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FROM DOUGLAS UNITED NUCLEAR, INC.

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REPORT TO THE WORKING COMMITTEE
FROM DOUGLAS UNITED NUCLEAR, INC.

by:

J. T. Stringer
Fuels Engineering Subsection
Research and Engineering Section

January 11, 1967

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SMALL AND K REACTOR PRODUCTION REVIEW

Uranium Core Quality Data

Grain Size

The average grain size of salt-water and induction-oil heat-treated cores received during the period August 1, 1966 through November 31, 1966, was unchanged and was between A-6 and A-7.

Inspection of Bare Cores

The combined visual and grain size reject rate of bare cores was 0.23 percent for the period September 1, 1966 through December 15, 1966.

Small and Large-Grain Cores

Five O3N fuel cores from ingot Z-64295 were rejected by the UT-2 testers as large-grain rejects. When the cores were manually retested, tester signals indicated that areas of small grains were mixed unhomogeneously in a matrix of large grains. This phenomenon is thought to be either 1) a segregation of chemistry in the ingot or rod and/or 2) heat transfer from the core when it was quenched. Studies are being conducted to determine the cause of such a phenomenon. Pictures of the core and core cross-sectional area, Figures 1 and 2, show that there are areas of small grains (A-6) surrounded by large grains (A-2).

Uranium Fuel Performance

Evaluation of Core Blank Diameter

This test was designed to determine if any significant differences in dimensional stability under irradiation exist between uranium fuel cores fabricated under similar conditions but machined from heat-treated core blanks of two different diameters.

The test was composed of 228 test elements and 288 control elements. The test lot contained heat-treated core blanks 1.437 inches in diameter while the control lot contained heat-treated core blanks 1.410 inches in diameter. Both the test and control core blanks were then machined to the O3N core diameter of 1.356 inches.

Goal exposure for the test was 600 MWD/T. At discharge, the average exposure was 594 MWD/T for the control lot and 589 MWD/T for the test lot. Average tube outlet temperature for all tubes in the test was 105.2 C. Average dimensional changes for the two groups are summarized in Table I.

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FIGURE 1

03N Fuel Core from Ingot Z64295 Etched to Show Grain Structure

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FIGURE 2

Cross-Section of 03N Fuel Cores from Ingot Z64295 Exhibiting Patches of Large Grains and Small Grains

Large Grain Size	~ (A - 3)
Small Grain Size	~ (A - 6)

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TABLE I

Dimensional Changes, In. x 10⁻³

<u>Lot</u>	<u>ΔOD₁</u>	<u>ΔOD₂</u>	<u>ΔOD₃</u>	<u>Warp</u>
Test (1.437 in. core blanks)	-4.2	0.44	-0.19	4.7
Control (1.410 in. core blanks)	-4.6	0.22	-0.39	5.2

Based on the results of this test, it was concluded that core blank size within the limits of the test had little significant effect on dimensional stability during irradiation.

Irradiation Performance

The fuel performance continues to be excellent. Four fuel element failures occurred during the period September 27, 1966 through January 9, 1967. These failures are summarized in Table II.

TABLE II

Fuel Element Failures
September 27, 1966 - January 1, 1967

<u>Fuel Element Type</u>	<u>Reactor</u>	<u>Exposure (% of Goal)</u>	<u>Classification</u>	<u>Failure Date</u>
Natural Self-Support AlSi	KW	115	End Manufacturing Defect	10-21-66
Natural Non-Bumpered AlSi	D	33	Mechanical Damage	11-24-66
Natural Overbore HDS	C	83	Side Unclassified	12- 8-66
Natural Non-Bumpered AlSi	KE	93	Side Hot Spot	12-21-66

The hot-die-size overbore fuel failure which occurred on December 8 was very similar in appearance to the overbore failure which occurred on May 26 and which was described in the October 1966 Working Committee Report. The December failure was characterized by a blister approximately 1/2 inch wide and 1-1/2 inches long at the end of a support. The point of water entry was not obvious, and the cause of the failure has not yet been determined.

