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REVIEW OF UZr-HYDRIDE DRIVER FUEL ELEMENTS FOR THERMIONIC REACTORS

by

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ABSTRACT

The feasibility of using uranium-zirconium-hydride fuel in driver fuel elements for the thermionic fuel experiment (TFE) is appraised. The results of the extensive irradiation tests and operational experience in the SNAP reactors and in TRIGA reactors are summarized. It is concluded that the UZr hydride fuels are suitable for use as driver fuels in the TFE.

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I. SUMMARY AND CONCLUSIONS

The SNAP reactor experiments and capsule irradiation tests of UZr-hydride fuels have included the range of fuel compositions, temperatures and burnups called for in driver type thermionic reactors (UZrH-60TFE and UZrH-19TFE). The results of these experiments and irradiation tests clearly demonstrate the feasibility of using the UZr-hydride driver fuel in the thermionic reactors.

1. The parameter of maximum fuel temperature of 640°C (1184°F) for a thermionic reactor driver fuel is well below the threshold temperature of approximately 760°C (1400°F) above which this fuel shows significant swelling at much higher burnups than is required for the driver fuel applications.
2. The fuel burnup required for the thermionic driver fuels, 1.45%U (0.088 metal atom % burnup) per 10,000 hours is well below the levels at which significant irradiation damage or swelling is observed in UZr-hydride fuels at elevated temperatures.
3. The dimensions specified for the thermionic driver fuels can be readily met, since UZr-hydride fuels of much larger dimensions have been fabricated (e.g. 1.5-in. diam. for the TRIGA fuel elements).
4. The cladding materials are available and proven for the UZr-hydride fuel for this application. Although the Incoloy-800 specified is a candidate cladding material, there are other alloys that are also under consideration, namely, Hastelloy-X and Hastelloy-N. Perhaps Inconel-625 should also be considered, because of its superior high temperature strength, compatibility with the barrier coating, and irradiation stability.

5. This report furnishes a selective survey of the pertinent information that is available from the SNAP-reactor experiment programs. The salient features of the UZr-hydride fuel element development and operational experience are presented in a summarized form.

II. INTRODUCTION

The UZr-hydride driver fuel for the thermionic reactor may be compared with the UZr-hydride fuels that have been developed, since 1956, for the SNAP-reactors and the TRIGA research reactors. The driver fuel (UZrH-60 and UZrH-19) parameters for the Thermionic Fuel Experiment (TFE) are listed in Table I.

The SNAP-reactor experiments, and the concomitant fuel irradiation capsule tests, cover a wide range of parameters which include the parameters in the thermionic reactor design data for UZr-hydride fuels. The estimated maximum fuel temperatures and the relatively low burnups required for the driver hydride fuels are well below the levels attained in the SNAP reactor fuels development programs. Even in the TRIGA reactors, while operating at fairly low fuel temperatures (about 400°C maximum in steady state), burnups of approximately 0.52 metal atom % (75% burnup of the U-235) have been successfully attained in about four years' exposure in the TRIGA-Mk III at Gulf General Atomic.

The main difference between the UZr-hydride for the thermionic driver fuel and the SNAP reactor fuels and the TRIGA fuels is the higher uranium content in the thermionic (15% versus 10% in SNAP and 8.5% in TRIGA). However, the SNAP reactor fuel capsule irradiation tests have included samples containing 15% and 19% uranium. Also, the higher uranium weight content in the driver fuel is equivalent to a volume increase in U-content of only 2%. This does not have a significant influence on the properties of this fuel.

TABLE I

UZr-HYDRIDE DRIVER FUEL PARAMETERS

<u>Parameter</u>	<u>Units</u>	Space Station	10KWe
		<u>UZrH-60 TFE⁽¹⁾</u>	<u>UZrH-19 TFE</u>
Clad OD	in.	0.46	0.46
Clad OD/ID	---	1.05	1.05
Clad Material	---	Incoloy 800	Incoloy 800
Fuel Composition	---	UZrH _{1.7} + 15 w/o U	UZrH _{1.7} + 15 w/o U
Max. Clad Surface Temp.	°C	610	605
Burnup/10 ⁴ Hours	a/o U	1.45 a/o U = 0.088 metal atom %	0.85 a/o U = 0.052 metal atom %
Linear Power, Ave.	KW/ft.	0.85	0.5
Max. Surface Heat Flux	W/cm ²	12.7	7.5
Est. Max. Fuel Temp.	°C	640°C (1184°F)	630°C (1166°F)
Peak Fluence to Clad/ 10 ⁴ Hours	nvt	8.9 x 10 ²⁰ >0.1 Mev	5.2 x 10 ²⁰ >0.1 Mev

The purpose of this report is to summarize the available information on the irradiation stability of the UZr-hydride fuels. The data are selected on the basis of their relevance to the thermionic driver hydride fuel. Much of this information has been documented in numerous reports from the SNAP-reactor program. The pertinent figures and tables from these reports have been compiled as an appendix to this report.

III. SNAP-REACTOR FUEL CHARACTERISTICS

The UZr-hydride fuel for the SNAP-reactors is an integral fuel-moderator system. The zirconium hydride matrix contains 10 w/o fully enriched uranium as a fine metallic dispersion. The H/Zr ratio is nominally 1.8 ($N_H = 6$). Carbon is added as a grain refiner in alloying (up to 0.2 w/o). The fuel rods were clad in Hast-N; however, the alloys Incoloy-800 and Hast-X are now candidate cladding materials for the next generation SPF reactor fuel elements. The cladding has a continuous layer of ceramic on its inner surface to lower the hydrogen permeation through the cladding to tolerable levels. The SCB-1 ceramic coating is applied as an aqueous slip, air dried and fired for five minutes at 2125°F.

