 15. Creep Strength Data

Figure 16. Diametral Growth vs End-of-Life Hydrogen Content

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Figure 17. Diametral Fuel Growth for Low and Middle H/Zr S8ER Elements

Figure 18. Lattice Dimensions vs H/Zr for Zirconium Hydride

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CONFIDENTIAL RESTRICTED DATA (G-1)

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Figure 19. SNAP 10FS-3 Fuel Swelling

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Figure 20. NAA-82-1 Irradiation Test — Initial H/Zr 1.84

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TABLE 12
DIAMETRAL FUEL GROWTH DATA

Description*	Family	Sample Mean $\left(\% \frac{\Delta D}{D} - b\right)$	Sample Standard Deviation $\left(\% \frac{\Delta D}{D} - b\right)$
Hydrogen Corrected	$\delta/\delta+\epsilon/\epsilon$	0.525	0.185
Hydrogen and Carbon Corrected	$\delta/\delta+\epsilon/\epsilon$	0.366	0.172
Hydrogen, Carbon, and Time Corrected	$\delta/\delta+\epsilon/\epsilon$	0.447	0.163
Hydrogen, Carbon, Time, and Temperature Corrected	$\delta/\delta+\epsilon/\epsilon$	0.566	0.159
Hydrogen, Carbon, Time, and Temperature Corrected	$\delta/\delta+\epsilon$	0.552	0.116
Hydrogen, Carbon, and Time Corrected	$\delta+\epsilon/\epsilon$	0.386	0.096

*Corrected to 0.15 wt % carbon, time Group 1, and 1400°F average fuel temperature

time the boundary was present, and the temperature using the correlation above.

Two additional restrictions were applied to the data shown in Figure 12; all points which were greatly restrained axially (axial cladding growth >10 mils) are eliminated and the $\delta/\delta+\epsilon$ and $\delta+\epsilon/\epsilon$ groups are separated. The results of this separation are shown in Figures 21 and 22. The sample mean (24 samples) for the $\delta/\delta+\epsilon$ group is 0.552, with a sample sigma of 0.116 in terms of $(\% \Delta D/D - b)$. The sample mean (34 samples) for the $\delta+\epsilon/\epsilon$ group is 0.386, with a sample standard deviation of 0.096 in terms of $(\% \Delta D/D - b)$. For the first set, the sample standard deviation is $0.096 < S_x < 0.148$ at the 85% confidence level, and for the second set the sample standard deviation is $0.083 < S_x < 0.115$ at the 80% confidence level.

The groups of data are then separable, to a degree, and larger than previously predicted fuel growth is associated with each one. An offset is apparent when the data are extrapolated to zero burnup. The sample standard deviation on the individual sets of data is less than the uncertainty on the measurements.

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3-17-67 CONFIDENTIAL RESTRICTED DATA (GP-1)

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Figure 21. Diametral Fuel Growth - Unrestrained $\delta/\delta+\epsilon$ Group

3-17-67 CONFIDENTIAL RESTRICTED DATA (GP-1)

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Figure 22. Diametral Fuel Growth - $\delta+\epsilon/\epsilon$ Group

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Data which support the observed effects of hydrogen content and presence of postulated boundaries will be presented below. The data presented thus far merely indicate that fuel growth may be correlated with certain specific H/Zr contents and that the effects of carbon additive, time, and temperature are of the same order of magnitude as the uncertainty on the individual measurements. This study is continuing with an analysis of the density-change data, using an approach similar to that described here.

Samples which were not traversed by the "boundaries" discussed above have less fuel growth. The sample mean for this set was approximately 0.5 (% $\Delta D/D$ - b) less than that observed for the combined $\delta/\delta + \epsilon/\epsilon$ family. The sample standard deviations were approximately equal. This difference was calculated using as-measured diameter changes rather than the hydrogen corrected values shown in Figure 7. Since the effect of hydrogen loss is only approximately 0.07% $\Delta D/D$ per 0.1 reduction in H/Zr for H/Zr > 1.59, the difference between the boundary-affected and non-boundary-affected sets would be essentially unchanged if a hydrogen loss correction were made.