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Figure 3 shows the trend in fuel performance over the past two years. The abrupt improvement at the start of 1966 was due largely to the increased use of bumpered fuel in the B, C, and D Reactors and an improvement in the end bonding of the K Reactor self-supported fuel. The figure shows the number of failures per one million fuel elements discharged for all fuel models except the C Reactor overbore model. This model was excluded as changes in the fuel and the goal exposure make a comparison between present and previous performance more complicated. However, during 1965, there were 12 failures in this model, seven in AlSi-canned elements and five in hot-die-sized elements. During 1966 there were two overbore failures, both in hot-die-sized fuel. All of the overbore fuel is now fabricated by hot-die-sizing.

DEVELOPMENT REVIEW

Reactor Modernization

The first irradiation of elements for the reactor modernization program has been completed. In this program it is proposed that process channels be enlarged one inch and the present I&E fuel geometry be replaced with a rod-in-tube geometry. In the irradiation just completed, 24 assemblies were irradiated in the KER facility to an average exposure of 650 MWD/T. The dimensions of the fuel are:

Tube:

Fuel O.D.	2.530 inches
Core O.D.	2.430 inches
Fuel I.D.	1.295 inches
Core I.D.	1.402 inches

Rod:

Fuel O.D.	0.996 inches
Core I.D.	0.896 inches

The irradiation proceeded satisfactorily, but the elements have not yet been examined. It is planned to continue testing in the KER facility through 1967.

High Power Density Fuel Irradiation

Five columns of AlSi-bonded elements of 1.25 enriched uranium were irradiated in the KW Reactor during the last quarter of CY 1966. The columns were irradiated to an average exposure of 1050 MWD/T. The fuel elements operated at specific powers up to 135 kw/ft., or about 150 percent of current conditions. All cores contained the standard DUN alloy, which is 100 to 200 ppm Fe and 75 to 140 ppm Si.

Maximum fuel temperatures of about 405 C were experienced as compared with 350 C, average core temperature of .947 enriched fuel. No evidence of

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FIGURE 3 - 1965 AND 1966 FUEL PERFORMANCE

cavitation swelling was found in any of the elements nor was any unusual uranium behavior discovered during the destructive examination. The AlSi bond behaved normally in all respects.

The 8001-F aluminum cladding corroded uniformly at a maximum rate of about 8.5 mils per month. At the conclusion of the irradiation, a minimum of about 60 percent of the original clad bonding thickness remained. The uniform corrosion attack, with observed radial penetration variation of only ± 0.004 inches from average, was expected to be about eight mils per month.

Thoria Wafer Program

The October 1966 FEDC Report reported that higher closure weld yields on ThO_2 target elements could be obtained by using the combination of a heat-sinking chuck and a thoria wafer placed immediately below the end cap of the element. Therefore, three tons of GENLO wafers (approximately 7.8 g/cc dense) were ordered for use in the current thoria campaign to increase the closure weld yields.

Two groups of wafers were used in this campaign. Both groups of wafers had a thickness of 0.370 ± 0.025 inch and a density of approximately 7.5 to 7.8 g/cc. One group had a diameter of $1.330 \text{ inch} \pm 0.012 \text{ inch}$ and the other had a diameter of $1.365 \text{ inch} \pm 0.012 \text{ inch}$.

The combination of the thoria wafer immediately below the end cap and the heat sinking chuck substantially reduced the number of weld closure pinholes and blowouts. The yield percentages of this current campaign versus the yield percentages of the previous thoria campaign are summarized in Table III and Table IV. The quality of the thoria wafers was satisfactory. The data available thus far characterizing the thoria wafers are summarized in Table V and Figure 4.

Figure 4 is a graph which shows the results of the mercury penetrometer test on the pressed thoria wafer. The data show that the wafer has an apparent density of 10.00 g/cc and that there was little change in density at pressures up to 20 psi because the void space in the wafer is smaller than nine microns in diameter. Sixty-two percent of the void space in the wafer is between 2.0 and 0.06 microns in diameter.

Zirconium Plating of Uranium

As discussed in prior reports, attempts to produce adherent zirconium plates on uranium metal from a LiF-NaF-KF-ZrF_4 electrolyte have included the use of copper nickel-plated uranium substrates. Good quality zirconium plates adhered very well to the substrates; however, the substrates did not adhere to the uranium. Re-examination of earlier specimens which had been zirconium-plated over a nickel-uranium substrate revealed that it was not the zirconium which had peeled off the nickel-uranium substrate, but that the nickel plate had apparently split. Half of the nickel remained on the uranium metal and the other half adhered to the zirconium. This behavior is attributed to the formation of a brittle U-Ni compound, probably U Ni_2 . C. L. Angerman, Trans.

