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REPORT TO THE FEDC WORKING COMMITTEE
FOR THE PERIOD
SEPTEMBER 1965 to JANUARY 1966

M. LEWIS

(with contributions from J. E. Minor, ^{PNL}~~BNWL~~)

January 5, 1966

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REPORT TO THE FEDC WORKING COMMITTEE FOR THE PERIOD
SEPTEMBER 1965 to JANUARY 1966

N-REACTOR FUEL PRODUCTION

Production Status

First run shop yields for CY-65 were 62.5% for outer elements and 68% for inner elements resulting in an overall ~~54.1%~~ yield. These numbers are significantly lower than previous first run yields because approximately 5% of the fuel was rejected as a result of autoradiography of the end closures. These fuels were refilmed after a 90-day holding period, during which time uranium daughter products decay, and over 90% were found to be completely satisfactory.

Other outer losses resulted from a high incidence of "dimpling" which was later resolved (see below). A small number of outer elements and inner elements were rejected because of an incident believed to be related to some high-melting braze rings. This subject is being investigated further.

Extrusion Surface Defects

About 2% (with large monthly variations) of N-outer coextrusions have been rejected because of a phenomenon known as "dimpling." This phenomenon manifests itself as an extreme local variation in ID cladding thickness and it is often associated with unbondedness. The principal cause has been attributed to chips or lumps of graphite or burned lubricant in the bore of the billet assembly. It is postulated that the hard graphite particles are trapped between the billet ID and the mandrel OD during upset, resulting in a depression of Zircaloy and copper. There are many potential sources for such graphite chips, such as from the graphite blocks used in the preheat cans, the graphite follower block and burned lubricant on the mandrel. During coextrusion the particles are crushed and the void filled with copper, thus retaining the impression in the Zircaloy, but not in the copper. Dimples have now been created deliberately on test coextrusions by inserting graphite particles of assorted sizes in the bore.

As an immediate measure to prevent dimple defects, the ID of all billet assemblies are being lubricated and dried prior to insertion in the preheat cans, the ID of all billet assemblies are being inspected for lumps or particles adhering to the walls, and all ID's are being blown out with an air lance while in the horizontal position on the billet loader prior to coextrusion. As a longer term solution to the problem, experiments will be made with copper foil seals on the ID of the billet to prevent contamination of the surface prior to insertion of the mandrel.

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Off-Site Coextrusion Backup Capability

In the unlikely event of major damage to the Hanford coextrusion press, it is estimated that the press could be out of service for as long as 16 weeks. To prevent a loss of production during such a period it is planned to assemble billets at Hanford, ship them to Ashtabula along with coextrusion tooling, coextrude the billet assemblies, and return the uncut coextrusions to Hanford for finishing. To permit rapid startup of the Ashtabula facility in the event of a Hanford outage, liners and containers sized for N-outer and N-inner coextrusions are being stored at Ashtabula. In addition, four outer and four inner billet assemblies prepared at Hanford were coextruded at Ashtabula in September. These eight coextrusions constituted a pilot run to be followed with ten coextrusions in each geometry after evaluation of the initial eight. The fuel sections from six of the initial eight coextrusions (two outers were rejected at peel test) have been processed through the balance of the fuel fabrication operations at HAPO. Fabrication was satisfactory so plans have been made to proceed with the additional 20 coextrusions in January 1966. The data obtained on wall thickness variation, fuel diameter, warp, heat treat length growth, and cladding and bonding were all comparable to the results obtained with coextrusions produced at Hanford with the exception of cladding thickness variation for outer fuels. The cladding thickness variation obtained on the N-outer extrusions fabricated at Ashtabula is somewhat better than normally obtained at Hanford. Since the sample size was small this may not be significant. The reason that the two outer extrusions failed peel test is unknown, but it is felt that the problem was related to shipping damage and not to the coextrusion conditions at RMI.