F. OTHER OPERATIONAL VARIABLES

Other variables which apparently affected fuel growth include fuel rod breaks, rinds, and mechanical restraint. A very local effect of fuel rod breaks was observed for the density-change measurements, and because of its limited propagation about a break, this apparently did not grossly affect the diameter-change measurements. Some samples were prepared by making a single cut and taking the piece between the cut and the break. Cracks in the fuel material near the break apparently caused these samples to displace a greater volume. Investigation indicated that the wetting (i. e., displacement) could be changed by soaking the sample for longer times. The magnitude of this effect was calculated as ~0.5% $\Delta V/V$ by comparing the results from several different elements.

Dehydrided or chemical reaction rinds were observed on the S8ER elements that were cracked. The rind apparently had no appreciable effect on the density measurements, but did have an effect on the diameter measurements. An increase of about 0.5% $\Delta D/D$ (0.53 in. diameter fuel) was typically associated with the presence of a dehydrided rind. This effect was evaluated by comparing

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the results from otherwise similar elements. The ratio, $-\% \Delta \rho / \rho$ to $\% \Delta D / D$, against EOL H/Zr for the $\delta/\delta+\epsilon/\epsilon$ family set data may be used to show the effect of rind formation. Those which had a rind showed a ratio of ~ 2 and those which had none showed a ratio of ~ 3 .

Axial restraint effects apparently enhance local fuel diametral growth, especially in the $\delta/\delta+\epsilon$ family. Preferential radial growth is also indicated for the $\delta+\epsilon/\epsilon$ family by the ratio of density-to-diameter change, but the effect of axial restraint is less.

All of these effects are of peripheral importance to this study. Axial restraint may be eliminated by design. Rod breaks affect only a very local density and not the overall physical dimensions of the fuel. Rind effects are associated with cladding cracking which apparently allows a NaK-fuel interaction, and is extraneous to a hydride fuel growth study.

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III. PHENOMENOLOGICAL CORRELATIONS WITH PHASE BOUNDARIES

Much of the development presented has been postulated to relate to the effect of a σ - to ϵ -phase transformation in fuel material at least at some time during operation, but not necessarily during high temperature operation. The effect of these boundaries is indicated by the NAA-82-1 irradiation test results, by the S10FS-3 results, and by the S8ER results. Similar effects are observed for the NAA-115-1 and the NAA-117-1 test results, but these experiments had radial rather than axial temperature gradients. The composite of the data includes samples from both sets. All data - S8ER based as well as that from irradiation experiments - are from fuel that operated a significant (10 to 20%) time at low temperature ($<1000^{\circ}\text{F}$).

The major fuel growth contributors appear to be burnup and the postulated phase boundaries; i. e., hydrogen content. Secondary effects which have been considered include carbon additive, time a boundary was present, and temperature. The two major contributors have been treated as separable.

A. SUMMARY OF PHASE STUDIES

The results of a study of the ZrH_x phase diagram are given in Reference 18, and a proposed phase diagram is shown in Figure 23. The location of the cross-hatched lines shown and the temperatures at which they exist have not yet been fully investigated. Their presence, as shown in the figure, is indicated by the fuel growth data at H/Zr values of ~ 1.65 and ~ 1.75 .

Zirconium is an oxygen getter and the presence of oxides in the SNAP fuel hydrogen barrier represents a source. The presence of oxygen has been postulated as the principal cause for the variation in published values of H/Zr corresponding to the first appearance of ϵ -phase material. This same effect contributes to the published widths of the two-phase, $\delta+\epsilon$ zone.⁽¹⁹⁾ Thus, the two-phase region indicated by the fuel growth data may be a result of oxygen contamination, and may be variable extent depending on the level of contamination.