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TABLE III

Yields on Current Thoria Campaign Versus Previous Thoria Canning Campaign

Campaign	First Run Yield	Reclaim Weld Yield	Inspection Yield	Over-all Yield
	No. Elements Accepted	No. Elements to Radiograph	No. Elements to Storage	Total Elements to Storage
	Total Elements Canned	Total Elements Canned	No. Elements Inspected	Total Elements Canned
11- 1-66 to Present (Wafers Used)	93.8%	98.1%	93.7%	95 %
10-14-65 thru 3-18-66 (No Wafers)	+ --	89.6%	86.5%	84.6%

*Based on what data were available on present campaign but not indicative of the complete campaign.

+This number is unknown.

TABLE IV

Percent Rejects after Final Weld Inspection

Campaign	Percent Rejected for Helium Leaks	Percent Rejected for Voids	Percent Rejected for Oxides
10-14-65 thru 3-18-66 (No Wafers)	0.2	8.2	5.10
11- 1-66 thru Present (Wafers Used)	0.14	2.75	3.34

TABLE V
Data on Thoria Wafers

<u>Lots</u>	<u>Model</u>	<u>Wafer/lb.</u>	<u>Dissolution</u>	<u>LOI</u>	<u>U ppm</u>	<u>Wt. of H₂O Absorbed</u>
W-1	010T	7.05	99 %	-.005	5.7	
W-2	010T	6.96	97.2%	+	4.0	.06* grs.
W-3	010T	7.08	99 %	+	3.5	
W-4	010T	7.03	99 %	+	3.0	
W-5	010T	6.98	99 %	+	3.1	
W-6	010T	7.01	99 %	-.002	2.8	
W-7	010T	6.98	99 %	+	3.9	.08
W-8	CLT	6.68	99 %	+	4.2	.08
W-9	010T	6.96	99 %	+	-	.07
W-10	CLT	6.59	99 %	+	4.0	.09

*The average weight gain in 24 hours by wafers placed at room temperature in an atmosphere saturated with H₂O.

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Assc., 54, 267 (1961). As a result of these observations, work was resumed on plating of zirconium directly on uranium.

The ability to produce fluoride melts of increased purity by the use of an HF-H₂ sparge removed some of the problems which had been present in previous attempts to plate zirconium onto uranium. A series of experiments which was done in a LiF-NaF-KF melt containing approximately three wt. percent ZrF₄ at temperatures from 650 to 800 C and at cathode current densities from 10 to 20 A/cm² showed that it is possible to produce an adherent and non-porous, albeit thin, coating of zirconium on a bare uranium substrate. The most interesting results of this set of experiments were: 1) the ability to produce an adherent zirconium plate on uranium at a temperature of 650 C was demonstrated; 2) the plate which was produced at 800 C remained adherent at room temperature.

While the ability to plate at 650 C and adherence of the zirconium plate to the uranium substrate were encouraging, problems remained with the morphology of the zirconium plate itself. Only that part of the zirconium which extended for a distance of one or at most two mils from the uranium was of the proper character. From there on out, the plate became more and more granular until finally the zirconium was deposited in the form of individual crystals which were very easily removed from the surface. This morphology seemed to be independent of concentration or temperature except that at higher temperature the crystals were larger.

Preliminary results of a metallographic examination of samples of zirconium-plated uranium seem to indicate that the reason for the drastic change in the morphology of the zirconium plate must be something other than the formation of a uranium-zirconium alloy substrate. Extensive diffusion of zirconium into the uranium does take place, however. Before any definite conclusions can be drawn, a more careful examination of the samples will be necessary.

Initial results of electroplating experiments which were done in CaF (65 mole percent)-LiF (35 mole percent) (m.p. 475 C) electrolyte were very encouraging. It was possible to produce non-porous, adherent zirconium plates up to 25 mils in thickness at temperatures as low as 625 C. Only a small fraction of the zirconium went to form modules or needles. In a few cases, at temperatures greater than 625 C, corrosion products formed on the surfaces of the uranium and these prevented the adherence of portions of the plate.