The dimension of the gap between the fuel and the cladding is governed by two conflicting requirements, namely, (1) accommodation of fuel growth, and (2) heat transfer from the fuel to the coolant. The size of the gap must be minimized to reduce the fuel temperature for reactivity loss considerations and control of irradiation damage.

The hydrogen dissociation pressure for fuel with H/Zr ratios of 1.7 to 1.9 is given by:⁽²⁾

$$\log P = 10.44 (H/Zr) - 10.47 - 8538/T$$

where the dissociation pressure, P , is in atmospheres and the temperature, T , is in degrees Kelvin.

Excessive loss or migration of hydrogen leads to phase changes that are undesirable because of the resultant volume changes and increased swelling under irradiation.

The phenomena arising from the axial and radial temperature gradients which exist across the fuel elements in an operating SNAP-reactor may be described as follows:^(3, 4) "The axial profile results in operation of the higher-temperature zone of the element, near the reactor outlet, in the delta-phase. The lower-temperature end of the element, by reason of the temperature gradient, operates in the epsilon-phase region. The radial gradient superimposes a segregation of delta-phase along the centerline, and epsilon-phase to the surface in the boundary region⁽⁴⁾. . . . The fuel rod axial temperature distribution can be calculated for a given element if the axial hydrogen distribution of the fuel rod and a normalizing temperature (e.g. coolant inlet temperature) are known. The TEMPRO - code was used to make these calculations⁽³⁾."

IV. FUEL ELEMENT EXPERIENCE IN THE EXPERIMENTAL AND DEVELOPMENTAL SNAP-REACTORS

The early experimental SNAP-reactors operated successfully with NaK coolant outlet temperatures of 1000 to 1200°F and burnups of 0.02 to 0.025 metal atom %. These included the SER reactor (50 Kwt), S2DR (65 Kwt), and SLOFS-3 (38 Kwt). No fuel damage was detected in any of these reactors.^(5, 6, 7)

The S8ER experimental reactor operated in the range of 400 to 600 Kwt, with NaK coolant outlet temperature of 1300°F.⁽⁸⁾ The reactor was shut down in April, 1965, after accumulating one year of operation and fuel burnup of up to 0.25 metal atom % (5.1×10^6 Kwh of energy was released). Fuel growth resulting from high temperatures caused cladding failures in the majority of the fuel elements.

The SNAP 8 Development Reactor (S8DR) was a 600 Kwt, 1300°F NaK cooled UZr-hydride fueled, compact reactor which operated for about 7,000 hours before being shut down in December, 1969. ⁽³⁾ This reactor was designed, fabricated, and operated by Atomics International at its Santa Susana Nuclear Development Field Laboratory, as a ground test prototype of a flight configured reactor intended for use as the primary power supply in space applications. ⁽³⁾

The S8DR test was terminated because of indications of premature fuel element cladding ruptures. The 211 fuel elements in the core consisted of 16.825-inch long by 0.5929-inch O.D. fuel rods, with a composition of 10.5% U-ZrH_{1.75} with a nominal N_H of 6.05 (6.05 x 10²² atoms of H₂/CC of fuel). The cladding was 11-mil Hastelloy-N coated on the ID surface with a 2-mil layer of SCB ceramic barrier. The nominal gap between the fuel rod and the cladding was 7 mils (diametral).

The conclusions drawn from the results of the post-test examination of S8DR fuel elements were as follows: ^(9, 10)

1. "The fuel element cladding ruptures were the result of excessive cladding strain caused by fuel swelling due to temperatures in excess of the design values.
2. The fuel growth exceeded design limits due to overtemperature conditions in the core resulting from hydraulic maldistributions attributed to fuel element bowing and clustering patterns, greater than anticipated coolant bypass flow near the core periphery, and the detrimental effects of surge tank gas (which was entrained in the NaK coolant) on the fuel element-to-coolant heat transfer.
3. The Hastelloy-N cladding material strain-to-rupture characteristics observed in S8DR were in good agreement with design values.

4. Excluding elements which sustained major damage (e.g., cracked elements), it appears that some measure of barrier damage was incurred relatively early in life, particularly in the tube section region of the fuel elements; the damage seems to have occurred primarily as the result of stresses caused by element-to-element loadings resulting from bowing and clustering of the elements."
5. Fluctuations in outlet temperature and the core power, during initial power operations, were due to a mechanical instability of the fuel elements arising from an effective hydriding expansivity opposite to and greater than the thermal expansivity of the fuel rod.

The conclusions regarding the appearances of the S8DR fuel elements have been as follows: ⁽³⁾

- "On those elements characterized by little or no cladding strains ($< 0.3\%$ diametral strain) the fuel rods were generally intact, exhibited a shiny/matte appearance and the hydrogen barrier appeared to be in good condition."
- "Cracked elements indicated typical NaK fuel/ceramic reaction zones."
- "In those elements which sustained high hydrogen loss (and entered locally beta-phase at operating temperatures) the ceramic had been spalled from the cladding ID and had adhered to the fuel rod OD. The fuel rods exhibited numerous longitudinal cracks in the immediate vicinity of the high hydrogen loss zone in the fuel rod."
- "Eight elements exhibited longitudinal cracks in the fuel rods in the inlet region of the elements."
- "The original S8DR core design was based on a hydraulic test which did not mockup either the thermally bowed core condition or the rounded top on S8DR fuel elements. Changes from design flow

conditions in S8DR are attributed to:

- Clustering of fuel rods,
- An increased flow bypassing the active core region,
- The presence and concentration of surge tank gas bubbles within the core,
- A changed radial flow profile."