There are some data which support the postulated boundary locations in the operating domain of the irradiation experiments, S8ER and S8DR (800 to 1500°F). Hardness measurements (Figure 24) and room temperature x-ray diffraction and metallography were used by Beck⁽¹⁹⁾ to define the H/Zr at the onset of the ϵ -phase

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Figure 23. Zirconium Hydrogen Phase Diagram (Maki)

Figure 24. Zirconium Hydride Hardness Data

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material (Figure 25). Room temperature x-ray diffraction was used by Korst⁽²⁰⁾ to define the two-phase $\delta + \epsilon$ zone, and the H/Zr range was 1.64 to 1.73. A discontinuity in the slope of the axis ratio c_o/a_o at H/Zr of 1.75 was found, and the data are shown in Figure 26.

Some data are available, such as pressure-temperature-concentration measurements, which support the presence of the postulated boundaries at elevated temperatures. Theoretical expressions for dissociation pressure have been derived⁽²¹⁾ having the form

$$\ln P + A + N \ln \frac{H/Zr}{2 - H/Zr} + G(T) ,$$

where A and N are constants and G(T) is a temperature-dependent term. The data of Libowitz⁽²²⁾ and Raymond⁽²³⁾ are shown in Figures 27 and 28, respectively. Both plots appear to show a change in slope for H/Zr values of ~ 1.65 and ~ 1.75 . A change in slope represents a change in G(T) which would be expected at a phase boundary.

If the transformation from delta to epsilon is a pseudo-martensitic diffusionless transformation⁽²⁴⁾ it should have zero width. The finite width indicated by the fuel growth data is probably due to contamination of the samples by oxygen.

It should be noted that all of the fuel growth data came from samples that were irradiated at both high and low temperatures. For example, the S8ER operated at 450 kw, 900°F coolant outlet temperature for a total of about 43 days in the spring of 1964 prior to the 600 kw, 1300°F run. All irradiation experiments operate for an appreciable fraction ($\sim 20\%$) of their irradiation time on pure helium during purges, shutdown, startups, etc. The result is a fuel temperature in the 800 to 1000°F domain. It is, therefore, uncertain whether the observed growth for H/Zr ~ 1.65 and ~ 1.75 material represents the effect of two existing boundaries at operating temperatures in the 1200 to 1500°F domain, or the effect of a single boundary at H/Zr ~ 1.65 at low temperatures (800 to 1000°F) and at H/Zr ~ 1.75 at 1200 to 1300°F. Either hypothesis is suitable for the purposes of the empirical correlation presented here.

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Figure 25. Zirconium Hydrogen Phase Diagram (Beck)

Figure 26. Zirconium Hydride Lattice Constants

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$$\left[\text{Theoretical form of pressure is } P \propto \left(\frac{H/Zr}{2-H/Zr} \right)^4 T e^{-C/T} \right]$$

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Figure 27. Zirconium Hydride Partial Pressure Data (Libowitz)

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66-7695-168-14A

Figure 28. Zirconium Hydride Partial Pressure Data (Raymond)

B. PHYSICAL PROPERTIES OF δ/ϵ MATERIAL

The effects of uranium and carbon on the SNAP reactor fuel have been the subject of considerable study. The uranium is necessarily present as a precipitate, but the carbon is added primarily to give a finer grain structure. Carbon affects the dissociation pressure indirectly by forming ZrC, which increased the effective H/Zr ratio. This effect provides an incentive to produce high-yield material by some other method;⁽²⁵⁾ however, no superior grain modifier has yet been developed.

1. Fuel Microstructure

Optical microscopy has revealed that uranium precipitates from the solid solution (Zr-10U-H_x) at an H/Zr between 0.5 and 1.0.^(25, 26) Platelets of ZrC also appear in the modified fuel and examples of S8ER fuel are shown in Reference 28. Stoichiometric ZrC does not hydride; however, it apparently serves as a nucleation point. The ZrC is generally found within the hydride grains and is predominantly absent from the grain boundaries. Increasing the amount of ZrC added results in larger platelets and finer grains, with the optimum value corresponding to 0.15 wt % carbon.⁽²⁵⁾

2. Compressive Strength

Both creep and short-term tests have been made to determine the compressive behavior of carbon modified fuels. At room temperature, the addition of 0.15 wt % carbon more than doubled the compressive strength of SNAP reactor fuel.⁽²⁷⁾ This is shown graphically in Figure 29. Another result that was noted was that hydrogen was lost during compressive testing. One postulate was that the strain imposed on the samples forced the hydrogen out of solution. Data were obtained for unmodified (0.05 wt % C) fuel and for 0.15 wt % carbon-modified fuel. The modified fuel had significantly better creep properties, as shown in Table 13. Additional information is shown in Figure 30.