Nickel Plating of Uranium

Previous work on the theoretical aspects of pulsed current plating indicated there was considerable difference between anions in various nickel-plating solutions. One of the important factors was relaxation times, which is the time required for the voltage to drop to zero after it has been applied for a specified time. Considerable difference was found in the relaxation times of the sulfate and perchlorate systems, the sulfate system decaying much faster than the perchlorate system. The relaxation times of the sulfamate and bromide solutions fall between these two extremes. The following experiments were designed to establish correlation between relaxation times and the stresses of specimens plated by a pulsating current. During the

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course of making these stress measurements, several different nickel-plating solutions were studied for stress under standard direct current plating. The results of these experiments are tabulated below:

TABLE VI

Stress Measurements Table

<u>Anion</u>	<u>A Standard D.C. Current Plating Condition (P.S.I.)</u>	<u>A Standard Pulsed Current Plating Condition (P.S.I.)</u>
Sulfamate	13,000	34,500
Perchlorate	28,200	45,200
Sulfate	35,800	57,200
Bromide	39,200	41,400
Chloride	58,000	50,300

Comparison of the perchlorate and sulfate systems indicates that the relaxation times have little effect on stress. In general, specimens plated by a pulsed current had a much higher stress than those specimens plated by direct current. The pulsed current plating experiments also showed that plates with a small grain size have very high internal stresses.

Several stress experiments were carried out on direct current plating. These results are tabulated below.

TABLE VII

Stress Measurements Table

<u>Bath</u>	<u>pH</u>	<u>Stress (P.S.I.)</u>
Watts	2.0	31,900
Watts	4.0	25,400
Bromide	2.0	30,800
Bromide	4.0	15,800
Sulfamate (Cl ⁻)	4.0	30,000

This data shows there is little difference between the Watts and Bromide baths at a pH of 2.0. The bromide bath shows a marked decrease in stress when the pH is raised to 4.0; the Watts bath also shows a decrease but not as much as the bromide.

High Pressure End Bonding for Hot-Die-Sizing

Development of a high-pressure, low-temperature cycle was undertaken to increase equipment throughput capability of the hot-die-sizing process. Feasibility

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testing was completed⁽¹⁾ and a statistical test is currently in progress to provide quantitative data for evaluation. Results to date indicate no loss of fuel quality because of the higher end bonding pressure (six tsi vs four tsi) or lower bonding temperature (380 to 420 C vs 560 to 580 C). Bond strengths on the flat ends of the test fuel were comparable to those produced on standard production fuels. Bond thicknesses of the cap end of the fuel are, on the average, the same for both the standard and the high-pressure end bonding process. The base end bond thickness on the high-pressure fuel, however, was found to be much thinner than that observed on standard fuels. A comparison of end bonds of standard and test fuel is shown in Figure 5. X-ray diffraction work indicates that the increase in end bonding pressure has had little effect on longitudinal growth indices of the fuels comparison of which is shown in Figure 6.

Anodized Spacer Elements for Process Tube Protection

Anodizing of aluminum spacers used in the downstream non-irradiated portion of Zircaloy process tubes has been found to protect the process tubes from hydride formation. The non-conductive anodic film serves to prevent coupling between the aluminum spacer and the Zircaloy and subsequent hydride formations which reduces tube life. Because of the need for anodizing both new and recycle spacers, an anodizing facility was installed. Although recycle spacers have been decontaminated, they still bear a slight amount of residual reactivity requiring process control. Production of anodized spacers is underway as authorized by provisional process specifications (DUN-1650). Process development for the purpose of producing a more continuous and uniform anodic film is continuing. A mathematical model of the anodizing step showing the relationship between the independent variables (current density, electrolyte solution concentration, anodize time, and electrolyte temperature) and the dependent variables (anodic film thickness and electrical resistance) has been developed for both new and recycled spacers. Response surfaces developed from the mathematical models are shown in Figures 7 through 10. Adequate precleaning of both new and recycled spacers was achieved in all cases with a 20 to 25 weight percent caustic etch at 70 to 100 C followed by a cold nitric acid dip. Best anodizing conditions within the limits of the test were found to be 70 minutes anodize time at 20 amp/ft.² current density in a seven weight percent oxalic acid solution maintained at about 18 C. Under these conditions, average film thickness and resistance values of 0.002 inch and 17,000 ohms, respectively, were obtained. The above mentioned cleaning process was not used initially for 313 production, but was adopted two weeks after startup because of poor film resistivity resulting from inadequate cleaning. The mathematical model has been extremely valuable in maximizing production capacity within the range of parameters required for acceptable film quality.