N-REACTOR IRRADIATION EXPERIENCEUranium Swelling

In the last report, preliminary data were presented on uranium swelling for inner and outer fuel elements from monitor columns that had reached tube average exposures in the range from 1554 to 1631 MWD/T. Those data have been subjected to review and correction and are again compared with the uranium swelling values previously attained in KER test irradiations of N-Reactor fuel elements.

In the interim, monitor columns have been discharged at tube average exposures of 1750, 2000, and 2250 MWD/T and this report summarizes swelling results for tube average exposures up to 2247 MWD/T.

Figure 1 shows a plot of uranium swelling values measured on 24 inner fuel elements. These elements are from four different monitor columns whose tube average exposure had varied from 1549 to 2187 MWD/T. The plot shows measured swelling ($\% \Delta V$) versus the arbitrary index of the product

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Figure 1. SWELLING OF INNER URANIUM FROM N-REACTOR

of individual element exposure times the volume mean fuel temperature in degrees Kelvin ($\text{MWD/T} \times ^\circ\text{K} \times 10^{-6}$). Superimposed on the plot are two lines that frame the limits of uranium swelling measured on 55 inner fuel elements from 4 KER test columns. In Figure 1, the data shown by circles were presented in the last report; new data are shown as crosses. The highest exposures of an N-inner fuel element on this plot is 2460 MWD/T for N-Reactor and 2460 MWD/T for KE Reactor.

Figure 2 shows the equivalent plot of uranium swelling values measured on 34 outer fuel elements from 6 monitor columns which attained 1605 to 2247 MWD/T tube average exposures. Again the limits of uranium swelling for 55 outer fuel elements from KER tests are framed by two solid lines for comparison. The highest exposures of N-outer elements on this plot are 3113 MWD/T for N-Reactor and 3090 MWD/T for KE Reactor.

The swelling data for inner fuel elements show comparable results from KER loop tests versus N-Reactor monitor columns. The swelling data for outer fuel elements are beginning to indicate that slightly lower swelling values are obtained in N-Reactor compared to those obtained in the KER loops. Although the reduction in swelling is not large, and may not be real, lower swelling values are predicted because the KER test irradiations were frequently on single-pass coolant rather than recirculating coolant. The 4 KER tests averaged 63 to 86% of the irradiation period in recirculating coolant and it was frequently suggested that the periods of single-pass coolant could lead to increased rates of grain boundary tearing.

Nearly 30 monitor columns containing weighed and/or measured fuel elements are awaiting examination in the recently completed Metal Examination Basin. Heretofore, the Basin has been operable for separating the outer and inner fuel elements in an assembly, weighing to determine relative exposures of elements in a column, and weighing. At the present time, an automatic dimensional gauge that was operated in the KE Metal Examination Facility has been set up to obtain O.D.'s and warp. Work is in progress to set up a length gauge and a nondestructive tester that will measure the soundness of the uranium core metal and the soundness of the cap-to-core and the clad-to-core bonds.

Fuel Element Failures

There have been 12 instances (for a total of 16) of fuel columns being discharged from N-Reactor for failure signals since the last reporting period. In 6 instances, the failures could not be confirmed by either careful visual examination or a fission product sniffing method. (1)

(1) The sniffing method uses an open-bottom bell jar from which the storage-basin water is displaced by argon gas from a high-pressure cylinder. The argon gas is subsequently displaced from the bell jar by basin water and the gas is passed through an activated charcoal filter that is monitored for fission-gas radioactivity.

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Figure 2. SWELLING OF OUTER URANIUM FROM N-REACTOR

Table 1 is a summary of all confirmed failures (10) and the instances when failures could not be confirmed (6) since the startup of N-Reactor.