"Compressive inelastic deformation of Zr-10U-H_x proceeds by massive microcracking rather than a flow process."⁽²⁸⁾ Since the microcracks tend to be arrested by grain boundaries, the fine-grained, high-carbon material requires more mechanical work to cause ultimate failure. Many more microcracks form in the high-carbon samples and the plasticity of the material results from the

Figure 29. Room Temperature Ultimate Compressive Strength vs Carbon Content

TABLE 13
CREEP DATA FOR Zr-10U ALLOY HYDRIDES⁽²⁷⁾

Temperature (°F)	Stress (psi)	Carbon Content (wt %)	Hydrogen Content (H/Zr)	Creep Rate (%/hr)
1300	10,000	0.05	1.60	1.5×10^{-3}
1300	7,000	0.05	1.72	1.0×10^{-4}
1300	10,000	0.05	1.72	1.0×10^{-3}
1400	10,000	0.05	1.60	5.7×10^{-2}
1400	10,000	0.05	1.72	2.1×10^{-2}
1300	10,000	0.15	1.80	$2.8 \times 10^{-4*}$
1400	10,000	0.15	1.60	2.7×10^{-4}

*This sample showed periodic creep behavior and the value shown is the minimum.

formation of these microcracks. Each one propagates until it is arrested at some grain boundary and the Young's modulus is increased by the microcracks.

3. Tensile Creep

Tensile creep tests have been conducted using SNAP reactor fuel.⁽¹⁶⁾ Fuel material with a hydrogen content of H/Zr = 1.72 exhibited better creep resistance

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Figure 30. Room Temperature Compressive Stress-Strain Diagrams

than the $H/Zr = 1.60$ or 1.80 material. This was the basis of the statement made earlier that $\delta + \epsilon$ -phase material is stronger than either δ - or ϵ -phase material.

Tensile properties of SNAP reactor fuel are dependent on temperature, hydrogen content, and carbon content. The ultimate tensile stress increases rapidly with temperature for unmodified fuel, up to $1400^\circ F$,⁽²⁹⁾ whereas $0.15 \text{ wt } \%$ carbon-modified fuel shows almost no change. At $1.74 H/Zr$, the ultimate tensile strength varies rapidly between 1200 and $1400^\circ F$. This anomalous behavior supports a postulated phase change at $\sim 1.75 H/Zr$ ($0.15 \text{ wt } \%$ carbon). Young's modulus increases when this material is heated from 1350 to $1400^\circ F$, indicating the presence of δ -phase material. The effect of carbon content on the ultimate tensile strength is shown in Figure 31. Again, the $0.15 \text{ wt } \%$ carbon modified fuel has the best qualities at $1.74 H/Zr$, which corresponds to the initial value for S8DR fuel.

4. Fission Gas Sweeping

Low-level irradiations of SNAP reactor fuel have been made to determine the effect of phase-boundary sweeping on fission gas release.⁽¹⁷⁾ The test results

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Figure 31. Variation of Ultimate Tensile Strength of Some Zr-10U Alloy Hydrides With Carbon Content

consistently showed that crossing the δ/ϵ -phase boundary did not result in fission gas release.

Fission products have ranges on the order of 10μ in homogeneous SNAP reactor fuel, and the majority are therefore distributed randomly. They must then migrate, or diffuse, before they can significantly affect fuel growth. The main parameters which determine the behavior of fission gas bubbles are the temperature gradient, the bubble radius, the surface diffusion coefficient, and the dissociation pressure.⁽³⁰⁾ However, these variables are not important if the fission gas does not have time to agglomerate appreciably. Small bubbles may be held by grain boundaries and still smaller ones by dislocations. Stresses applied to the material will cause the dislocations and associated bubbles to move; hence, to assist in agglomeration. The strength of the δ -, $\delta+\epsilon$ -phase material resists bubble growth, however.