The rather drastic deviations from the design flow conditions of the coolant caused significant increases in the fuel temperatures. The fuel growth characteristics and damage in S8DR generally matched the regions of lower coolant flow. Also, the surge tank gas entrainment and concentration resulted in reduced coolant mass flow rates and increased fuel element surface resistance to heat flow. ⁽³⁾ The maximum design fuel temperature was 1425°F, whereas the maximum hydrogen-deduced temperature was 1524°F.

The maximum corrected volume increase in the S8DR fuel was 6.825%, with the fuel temperature at 1431°F (estimated) and a burnup of 0.075 metal atom %. ^(3, 11) The minimum corrected volume increase in the fuel was 0.479%, with the fuel temperature at 1201°F (estimated) and a burnup of 0.067 metal atom %. The maximum burnup was 0.172 metal atom % at 1412°F (estimated) with a $\Delta V/V$ of 3.173%. The minimum burnup was 0.042 metal atom % at 1212°F (estimated) with a $\Delta V/V$ of 0.545%. The maximum fuel temperature was 1524°F (estimated) with 0.123 metal atom % burnup and a $\Delta V/V$ of 5.364%. Over times of greater than 2000 hours, the fuel in the delta-phase (1285°F, estimated) showed a diametral growth $\Delta D/D$ of about 0.4%, the fuel in the epsilon-phase (1240°F, estimated) showed a $\Delta D/D$ of 0.9%, and the fuel in the delta plus beta phase (1475°F, estimated) showed a $\Delta D/D$ of 2.1 to 2.3%. Delta phase fuel at 1150°F had a $(\frac{\Delta D}{D} - B)$ of 0.4% (where B is the metal atom % burnup).

In the S8DR fuel elements the fuel-to-barrier adherence occurred in certain intact elements, as was observed in the S8ER elements. This factor, coupled with the differential motion between the fuel rod and cladding during reactor scram thermal transients, led to the step increases in reactivity loss rate observed during portions of the S8DR operation.⁽³⁾ However, the undamaged elements had lost only the amounts of hydrogen that would be expected from little or no barrier damage.

V. IRRADIATION TESTS OF UZr-HYDRIDE FUELS

The UZr-hydride fuels were irradiated in a number of capsule tests designed to establish the irradiation behavior of these fuels over wide ranges of temperatures, burnups, and fuel compositions. The results of these tests have led to an understanding of the mechanisms of swelling and to formulations of empirical correlations between swelling and the important variables of temperature, burnup, and fuel composition.

1. Mechanisms of Swelling

The swelling of the UZr-hydride fuels is governed by three basic mechanisms:

- (1) The accommodation of solid fission products resulting from fission of U-235. This leads to an estimated growth of approximately 1.2 to 2.3% $\Delta V/V$ per metal atom % burnup. It is relatively temperature insensitive.
- (2) The agglomeration of fission gases at elevated temperatures (above 1300°F). This takes place by diffusion of the xenon and krypton to form gas bubbles.
- (3) A "saturable cavity nucleation" phenomenon, which results from the nucleation and growth of irradiation-formed vacancies into voids over a certain range of temperatures where the voids

are stable. The saturation of growth by this mechanism was termed "offset" swelling. It was deduced from the rapid decrease in fuel-to-cladding ΔT experienced during the early part of the irradiation. ⁽⁴⁾ The saturation was reached in approximately 1500 hours.

The "offset", or equilibrium, bulk average fuel temperature is the temperature which the fuel reaches after offset swelling has been completed. This temperature has been used in correlating the UZr-hydride fuel swelling data from different irradiations. ⁽⁴⁾

The most excessive swelling occurs in the beta-phase at elevated temperatures by the fission gas agglomeration because the low creep strength of the beta-phase cannot accommodate the fission gas pressures in the gas bubbles. "Sweeping" of fission gases may occur by phase boundary motion if the beta phase forms in the irradiated fuel. For example, the beta-phase fuel specimens ($H/Zr = 1.2$ and 1.4) were post-irradiation annealed after low temperature ($700^{\circ}F$) irradiation. Annealing for 211 hours at $1300^{\circ}F$ produced small amounts of shrinkage, whereas annealing for 75 hours at $1600^{\circ}F$ produced 15 to 25% swelling. Annealing delta-phase fuel ($H/Zr = 1.6$ to 1.9) under the same conditions produced small amounts of shrinkage or swelling - less than 1.5% in all cases. ⁽¹²⁾ The shrinkage of the fuel upon post-irradiation annealing is ascribed to recovery of the matrix from damage caused at temperatures lower than those employed during annealing. Anomalous shrinkage may also be attributed to hydrogen loss.

The samples exhibiting large decreases in density showed cracks and voids which suggest fission gas agglomeration. ⁽¹²⁾ The void clusters in high-hydride samples could be correlated with epsilon-phase banding, and led to the conclusion that some damage mechanism (other than fission gas agglomeration) takes place based upon an apparent delta-epsilon phase boundary damage phenomenon. ⁽²⁾ The epsilon-phase irradiation data indicate the presence of stress-related fuel growth phenomena since the growth is highly radial oriented. ⁽¹³⁾ An important consideration in integrating the irradiation

results is the presence of thermal gradients in the fuel samples in the tests. It is evident from the data that the fuel in the reactor tests, and in the NAA-121 capsule test (which closely simulated reactor operating conditions and thermal gradients), showed a different temperature-swelling relationship compared to the other capsule fuel experiments where the fuel temperatures were more uniform. (13)

The sequence of events arising from conditions which lead to phase changes into the beta phase has been described as follows: (13)

"If a single area of the fuel were starting into beta, it would start to shrink, and therefore get hotter and lose more hydrogen, and shrink more, etc. The delta-beta transformation is accompanied by a large volume reduction; and if this process continued until an entire diametral section were in beta-phase, the ΔD would be on the order of -25 mils. This would cause an effective radial barrier at this position; furthermore, the large amount of hydrogen evolved that would necessarily be absorbed in adjacent sections could cause sufficient diametral increase to provide the short circuit. It can be imagined that the very hot beta fuel, as it accumulated fission gas, would start to swell; and, as it swelled and cooled, it would absorb back its hydrogen and continue swelling, causing the very large observed 30-mil cladding ΔD ."