Three of the most recent failures number 11, 15, and 16 bear a strong physical resemblance to each other. The first of these failures, number 11, was seen as a single, large blister that was centered under the arch of a support on the outer fuel element. Radiometallurgy laboratory investigation showed the point of water entry at the center of a missing volume of cladding under the support. A short chip of Zircaloy protruded from under the support and unmistakable evidence of fretting was present. Failure number 15 has a blister of almost identical appearance to that of failure number 14 and fretting is the suspected mechanism of this failure. Failure number 16 also shows a blister at a support. In this case though, the blister is semicircular in shape and is adjacent to one of the two tabs that is used to affix the support by projection welding. Although the cause of this failure cannot be positively stated, the strong physical resemblance again suggests fretting.

If failures 15 and 16 are confirmed to be caused by fretting, considerable insight will be gained in the incubation time necessary for this type of failure to develop. The columns of fuel elements involved in failures 15 and 16 were charged to the reactor at the beginning of December 1965 and had accumulated 8.5 and 14.5 operating days respectively prior to their failing.

Failures number 2 and 3 were also caused by fretting. The chromium wire that was known to have caused number 3 is suspected to have caused failure number 2 and came from the bonnets of primary loop valves that were being repaired over a period of several months prior to the dates of these two failures. The actual date on which the chromium chip fretting members became lodged on supports of inner fuel elements and were positioned to fret the inner claddings of the outer fuel elements cannot be accurately determined.

In the case of failure number 11, the Zircaloy fragment that was the fretting member has been identified as a right-triangular section sheared off the end of a support for an outer element. Investigation has shown that these sections are readily sheared off by collision of the supports on outer fuel elements with sharp bevelled edges of the baskets used to support fuel elements during autoclaving. Figure 3 is a schematic drawing of three general shapes of Zircaloy fragments that have been sheared from the ends of supports and were found in the autoclave baskets. (The shapes infer collision of the rectangular sections of the supports with bevelled edges on the circular openings in upper and lower plates in the baskets.)

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TABLE I

SUMMARY OF CONFIRMED AND UNCONFIRMED FUEL ELEMENT FAILURE IN N-REACTOR

<u>Failure No.</u>	<u>Date of Failure</u>	<u>Column and Element Exposure MWD/T</u>		<u>Element Type</u>	<u>Type of Failure</u>	<u>Cause of Failure</u>
1	7-7-64			Inner	End closure associated	Defective end cap
2	5-15-65	720;	475	Outer	Fretting - inner clad	Fretting member not found
3	6-6-65	1449;	809	Outer	Fretting - inner clad	Fragment of chromium plate
4	6-15-65	1087;		Driver	Corrosion of inner cladding	Thought to be F ⁻ contamination
5	7-23-65	1672;	--	--	UNCONFIRMED	--
6	9-1-65	1756;	--	--	UNCONFIRMED	--
7	9-3-65	1850;		Inner	Undetermined	Under current study
8	9-10-65	1549;	--	--	UNCONFIRMED	--
9	9-26-65	2170;	2430	Outer	Undetermined	Under current study
10	10-17-65	595;	--	Driver Prototype	UNCONFIRMED	--
11	10-25-65	91;		Outer	Fretting - outer clad	Zircaloy fragment
12	11-1-65	1643;	--	--	UNCONFIRMED	--
13	11-12-65	806;	--	Driver Prototype	UNCONFIRMED	--
14	12-12-65	2299;	2778	Inner	Undetermined	Under current study
15	12-16-65	73;	96	Outer	Associated with a support	Under current study
16	12-25-65	;	145	Outer	Associated with a support	Under current study

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Above: Half-section of outer fuel element supports

Below: Typical sheared sections

FIGURE 3
SCHEMATIC OF SHEARED SECTIONS OF
OUTER FUEL ELEMENT SUPPORTS

Considerable effort has been devoted by Battelle Northwest Laboratory personnel to metallographic inspection of failure number 7. Their study has not succeeded in determining either the point of water entry or the mechanism of the failure. The failure is characterized by a complex split in the outer cladding that is axially aligned from the base of one end cap and extends to the nearest set of supports. The same crack branches and becomes circumferentially oriented at the base of the end cap and extends over nearly three-quarters of the circumference. The uranium fuel core incurred extensive fragmentation but was not badly corroded by reaction with the water coolant. However, when the

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fuel element was cross-sectioned near the end of the axial crack, the pieces of core metal fell out in evidence of either prior unbonding or extensive corrosion attack of the compound layer between the core and the cladding during the failure.