If the δ/ϵ -phase boundary were diffusional, fission gas would surely be released by crossing it repeatedly. The absence of gas release supports the postulate that the δ/ϵ transformation is diffusionless.

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5. Hydrogen Diffusion

The diffusion of hydrogen in δ -phase material has been measured by nuclear magnetic resonance (NMR) techniques and the activation energy obtained was 12,690 cal/mol.⁽³¹⁾ This value is similar to the temperature dependence for the fuel growth that was shown in Figures 11 and 14. It is possible that hydrogen diffusion limits damage due to slip shear and twinning at the grain boundaries.⁽¹⁷⁾ There is a volume change associated with crossing the δ/ϵ -phase boundary.⁽²⁰⁾

6. Microcracking

It has been observed⁽³²⁾ for other alloys that the first crystals of a martensitic phase form in the plastic parent material and expand freely. If the second layer of crystals contacts the first, high localized elastic tensile stress is developed. Minute cracks have been detected in the martensitic structure at high magnification.

Microcracking has also been observed during post-irradiation examination of many SNAP reactor fuel samples. The effect⁽²⁸⁾ of microcracks on ϵ -phase material during compression testing is shown in Figure 32. The microcracks

HF CHEMICAL ETCH
+10% OXALIC ACID
ELECTROLYTIC ETCH

4694-1-2

Figure 32. Compression Tested Sample of
Zr-10U Alloy Hydride

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have served to pin the motion of the twin boundaries, and the resultant drag on the grain boundaries has caused an apparent bowing during deformation. The plasticity of the material, indicated by the large number of microcracks which have formed, implies an improvement of material properties with carbon content. Since grain boundaries serve as end points for the microcracks, an increase in the number of grain boundaries represents an increase in the number of microcracks. The finer-grained, high-carbon material can be expected to accommodate more stress.⁽²⁸⁾ Thus, the presence of microcracks in the fuel tends to indicate that a stress was encountered during irradiation. The finer-grained material tends to show larger amounts of macrodeformation at failure.

C. SUPPORTING DATA

Several examples of the effects of carbon content on the physical properties of SNAP reactor fuel were mentioned above. The presence of carbon in the fuel material serves a dual purpose: it increases the yield of elements during hydriding and it improves the strength properties. If zirconium carbide is added to the fuel during melting, platelets of ZrC form and serve as nucleation centers for hydriding.⁽²⁶⁾ The result is finer grains and thinner epsilon banding, with an optimum at 0.15 wt % carbon. The postulated growth mechanism of microcracking suggests that fine grains would tend to increase the overall effect by introducing more boundaries to serve to pin microcracks in the stressed δ/ϵ boundary region.

Lattice mismatch damage is not a new phenomenon. Volume change of certain anisotropic polycrystalline metals in the percentile range have been induced by thermal cycling. Volume changes observed during thermal cycling for some anisotropic metals are shown in Figure 33.

No suggestion of thermal cycling instability in ZrH_x is to be inferred here. Indeed, repeated isothermal cycling of unstressed ZrH_x in the δ/ϵ domain has been undertaken and no thermal cycle instability detected. The comparison presented above is directed to pointing out a non-burnup oriented growth mechanism that might occur in stressed ZrH_x fuel. Similar lattice oriented mechanisms could account for a portion of the burnup-independent increment of growth empirically correlated here with $H/Zr = 1.65$ and 1.75 as could a variety of

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Figure 33. Volume Change in Polycrystalline Metals
From Thermal Cycling Induced Intergranular
Stresses

other mechanisms. The comparison with volume changes in anisotropic metals under thermal cycling conditions is made solely to illustrate a non-irradiation growth mechanism, hence to indicate that the observation of an apparently non-burnup growth increment in ZrH_x is not a singular phenomenon.