2. Swelling Correlations

A number of attempts have been made to correlate the measured swelling of SNAP-reactor fuels with burnup, temperature, and hydrogen content. Large uncertainties in each of these parameters have made it necessary to use a statistical sample, (5) based upon clusters of points rather than upon individual datum points.

The observed dimensional changes indicate variations in the ratio of volumetric change to diameter change, "large enough to establish the importance of diameter as an engineering variable." (5) The fuel growth

is isotropic only when the ratio $\Delta V/V$ to $\Delta D/D$ is 3. The fuel growth is preferred in the radial direction when the ratio is between 2 and 3 and in the axial direction when the ratio is greater than 3. Axial shrinkage gives rise to a ratio of less than 2. The mean value of the ratio for the delta-phase is 2.8 ± 0.4 , and for the epsilon-phase is about 1.6 ± 0.5 .

In the swelling correlations used for the SNAP-reactor fuels, the bulk average temperature is considered to be identical to the arithmetic average of the peak centerline temperature and the surface temperature.⁽¹³⁾ The time-variation of this temperature was calculated from the beginning-of-life (BOL) temperature, based on the cladding thermocouple readings, the measured end-of-life (EOL) fuel $\Delta D/D$, and the following fuel swelling model: ⁽¹³⁾

$$\Delta V/V = 3B + \exp (-K/T)$$

where B is the burnup in metal atom %, K is a constant ($\sim 30,000$), and T is the bulk average fuel temperature (in degrees R).

As stated earlier, the greatest success has been achieved by using the "offset", or equilibrium, bulk average fuel temperature which the fuel reaches after offset swelling has been completed.⁽⁴⁾ The fuel swelling observed in the S8ER and S8DR reactor experiments were generally lower than those predicted from the capsule tests, except for the NAA-121 capsule test in which the temperature profiles were similar and the data correlate well with the S8DR data.⁽¹³⁾

A model of swelling based on burnup and temperature led to the relationship for volumetric growth of: ⁽⁴⁾

$$\% \frac{\Delta V}{V} = \alpha B + \beta \exp (-\lambda/T)$$

where $\% \frac{\Delta V}{V} = 2 \frac{\Delta D}{D} + \frac{\Delta L}{L}$; B = burnup in metal atom %; T = bulk average fuel temperature; and α , β , and λ are constants.

The fuel swelling data are usually plotted as log (corrected volume growth) versus $1/T$. The corrected volume growth for "offset" swelling has been given variously by:

$\frac{\Delta V}{V} - 2.76 B$, $\frac{\Delta V}{V} - 2.8 B$, and most recently $\frac{\Delta V}{V} - 3 B$ (the Bonzer-Swenson correlation).^(3, 13)

The S8DR data tend to show a greater temperature sensitivity than that shown by the Bonzer-Swenson correlation,⁽³⁾ but "the bulk of the data do fall within the Bonzer-Swenson scatter band and indicates a higher temperature sensitivity only if considered as a separate data set."⁽³⁾ The S8DR data show a correlation of $(\frac{\Delta V}{V} - 3 B)$ of 0.8% at 1250°F, 0.4% at 1200°F, and 0.2% at 1150°F.

3. Summary of Fuel Growth Data from Irradiation Tests

The most pertinent data on the growth of UZr-hydride fuels are presented in the tables and figures in the Appendix to this report. These data were reproduced from a number of SNAP-reactor project reports. Some of the salient features of these data are as follows:

- (1) The S8DR and capsule data indicate a swelling correlation volume growth $[(\Delta V/V) - 2.8 B]$ of 1.0 to 2.5% at 1250°F and 0.6 to 1.7% at 1200°F.⁽¹⁾ A corrected volume growth based on $[(\Delta V/V) - 3 B]$ gives 0.8% for 1250°F, 0.4% for 1200°F, and 0.2% for 1150°F.⁽³⁾
- (2) The NAA-121-1 capsule irradiation test most closely simulated the S8DR reactor conditions and showed comparable swelling rates.⁽¹³⁾ In this test four S8DR prototype fuel elements were used, with Hast-N cladding, SCB-1 hydrogen barrier, and 10% U-ZrH_{1.73} fuel.⁽¹³⁾ The irradiation was carried out in the Hanford-KE reactor. The clad temperature was 1100 to 1300°F, the fuel temperature was 1044 to 1450°F, and the burnups attained were 0.16 to 0.32 metal atom %.

The post-irradiation examination revealed no cladding damage or measurable dimensional changes, and nominal fuel growth of 1/2 to 5-1/2 mils diametral with no fuel-cladding interference. The hydrogen loss was low (2%) and indicated satisfactory barrier performance. Fission gas release from the fuel was very low (0.005% max).