Examination of the outer and inner claddings has disclosed nothing unusual with the exception of some shallow depressions on the inner cladding and areas of very thick intermetallic-appearing compound where the uranium and zirconium intermetallic compounds are usually found.

One of the major cracks in the uranium core passed through a plane containing both shallow depressions in the inner cladding. This fracture path suggests that hard spots in the uranium were instrumental in first causing the depressions and then serving as stress risers to direct the crack in the uranium matrix.

The lack of general corrosion of the uranium fragments suggests that the uranium may have fragmented during irradiation rather than during the failure event. In nondestructive tests of the soundness of the uranium cores of N-Reactor fuelelements irradiated in the KER loops, cracks were found near the ends of fuel elements. Those cracks were radially oriented and extended axially for a distance of nearly two inches beneath the end caps. To determine if macrocracking of the uranium cores will progress as exposure levels increase, attention is now being directed to nondestructive testing of the soundness of irradiated core metal.

The thick intermetallic compound layer remains unidentified. Samples of the cladding with layers nearly two mils thick have been submitted for gamma spectral analysis but results indicate only "zirconium contaminated with fission products." A fuel element that was immediately adjacent to the failed element in the coextrusion has been removed from storage and is being nondestructively re-examined and tested for conformance to specifications. It is planned to section this element and examine the intermetallic compound layer between the uranium core and the Zircaloy cladding and the nature and distribution of second phase impurity particles in the uranium.

Preliminary study of the fracture surfaces in the cladding of failure number 9, a failed outer element, has failed to disclose either the point of water entry or the cause of the failure. The uranium has incurred the most corrosion damage at a location near one of the supports but the support does not appear to have been involved. The fuel core was not badly corroded during failure but structural integrity was adversely affected and the element incurred gross damage as a result of free-fall discharge. Numerous cracks and fractures that resulted from collision and impact during discharge made the analysis of this failure very difficult.

Failures number 14, 15 and 16 are awaiting shipment to Radiometallurgy after cooling to allow decay of fission product radioactivity.

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The group of elements in the class of unconfirmed failures are being subjected to special tests to determine why certain of the elements have liberated fission products to the coolant but have not incurred visible damage. One may conclude that in these supposed failures, the uranium core metal has not been attacked by the coolant but the elements have sloughed zirconium oxide that is enriched in uranium. The Zircaloy claddings have less than 5 ppm of uranium content but the brazed plus welded end closure has considerably more uranium contamination. Specifications to limit the contamination of uranium in the welds were enforced and monitored by autoradiography and alpha counting with the former indicating the intensity and location of areas of contamination and the latter the total level of all contaminated areas. The specification for the first load of fuel elements was one weight percent uranium (10,000 ppm uranium by weight). Since the specification was enforced without benefit of large-scale testing to assess the effectiveness of the limit, attention has been subsequently directed to laboratory corrosion tests of the ternary alloy system Zr-2 + Be + U. Results have been obtained on alloys with three levels of Be content (1, 2, and 3 w/o) and five levels of uranium (0.1, 0.5, 1, 2 and 4 w/o) in autoclave tests where there has been some replenishment of water. Weight gain data that were obtained for 300 and 360°C water after two weeks and four weeks are given in Table 2.

The data of Table 2 show that both Be and U contamination cause a decrease in corrosion resistance of Zircaloy-2 and there is a change from a coherent zirconium oxide to a sloughing oxide at uranium contaminations greater than 1 w/o.