The unit cell dimensions are shown in Figure 34 for ZrH_x . The magnitude of the lattice mismatch between ϵ -phase and δ -phase material corresponds to $\sim 0.4\% \Delta V/V$ at room temperature. The effect of phase boundary cycling would be expected to be additive, since the initial mismatch would be expected to be relieved by microcracking. (32)

The activation energy for hydrogen diffusion in ZrH_x has been experimentally determined using NMR techniques. The activation energy is given as

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Figure 34. Zirconium Hydride Unit
Cell Volume

$12,690 \pm 3440 \text{ cal/mol}$,⁽³¹⁾ and the corresponding values in the units used here are $11,500 \pm 3100^\circ\text{R}$. The range includes both values shown in Figures 11 and 14.

Studies are presently being conducted to measure the effects of thermal cycling a stressed SNAP fuel sample. The results will show whether the simple axial loading increment on diameter that has been postulated is correct, or whether other phenomena must be considered.

The empirical fuel growth correlation of the data does not depend on the interpretation of the causes. The results have been observed and the cause of the fuel growth is the quantity which requires further definition.

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IV. CONCLUSIONS

The results of this study are based primarily on the observed diametral fuel growth in S8ER and the developmental irradiation experiments listed earlier. Whereas study and correlation of volumetric growth behavior would be fundamentally more acceptable, the application of volumetric results to the S8DR fuel element design problem would necessitate the correlation of the observed anisotropy in the direction of fuel growth. In effect, this would increase the uncertainty in the predicted diametral fuel growth which is a key design parameter.

For the available data, a statistical treatment of the effects of burnup, hydrogen content, carbon content, and temperature on fuel diameter changes has resulted in a reasonable correlation. For the S8DR fuel composition and operation conditions (i. e. , 0.15 wt % carbon, 1400°F maximum radial average fuel temperature, and approximately 12,000 hr of irradiation in the H/Zr range of ~ 1.65 to ~ 1.75), the following correlation provides the best fit of the currently available data. The value obtained using this correlation represents the mean fuel growth at the conditions listed above:

$$\% \frac{\Delta D}{D} = 0.56 + b \quad ,$$

where b is the fuel burnup in metal atom percent. The data scatter about this mean has a one-sigma value of 0.137 (Figure 12). This compares with a diameter measurement uncertainty of 0.13% $\Delta D/D$ (± 0.0005 in.) for the S8ER fuel.

As is obvious in the above expression, the burnup-independent term dominates the fuel growth. This term is believed to be quantitatively related to both time and thermal history. The data are currently considered insufficient to permit extrapolation to conditions significantly different from the conditions representative of the original data, in particular irradiation times differing significantly from approximately 10,000 hr.

The mechanisms causing the burnup-independent portion of the growth are not totally understood. The postulated extension of the $\delta/\delta+\epsilon$ and $\delta+\epsilon/\epsilon$ phase boundaries into the characteristic fuel operating temperature region is only one

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of several possible explanations. Currently, however, it is considered to be the most likely.

The temperature dependence of fuel growth found in this study is smaller than previously reported.⁽⁴⁾ The present radial growth data are best correlated with an activation energy of 12,500°R compared with a value of about 22,600°R originally derived from the NAA-53 irradiation test data. A larger value, however, closer to the 18,000°R, is obtained from current studies of fuel-volume-change data. With respect to activation energy, Chang⁽²⁴⁾ reports that "the large internal friction peak found in $ZrH_{1.92}$ (but absent in $ZrH_{1.6}$) suggests that it is associated with stress-induced twin interface motion. The relaxation phenomenon appears to be controlled by a diffusion process, presumably that of hydrogen, with an activation energy of about 20,000 calories per mole (18,000°R)." This twin interface movement is also a postulate in agreement with the temperature dependence shown in Figures 13 and 14.

The findings of the present study are primarily of interest as they apply to the SNAP 8 Developmental Reactor (S8DR). The design lifetime of 12,000 hr is reasonably close to the time for which correlation is applicable. For a minimum gap of 6.4 mils in S8DR, less than one element should have fuel clad contact. This estimate is based on operation at 600 kw, 1300°F coolant outlet for 12,000 hr.

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