The results of the NAA-121-1 tests showed a swelling correlation ($\% \frac{\Delta V}{V} - 3 B$) of 0.08% at 1200°F. The actual dimensional changes were as follows:

$$\% \Delta D/D = 0.39\% (1150^{\circ}\text{F}) \text{ to } 1.04\% (1420^{\circ}\text{F}).$$

$$\% \frac{\Delta V}{V} = 0.83\% (1150^{\circ}\text{F}) \text{ to } 2.75\% (1420^{\circ}\text{F}).$$

- (3) The NAA-120-4 capsule irradiation test contained a large number (17) of specimens which attained burnup of about 0.85 metal atom % in the Hanford reactor. ⁽⁴⁾ The cladding was Hast-X-285 with SCB-1 ceramic barrier. The cladding temperatures were up to 1300°F. The fuel was 10% U-ZrH_{1.70} to 1.85 and operated at temperatures of 1300 to 1600°F. The post-irradiation examination showed that the cladding was intact on all the specimens. The swelling of the fuel specimens ranged from 4.2% $\Delta V/V$ for H/Zr = 1.71 to 27.76% $\Delta V/V$ for H/Zr = 1.49 and 25.6% $\Delta V/V$ for H/Zr = 1.26 (see Table 15, Reference 4).
- (4) The NAA-115-2 capsule irradiation test contained twelve sublength S8DR-design fuel elements which were exposed for 10,000 hours in the Hanford reactor. ⁽²⁾ The cladding was Hast-N, and the clad maximum temperature was 1400°F. The fuel was 10% U-ZrH_{1.6} to 1.85 and the fuel temperatures ranged from 1300 to 1600°F. The burnup attained was 0.3 metal atom %. The fuel was found to have swelled from 1.5 to 4.5 % $\Delta V/V$. The average diametral changes ranged for 0 to 0.35% and the length changes were from 0 to 0.12%. Eight of the elements were in excellent condition, and four had cladding fractures which were attributed to fuel swelling.

- (5) Irradiation Capsules NAA-Series, Numbers -43; -53; -75; -115-1; -116-1; -116-2; and -117-1. These irradiation tests included a wide variety of fuel compositions, burnups, and fuel temperatures. (5,12) Some of the pertinent data are summarized in Table II as examples of the irradiation behavior of the U-Zr hydride fuels.
- (6) SNAP-4 Capsule Irradiation Data. A series of capsule irradiation tests (NAA-43-3, 5, and 6) of UZr-hydride fuels were carried out during the period 1959-1964 in support of the SNAP-4 reactor concept. The samples used in these tests were 10% U-ZrH_{1.08} to 1.41 and were very small (0.187-in. diameter). The burnups attained were very high (up to 2.15 metal atom % over a 2 year exposure), and the fuel temperatures were as high as 1000°F. All the samples survived intact and no diametral deformation of the fuel was observed. The length increases of the samples were reported to be less than 1%. Perhaps the small size and low temperatures of the samples were factors in the remarkable stability of the fuels in these tests.

VI. HYDROGEN-BARRIER

The 2-mil-thick ceramic barrier coating on the inner surface of the cladding was designated SCB-1. It was applied as an aqueous slip, air dried, and fired five minutes at 2125°F. This barrier has performed satisfactorily in fuel elements when there was no excessive fuel growth or cladding strain. Hence, the reactor environment does not appear to be a factor in barrier damage. The damage to the barrier which was observed in the S8DR elements was ascribed to fuel-to-barrier adherence, coupled with the differential motion between the fuel rod and cladding during reactor scram thermal transients. (3)

TABLE II (5,12)

SELECTED DATA FROM CAPSULE IRRADIATION TESTS

NAA							
<u>Test</u>	<u>U-content</u> (Weight %)	<u>H/Zr</u>	<u>Burnup</u> (Met. at. %)	<u>Temperature</u> (°F)	<u>Time</u> (Hours)	<u>Swelling</u> (% $\Delta V/V$) (% $\Delta D/D$)	
-43	10	1.0	2.05	768	9,652	7.1	--
"	"	"	1.89	---	9,652	7.1	--
"	"	"	3.35	730	13,370	7.3	--
-53	"	1.77	0.68	1300	2,076	1.3	--
"	"	"	1.03	1213	3,876	4.5	--
"	"	"	0.36	1515	1,385	2.7	--
-115-1	"	1.69	0.42	1550-1600	---	1.65	--
"	"	1.63	0.41	1560-1610	---	4.01	--
-116-1	"	1.6 to 1.9	0.15	< 700	---	1.60	0.59
-116-2	"	"	0.65	< 700	---	3.45	--
-116-1	19	1.76	0.29	500	---	1.56	0.59
-116-1	19	1.75	0.29	500	---	2.56	0.97
-117-1	10	1.55 $\rightarrow$ 1.35	0.77	1492	---	5.06	--
"	"	1.69 $\rightarrow$ 1.62	0.77	1496	---	2.85	--
"	"	1.69 $\rightarrow$ 1.53	0.77	1539	---	2.82	--
"	"	1.60 $\rightarrow$ 1.61	0.31	1511	---	1.06	--
"	"	1.70 $\rightarrow$ 1.44	0.31	1449	---	0.91	--
"	"	1.35 $\rightarrow$ 1.62	0.37	1287	---	1.07	--
-75	15% U + 6.5 % H _F	1.44 $\rightarrow$ 1.23	1.05	1075	5,253	3.77	--
"	"	1.45 1.48	2.15	1011	5,253	7.11	--
"	"	1.41 1.33	0.5	1059	2,955	4.94	--
"	"	1.40 1.52	1.26	914	2,955	4.11	--

Recent out-of-pile work at AI on the SNAP-reactor program includes studies of the creep-strain capability and the thermal cycling resistance of the barrier material on Incoloy-800 and Hastelloy-X cladding. ^(9,10) The Incoloy-800 fuel elements have accrued over 6000 hours of endurance testing at 1400°F, including ramp thermal cycles at intervals. The average leak rate at present is 0.25 cc (STP)/hr and the hydrogen loss is 2.3%. Another series of tests with Incoloy-clad elements lasted 20,000 hours at 1400°F and 29 ramp thermal cycles. The leak rates were 0.51 cc (STP)/hr and the average cumulative hydrogen loss was 8.42%. Hastelloy-X clad elements are being permeation tested at 1300°F and 1400°F. At 1000 hr exposure the permeation rates were approximately 0.25 cc (STP)/hr.