To be more applicable to in-reactor fuel element performance, data should be obtained on the same ternary alloy compositions in 250°C water the effect of neutron irradiation on corrosion rates should be determined. However, in their present form the data show the importance of minimizing uranium contamination and trying to develop brazing and welding techniques for the thinnest possible braze layers. (The thin braze layers lead to lower Be contents in the metal alloy that is mixed by the welding process.)

To pursue the hypothesis that uranium contamination in the welds is causing spurious fuel element failure signals, a device has been constructed in which the end closures of irradiated fuel elements from unconfirmed failure columns may be acid etched while the elements are shielded with storage basin water. An open bottomed bell jar is supported on three-foot high legs (resting on the floor of the storage basin) and is emptied of storage basin water by purging with argon gas. Inside the bell jar is a small dish which can be filled with acid (approximately 30% nitric acid with ammonium bifluoride). After the bell jar is blown down with argon, the acid dish is filled and the irradiated element is stood on end in the dish. The etching solution is pumped into plastic bottles and later analyzed by gamma spectral methods for fission products and activation products. The end-etching device was made operative late in December and with 21 feet of covering water represents a safely operating device to locate closures that are contaminated with uranium.

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TABLE 2
WEIGHT GAIN DATA FOR Zr-2: Be: U
TERNARY ALLOYS IN pH 10 WATER (NH₄OH)
AFTER TWO WEEKS

(Autoclave Tests with Replenishment of Water)

300 C Water

<u>w/o Uranium</u>	<u>1 Percent Be</u>	<u>2 Percent Be</u>	<u>3 Percent Be</u>
0.1	26.5 mg/dm ² (36.5)	35.6 mg/dm ² (47.5)	36.2 mg/dm ² (53.0)
0.5	23.8 mg/dm ² (33.2)	35.9 mg/dm ² (41.7)	44.4 mg/dm ² (63.1)
1.0	25.9 mg/dm ² (39.6)	39.9 mg/dm ² (61.1)	48.0 mg/dm ² (75.2)
2.0	47.6 mg/dm ² (84.1)	59.8 mg/dm ² (100)	58.1 mg/dm ² (78.2)
4.0	70.4 mg/dm ² (100)	75.5 mg/dm ² (85.3)	89.7 mg/dm ² (-114)

(xx.x) represents weight gains after 4 weeks exposure.

Note: Samples below the dashed line are undergoing sloughing corrosion.

360 C Water

<u>w/o Uranium</u>	<u>1 Percent Be</u>	<u>2 Percent Be</u>	<u>3 Percent Be</u>
0.1	39.9 mg/dm ²	42.5 mg/dm ²	55.3 mg/dm ²
0.5	37.7 mg/dm ²	57.7 mg/dm ²	59.7 mg/dm ²
1.0	67.1 mg/dm ²	63.6 mg/dm ²	86.4 mg/dm ²
2.0	151 mg/dm ²	155 mg/dm ²	137 mg/dm ²
4.0	295 mg/dm ²	268 mg/dm ²	225 mg/dm ²

Note: Samples below the dashed line are undergoing sloughing corrosion.

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N-REACTOR FUEL DEVELOPMENT

Upset-Forging of Uranium Billets

Upset-forging of raw billet blanks to finished dimensions continues to be evaluated as a potential primary fabrication process for N-Reactor uranium. In this process the cast uranium ingot is extruded to a diameter less than the desired finished billet size, but into billets, the billet surfaces conditioned by localized removal of residual surface defects, and the billets reinserted in the extrusion press and upset to finished dimensions. The finished billets are then pickled in nitric acid, inspected, and shipped to Hanford for coextrusion. This process eliminates billet machining and enables a larger portion of the case ingot to be utilized as finished material.

Seventeen coextrusions were made with upset-forged billets up through September. Evaluation of the bonding, general and localized cladding variation, etc., indicate the coextruded fuel sections to be equal to or better than standard production material.