The ceramic barrier strain capability was measured under creep conditions in 25-mil Incoloy-800 cladding. ^(9,10) The average creep strains at failure were 3.2% at 1300°F and 0.22% at 1100°F

VII. FUEL CLADDING MATERIALS

Hastelloy-N was the fuel cladding material used for the SNAP-reactor fuel elements. During the past year two other alloys have been introduced as candidate materials for fuel cladding in the SPF reactor program, namely, Incoloy-800 and Hastelloy-X. These alloys have a number of advantages over Hastelloy-N, such as coatability with the ceramic barrier and more attractive strength and ductility properties in the irradiated condition. Work is in progress on evaluation of hobbing techniques for processing fins on the cladding surfaces, welding studies for final closure parameters for both Incoloy-800 and Hastelloy-X, and creep rupture strength determinations.

APPENDIX
SELECTED FIGURES AND TABLES SHOWING DATA ON UZrH FUEL

Fig. A-1. Comparison of design temperatures with deduced operating temperatures (from Ref. 9)

7759-51206

Hydrogen Distribution

7759-51207

Temperature Distribution

Fig. A-2. Element 7-16 EOL conditions (from Ref. 3)

7759-51211

Fig. A-3. Fuel growth in S8DR (from Ref. 3)

Fig. A-4. Comparison of design temperatures with deduced operating temperatures (from Ref. 3)

Fig. A-5. Zirconium hydride fuel swelling correlation compared with S8DR observations (from Ref. 1)

Fig. A-6. Creep properties of zirconium hydride; comparison of β with δ and $\delta + \epsilon$ phase material (from Ref. 1)

Fig. A-7. Zirconium hydride fuel swelling (from Ref. 1)

Fig. A-8. S8DR operating sequence; time zero to ~ 1000 hours (from Ref. 1)

Fig. A-9. S8DR operating sequence; time >1500 hours (from Ref. 1)

7759-51244

Fig. A-10. Fuel swelling correlation; hydrogen and burnup corrected (from Ref. 13)

7695-51405

Fig. A-11. Zirconium-hydrogen phase diagram (from Ref. 4)

Fig. A-12. Radial fuel growth vs time (based on fuel-cladding ΔT) (from Ref. 4)

Fig. A-13. Volume increase vs burnup for the 1300°F subcapsules (based on fuel-cladding ΔT) (from Ref. 4)

Fig. A-14. Volume increase vs burnup for the 1500°F subcapsules (based on fuel-cladding ΔT) (from Ref. 4)

Fig. A-15. Volume increase vs burnup for the 1600°F subcapsules (based on fuel-cladding ΔT) (from Ref. 4)

Fig. A-16. Fuel volume changes vs fuel diameter changes (corrected for hydrogen loss) (from Ref. 4)

7750-51148

Fig. A-17. Burnup-corrected volume change vs reciprocal calculated fuel temperature (from Ref. 4)

a. Zirconium-Hydrogen Phase Diagram

c. Variation of Degree of Tetragonality of ϵ Hydride With Hydrogen Content

b. δ and $(\delta + \beta)$ Zirconium Hydride Phase Boundary

Fig. A-18. Zirconium-hydrogen phase relationships (from Ref. 2)

Fig. A-19. Hydrogen dissociation pressures of zirconium hydride (from Ref. 2)

7568-51692

Fig. A-20. Specimen internal pressure from hydrogen dissociation for 1.85 H/Zr fuel (from Ref. 2)

7568-51693

Fig. A-21. Density diagram (from Ref. 2)

Fig. A-22. Fuel growth vs burnup with NAA-53, NAA-115-1, and empirical correlations (from Ref. 5)

Fig. A-23. Fuel growth vs burnup with NAA-53 and NAA-75 data (from Ref. 5)

Fig. A-24. S8ER fuel growth data (from Ref. 5)

Fig. A-25. Diametral fuel growth vs burnup with NAA-82-1, NAA-115-1, and S8ER data (from Ref. 5)

Fig. A-26. Diametral fuel growth vs temperature - $\delta/\delta + \epsilon$ group (from Ref. 5)

Fig. A-27. Diametral fuel growth, hydrogen, carbon, time, and temperature corrected (from Ref. 5)

Fig. A-28. Diametral fuel growth - unrestrained $\delta/\delta + \epsilon$ group (from Ref. 5)

Fig. A-29. Diametral fuel growth - $\delta + \epsilon/\epsilon$ group (from Ref. 5)

TABLE A-I
S8DR FUEL GROWTH CHARACTERISTICS
(As inferred from hydrogen-deduced
fuel temperatures) (from Ref. 3)