Eleven additional ingots were processed by RMI in October and November to provide more forged billets for conversion to N-outer fuel tubes. This material was processed from ingots cast to an OD of 13-1/2 inches and an ID of 4.7 inches. It was necessary to bore the ID of the billet blanks prior to forging to correct excessive wall thickness variation originating in the primary extrusion. Surface defects on the billet blanks were removed by localized grinding or, occasionally, complete surface machining. No post-forging surface defects were disclosed by inspecting the finished billets in the pickled condition. A total of 46 billets have been received at Hanford. Four were shipped to NLO for metallurgical evaluation. In addition, seven short "nose" billets are being processed by RMI for evaluation. These billets were cropped as little as necessary rather than at the first full cross-section of the primary extrusion.

Preliminary evaluation of the processing data by RMI indicates that wall thickness was controlled and ID eccentricity was limited to 30 mils at the billet centers. All billets met or slightly exceeded the design weight. The apparent yield (based on assuming a non-oversized ingot) was about 86% of the finished ingot weight and about 75% of the initial furnace charge weight.

Three of these later billets have been satisfactorily coextruded, but the detailed evaluation is not complete. Further extrusion tests are scheduled for late January 1966. A development order of upset-forging billets for N-inner fuel tubes has been placed.

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Fabrication of Alloy Fuels

The program to evaluate the fabrication and irradiation performance of dilute uranium alloys is continuing. Six coextrusions have been made with uranium enriched to 1.25% in U-235 and containing 350 ppm iron and 800 ppm aluminum. These fuels were fabricated in the N-outer geometry for use as spike enrichment. Fabrication of this alloy went satisfactorily. The only significant difference was an improvement in clad thickness variation that has also been found in previous alloy coextrusions. Clad thickness variation for the six test coextrusions was compared with that for the six spike extrusions processed immediately before the test fuels. The control fuels were of standard alloy (150 ppm iron, 100 ppm silicon).

	<u>Clad Thickness Variation (Mils)</u>	
	<u>Alloy Fuels</u>	<u>Control Fuels</u>
I. D.	4.44	5.19
O. D.	2.68	3.52

Additional coextrusions are planned in this geometry and with other dilute alloys of iron, silicon, and aluminum.

In preparing these alloy billets for coextrusion, a large fraction of the billets on hand show cracks on the outer surface that were probably initiated by billet heat treatment. These cracks appear only in those alloys with relatively high silicon contents which do not also have high aluminum content. They were not apparent upon pickled billet inspection at the feed site so there is a possibility that some cracking may be caused by post-heat-treatment metallurgical changes in conjunction with residual quenching stresses.

Coextrusion and evaluation of the alloy test billets on hand should be completed in February 1966.

Larger scale production testing of the 350 Fe-800 Al alloy is planned. The planned demonstration of coproduct fuel in N-Reactor will utilize uranium of this composition. Swelling data obtained at HAP0 indicates that this alloy will swell only about one-half as much as the swelling expected with the standard alloy (150 ppm iron, 100 ppm silicon), under coproduct exposure conditions.

Production test quantities of iron-aluminum alloy billets in N-outer tube size have been ordered to serve as a basis for establishing a firm capability at the feed sites and in the Coextrusion Shop and to provide N-Reactor evaluation of a significant test quantity in this geometry. Fifteen tons per month of the N-outer tube requirement will be procured. In addition, a single quantity of ten tons will be procured in the N-inner size to provide some experience with this geometry in the event that higher exposures in tube-tube alloy are required.

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Hanford has requested that the majority of the billets to be provided from direct casting development at NLO also be in the 350 Fe-800 Al alloy. Alternate alloying practices to supplant double melting are being investigated with emphasis on a larger scale demonstration of adding aluminum in the ingot mold as developed by NLO.