Element	Z/L	Hydrogen-Deduced Fuel Temperatures (°F)					Fuel Growth Parameters (See Table 2)		
		$\bar{T}_{BOL}$	$\bar{T}_{OFFSET}$	$\bar{T}_{Mwt}^*$	$\bar{T}_{Mwt}^\dagger$	$\bar{T}_{EOL}$	$(\Delta V/V)_{corr'd}$	Burnup (metal at. %)	Fuel Phase
2-06	0.39	1385	1356	1312	1274	1278	1.432	0.170	δ
	0.78	1460	1410	1389	1317	1326	3.267	0.129	δ
	0.87	1428	1395	1337	1287	1309	2.956	0.097	δ
	0.96	1390	1376	1272	1252	1287	1.857	0.060	δ
3-09	0.12	1211	1199	1082	1065	1117	0.829	0.091	ε
	0.33	1332	1303	1267	1234	1236	1.418	0.154	δ
	0.42	1379	1330	1336	1271	1267	2.271	0.168	δ
	0.51	1412	1349	1382	1295	1285	3.173	0.172	δ
	0.60	1439	1366	1415	1313	1302	4.083	0.167	δ
	0.69	1452	1376	1429	1320	1306	4.882	0.151	δ
	0.78	1450	1382	1421	1320	1308	4.991	0.131	δ
	0.87	1435	1381	1386	1306	1302	4.882	0.095	δ
5-05	0.96	1403	1376	1330	1287	1289	3.790	0.058	δ
	0.30	1325	1304	1243	1216	1217	0.996	0.136	δ
	0.39	1397	1363	1340	1296	1277	1.380	0.151	δ
	0.48	1464	1416	1434	1366	1330	2.381	0.157	δ
	0.57	1501	1435	1485	1382	1346	3.595	0.154	δ
	0.66	1520	1445	1507	1391	1354	4.606	0.143	δ
	0.75	1524	1455	1505	1400	1362	5.364	0.123	δ
	0.84	1507	1452	1476	1387	1359	5.710	0.097	δ
6-01	0.93	1474	1435	1423	1359	1341	5.510	0.065	δ
	0.12	1212	1200	1075	1058	1110	0.545	0.042	ε
	0.21	1278	1264	1174	1153	1176	0.795	0.074	ε
	0.30	1332	1322	1254	1240	1236	0.945	0.127	δ
	0.39	1380	1360	1326	1295	1276	1.214	0.141	δ
	0.48	1415	1386	1376	1328	1301	2.079	0.146	δ
	0.69	1460	1418	1430	1365	1330	5.183	0.129	δ
	0.78	1456	1416	1418	1356	1328	5.878	0.107	δ
6-02	0.87	1435	1401	1380	1328	1309	6.396	0.080	δ
	0.98	1401	1381	1324	1289	1285	5.566	0.049	δ
	0.09	1201	1189	1062	1042	1134	0.479	0.067	ε → δ §
	0.17	1253	1243	1138	1125	1209	0.579	0.093	ε → δ §
7-02	0.96	1405	1386	1335	1309	1318	3.964	0.049	δ
	0.15	1226	1214	1093	1080	1146	0.628	0.081	ε
	0.24	1286	1273	1185	1171	1215	0.710	0.104	δ
	0.58	1436	1377	1401	1311	1330	4.585	0.132	δ
7-31	0.87	1437	1418	1381	1357	1352	2.653	0.075	δ
	0.12	1199	1183	1050	1032	1112	0.578	0.072	ε
	0.21	1261	1247	1140	1121	1187	0.828	0.097	ε → δ §
	0.32	1330	1310	1233	1210	1262	1.062	0.120	δ
	0.42	1376	1337	1290	1241	1295	2.165	0.132	δ
	0.51	1411	1354	1335	1254	1310	3.679	0.136	δ
	0.60	1428	1375	1355	1274	1329	5.147	0.131	δ
	0.69	1450	1395	1373	1262	1307	6.434	0.118	δ
	0.78	1452	1402	1368	1281	1308	6.639	0.099	δ
	0.87	1431	1394	1327	1266	1298	6.825	0.075	δ
	0.97	1379	1365	1246	1220	1272	4.086	0.042	δ

*Fuel temperatures calculated at 1-Mwt conditions, assuming BOL fuel-to-barrier diametral gaps
†Fuel temperatures calculated at 1-Mwt conditions, assuming EOL fuel-to-barrier diametral gaps
§BOL phase was ε, with subsequent transition to δ; these points not used in fuel growth plot

TABLE A-II
COMPOSITION OF SCB-1 CERAMIC
HYDROGEN BARRIER
(from Ref. 4)

<u>Frit</u>	
Material	wt %
SiO ₂	45.2 ± 0.18
TiO ₂	12.6 ± 0.05
Al ₂ O ₃	7.2 ± 0.03
BaO ₂	35.9 ± 0.14
<u>Slip</u>	
Material	Weight (gm)
SCB-1 (Frit)	485.0 ± 1.95
Ferro Black Label Clay	15.0 ± 0.06
Chrome Oxide (Cr ₂ O ₃)	5.0 ± 0.05
Bentonite	1.25 ± 0.05
Distilled Water	350.0 ± 1.40

TABLE A-III

SUMMARY OF NAA 120-4 EXPERIMENT DATA
(from Ref. 4)