Uranium Alloy Capsule Irradiations

Studies are in progress to correlate irradiation performance and corrosion behavior with composition and fuel fabrication history for a number of uranium-base alloys.

The irradiation step is complete for all 120 specimens of 16 different alloys in various heat treatments. A list of the alloys with their nominal compositions and heat treatments are shown in Table 3. Each specimen was 0.6" diameter x 3.0" long and coextrusion clad with 0.025" Zr-2. The irradiations were conducted in NaK capsules with three specimens to a capsule and eight capsules to a column. Each of the five columns had a temperature-monitoring thermocouple capsule at the downstream end. Flux distribution was measured along each column of capsules. From the temperature and flux measurements the mean fuel temperature and exposure for each specimen was calculated.

All 120 specimens have been visually examined and their diameters measured. Fifteen suffered cladding failure to the extent that their diameters could not be measured. The diameter measurements have been reduced to a $\% \Delta \text{Area}$ and then normalized to an R value for $\frac{\% \Delta \text{Area}}{\% \text{B.U.}}$.

The R values were plotted against volume mean fuel temperature for a comparison of the effect of heat treatment and composition. From these plots the following trends (not conclusions) are indicated:

1. The reference alloy (see Table 3) exhibits a swelling maximum at 425-500°C with R values of 15, followed by a decrease to a minimum in the temperature range from 500-550°C with R values of 7 to 8. From 550°C to 575°C, the upper temperature limit of this irradiation, the swelling increases again to an R value of 15.
2. For the low alloys (<1 wt% of any one additive) there is no consistent indication of greater swelling resistance for a β , $\beta + \alpha$, or γ heat treatment. For some specific alloys there are indications of definite heat treatment effects.
 - a. FACL: $\beta + \alpha < \beta$
 - b. SAC: $\beta < \beta + \alpha$ in 550-600°C - No data at < 550°C
 $\beta < \gamma$ in 425-475°C
 $\beta \approx \gamma$ in 550-600°C

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TABLE 3
URANIUM ALLOYS IN CAPSULE PROGRAM

<u>Alloy</u>	<u>Designation</u>	<u>Composition</u>					<u>Heat Treatment</u>	<u>Recrystallization and Hot Hardness Evaluation</u>
		<u>Fe</u>	<u>Al</u>	<u>Si</u>	<u>C</u>	<u>Other</u>		
1*	Reference	305	150	185	762		β	x
2	FACL	1053	895	150	470		β , $\beta+\alpha$, γ	x
3	FACS	390	810	30	400		β , $\beta+\alpha$, γ	
4	FSC	225	60	140	1545		β	
5	A	-	790	-	-		β , $\beta+\alpha$, γ	
6	S	-	-	660	-		β , $\beta+\alpha$, γ	x
7	SC	-	-	530	480		β , $\beta+\alpha$, γ	
8	SA	-	325	225	-		β , $\beta+\alpha$, γ	
13	P	-	-	-	-	470P	β	
16	SAC	-	960	315	525		β , $\beta+\alpha$, γ	
17	MS	100	190	350	460	500Mo	β , $\beta+\alpha$, γ	x
9**	Z	180	-	-	480	2.4 wt% Zr	γ , $\gamma+\alpha$	x
10	ZA	-	725	-	-	2.6 wt% Zr	γ , $\gamma+\alpha$	x
11	ZS	100	-	60	-	2.5 wt% Zr	γ , $\gamma+\alpha$	
14	NA	-	800	-	-	2 wt% Nb	γ , $\gamma+\alpha$	x
15	MA	-	1060	-	425	1.75 wt% Mo	γ , $\gamma+\alpha$	x

*Heat treatments for alloys 1 through 8, 13, 16, 17.

β : 730 C quench to 300 C in salt.

$\beta+\alpha$: β above + 600 C for 100 hours in vacuum furnace.

γ : 800 C quench to 300 C in salt.

**Heat treatment for alloys 9, 10, 11, 14, 15.

γ : 800 C quench to 300 C salt.