Specimen Number	Fuel Temperature (°F)		Beginning-of-Life		End-of-Life		Mass-Spectrographic Burnup (metal at. %)	Average % $\frac{\Delta V}{V}$ (hydrogen corrected)
	Design	Calculated Equilibrium Bulk Average	H/Zr	Phase *	H/Zr	Phase †		
E-1	1300	1293	1.85	ε	1.74	ε (δ + ε)	0.832	7.44
E-4	1300	1336	1.86	ε	1.81	ε (δ + ε)	0.859	5.38
E-5	1300	1317	1.86	ε	1.85	ε (δ + ε)	0.847	6.66
E-2	1300	1268	1.72	δ	1.73	δ + ε	0.816	4.69
E-3	1300	1293	1.72	δ	1.71	δ + ε	0.832	4.22
H-1	1500	1476	1.71	δ	1.66	δ + ε	0.832	9.57
H-3	1500	1489	1.70	δ	1.70	δ + ε	0.839	10.06
H-2	1500	1518	1.71	δ	1.59	δ + γ'	0.855	13.25
H-4	1500	1551	1.85	δ	1.74	δ + ε	0.874	8.40
H-5	1500	1585	1.85	δ	1.79	δ + ε	0.893	8.52
M-1	1600	1560	1.71	δ	1.49	δ + γ'	0.816	27.76
M-2	1600	1539	1.72	δ	1.44	δ + γ'	0.805	23.32
M-3	1600	1568	1.70	δ	1.26	δ + γ'	0.820	25.60
M-4	1600	1620	1.85	δ	1.69	δ + ε	0.847	13.68
M-5	1600	1620	1.84	δ	1.71	δ + ε	0.847	16.30
V-1	1500	1485	1.71	δ	1.49	δ + γ'	0.816	6.65
V-2	1500	1491	1.71	δ	1.62	δ §	0.851	16.96

*From phase diagram at operating temperature

†From postirradiation metallography (room temperature)

§Metallographic interpretation indicates V-2 entered β phase

TABLE A-IV
NAA-53 IRRADIATION TEST DATA
(from Ref. 5)

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 A-V
NAA-115-1 IRRADIATION TEST DATA
(from Ref. 5)

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

TABLE A-VI
NAA-43 IRRADIATION TEST DATA
(from Ref. 5)

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 A-VII
NAA-75 IRRADIATION TEST DATA
(from Ref. 5)

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.

TABLE A-VIII
NAA-82-1 IRRADIATION TEST DATA
(from Ref. 5)

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 A-IX
NAA-116-1 IRRADIATION TEST DATA
(from Ref. 5)

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

TABLE A-X
NAA-116-2 IRRADIATION TEST DATA
(from Ref. 5)

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 A-XI
NAA-117-1 IRRADIATION TEST DATA
(from Ref. 5)

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

TABLE A-XII
POST-IRRADIATION ANNEALING RESULTS — BETA-PHASE MATERIAL
(from Ref. 5)

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

TABLE A-XIII

FUEL SLUG DENSITIES AND DIMENSIONS
(from Ref. 12)

Capsule	Uranium (wt %)	Burnup (metal at. %)	Specimen	Diameter			Length			Volume			Density		
				Pre- Irradiation D_1 (in.)	Post- Irradiation D_2 (in.)	Change $\Delta D/D_1$ (%)	Pre- Irradiation L_1 (in.)	Post- Irradiation L_2 (in.)	Change $\Delta L/L_1$ (%)	Pre- Irradiation V_1 (cc)	Post- Irradiation V_2 (cc)	Change $\Delta V/V_1$ (%) $\pm 0.3\%$	Pre- Irradiation ρ_1 (gm/cc) $\pm 0.5\%$	Post- Irradiation ρ_2 (gm/cc) $\pm 0.2\%$	Change $\Delta \rho/\rho_1$ (%) $\pm 0.7\%$
116-1	10	0.15	14-2	0.2902	0.2918	0.55	0.7506	0.7521	0.20	0.8139	0.8245	1.30	6.287	6.209	-1.26
			15-2	0.2900	0.2916	0.55	0.7489	0.7504	0.20	0.8109	0.8215	1.31	6.068	5.995	-1.22
			16-2	0.2900	0.2917	0.59	0.7490	0.7521	0.41	0.8110	0.8240	1.60	6.052	5.983	-1.15
			17-2	0.2900	0.2920	0.69	0.7489	0.7509	0.20	0.8109	0.8238	1.59	6.074	6.000	-1.23
			18-2	0.2900	0.2914	0.48	0.7497	0.7506	0.24	0.8118	0.8206	1.08	6.071	6.002	-1.15
			20-2	0.2900	0.2926	0.90	0.7500	0.7531	0.41	0.8121	0.8302	2.23	6.184	6.091	-1.53
			21-2	0.2900	0.2922	0.76	0.7500	0.7524	0.32	0.8121	0.8271	1.85	6.121	6.035	-1.42
	20	0.29	19-2	0.2900	0.2917	0.59	0.7498	0.7527	0.39	0.8119	0.8246	1.56	6.546	6.451	-1.47
			22-1	0.2900	0.2928	0.97	0.7499	0.7545	0.60	0.8120	0.8328	2.56	6.580	6.400	-2.82
116-2	5	0.40	32-2	0.2900	0.2921	0.72	0.7496	0.7522	0.35	0.8117	0.8263	1.80	5.837	5.744	-1.61
			33-3	0.2900	0.2923	0.79	0.7495	0.7533	0.51	0.8116	0.8287	2.11	5.832	-	-
	10	0.65	24-3	0.2900	0.2936	1.24	0.7489	0.7535	0.61	0.8109	0.8363	3.13	6.093	5.857	-4.03
			25-2	0.2900	0.2942	1.45	0.7494	0.7552	0.77	0.8115	0.8416	3.71	6.361	6.161	-3.25
			26-1	0.2900	0.2936	1.24	0.7500	0.7572	0.96	0.8121	0.8404	3.48	6.096	5.853	-4.15
			28-2	0.2900	0.2939	1.34	0.7490	0.7568	1.04	0.8110	0.8417	3.79	6.101	5.812	-4.97
			29-3	0.2900	0.2940	1.38	0.7486	0.7536	0.67	0.8106	0.8387	3.47	6.166	5.908	-4.36
			30-2	0.2900	0.2938	1.31	0.7485	0.7538	0.71	0.8105	0.8378	3.37	6.152	5.886	-4.51

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