$\gamma+\alpha$: γ above + 550 C vacuum anneal for 100 hours.

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3. For the higher alloys:
 - a. NA and MA: $\gamma < \gamma + \alpha$ in 400-450°C
 - b. NA and MA: $\gamma \approx \gamma + \alpha$ in 575-600°C
4. Comparing swelling with γ heat treatment for both high and low alloys:
 - a. MA and NA < low alloy 400-500°C range
 - b. MA, SAC, and MS < other low alloys in 550-625°C range.
5. Those alloys which exhibited the least swelling and for which there are sufficient data over a temperature range from 400 to 600°C are shown below with R values.

Alloy	Heat Treatment	R = $\frac{\% \Delta \text{Area}}{\% \text{B.U.}}$	
		Temperature 400-500°C	Temperature 500-600°C
MA	γ	7.5	9.0
MA	$\gamma + \alpha$	8	9.0
SAC	β	8-10	8.5
FACL	$\beta + \alpha$	8.5	10.5
MS	γ	11.5	12.0

Certain of the alloys were selected for a study of their unirradiated recrystallization and hot hardness behaviors to determine if a correlation could be found between these properties and their irradiation swelling behavior. Hot hardness measurements have just been completed on the alloys as indicated in Table 3 at 300 and 400°C with three different loads at each temperature. These results are being analyzed. Recrystallization studies are not in progress on these same alloys to establish their recrystallization temperatures as determined by hardness and metallographic observation.

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Defect Behavior of Alternate Uranium Compositions

Small additions of iron, silicon and aluminum to uranium are being evaluated in a program designed to develop a more swelling-resistant fuel material than the present 150 Fe - 100 Si N-fuel composition. One facet of this program is to determine if these additions significantly affect the corrosion behavior of uranium in 300°C, 1500 psig water. To evaluate corrosion behavior, test specimens, prepared from prototypical, coextruded, Zircaloy-2 clad material of the N-single tube geometry, are defectected by drilling a 0.025" diameter hole through the outer cladding. The behavior of the defectected specimens are observed in a windowed autoclave. The windowed autoclave has a chamber 3" ID x 8" deep and a design rating of 3000 psig @ 650°F (343°C). In addition to direct visual observation, two cameras are used to record the behavior on film for further evaluation. One camera is used for time-lapse studies, the other is a standard 16 mm camera with a telephoto lens.

Specimens have been prepared from the five compositions being evaluated. These compositions cover a range of addition level from 150 to 350 ppm Fe, 0 to 350 ppm Si, and 0 to 800 ppm Al. Each composition is being studied in both beta and gamma heat treated conditions.

"Typical" behavior of a standard 150 Fe - 100 Si composition sample is that initially corrosion products, oxide and hydrogen gas, issue from the defect for the first half hour at which time the defect becomes plugged and a blister forms. The blister continues to grow for another half hour until it is about a half inch in diameter at its base at which time it begins to split around its base and the oxide pushes this cap up to where it is nearly normal to the clad. Associated with this cap-lifting is a sudden release of corrosion products. The corrosion rate, as determined by hydrogen gas collection, steadily increases as the area around the defect continues to swell. Between 3-1/2 to 4 hours after the start of the test, secondary splits open above and below the initial blister. No sudden increase in corrosion rate is associated with this secondary splitting. After about 4 hours at temperature the oxide from the defect has built up to where it obscures observations and the run is terminated. Examination of the specimen shows a blister area approximately 2 inches in diameter and about 0.4" high in the outer clad and that the inner clad has buckled inward about 0.3". No undercutting of core to clad bond area was observed. No significant difference in behavior was observed between beta and gamma heat treated material.

Beta and gamma heat treated specimens of 250 Fe - 250 Si- 250 Al composition, the only other composition to be defect tested to date, was found to behave similarly to the standard 150 Fe - 100 Si composition.

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