Corrosion Studies

It has been found that treating autoclaved aluminum-clad fuel elements in a relatively strong caustic solution produces surface attack patterns that

(1) DUN-1878, "Interim Report, High-Pressure, Low-Temperature End Bonding of Hot-Die-Size Fuel," G. M. Schweikhardt, 1/10/67. (Confidential)

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FIGURE 5 - GROWTH INDEX FOR HIGH PRESSURE END BOND

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CAP END

BASE END

STANDARD PROCESS

CAP END

BASE END

HIGH PRESSURE PROCESS

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FIGURE 6

DIFFUSION THICKNESS COMPARISON

— 30 Minutes
 - - 50 Minutes
 - - - 70 Minutes

FIGURE 7

1% Oxalic Acid 4% Oxalic Acid 7% Oxalic Acid

Three Dimensional Graphs Showing, for the Anodizing of New Spacers, the Relationship Between the Independent Variables - Oxalic Acid Concentration, Temperature, Current Density, and Time - and the Dependent Variable - Anodic Film Resistance.

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FIGURE 8

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— 30 Minutes
 - - 50 Minutes
 - - - 70 Minutes

1% Oxalic Acid

4% Oxalic Acid

7% Oxalic Acid

Three Dimensional Graphs Showing, for the Anodizing of Recycled Spacers, the Relationship Between the Independent Variables - Oxalic Acid Concentration, Temperature, Current Density, and Time-and the Dependent Variable - Anodic Film Resistance.

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FIGURE 10

Three Dimensional Graphs Showing, for the Anodizing of Recycled Spacers, the Relationship Between the Independent Variables - Oxalic Acid Concentration, Temperature, Current Density, and Time-and the Dependent Variable - Anodic Film Thickness

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resemble localized corrosion patterns seen on fuel elements irradiated in the Hanford reactors. The objective of this work is to develop ex-reactor test conditions by which the irradiation performance of the fuel may be predicted. Unlike prior corrosion tests, the accelerated corrosion tests are conducted by means of chemical reaction with the cladding with no mechanical or electrolytic effects.

In these and prior tests, it was found that autoclave films had a significant effect on corrosion behavior of the fuels. It was also apparent that corrosion behavior of the fuel could not be predicted based on visual inspection of the autoclave films involving continuity, color, apparent damage, etc. The accelerated test has been used solely for comparative purposes to evaluate a variety of fuel preparation techniques. A standard time, temperature, and concentration has been established for the test, and fuel surface protective properties are measured in terms of grams metal loss. Visual inspection of the tested sample provides qualitative evidence of film uniformity, film damage, etc.

Figures 11 through 15 indicate typical corrosion behavior of fuels tested under varying conditions.

Groove corrosion response of fuels included in effluent activity tests is being recorded in K Reactor. A recent water treatment test in K Reactor gave evidence that groove corrosion may be related to the standard reactor water treatment. Tests are now underway to establish any relationship between turbidity and groove corrosion. A newly acquired turbidity instrument is monitoring K Reactor influent and effluent to determine its relationship to corrosion.

Work in Corrosion Chemistry Laboratory is currently directed toward measurement of Zircaloy-aluminum alloy couple potentials and currents as well as galvanic properties of couples between various aluminum alloys. Potentials up to 0.7 volt have been measured in Zircaloy-aluminum couples while aluminum-aluminum couples have shown 0.1 to 0.25 volt. This work is being combined with Zircaloy tube hydriding studies.

Tests of coupons and fuels are underway in an ex-reactor flow tube to determine the anodic and cathodic effects of imposed electrical potential on fuels in a flowing system using reactor cooling water. These tests will be followed by test conditions simulating the galvanic couple potentials and currents which are being determined in the measurements described above. Ultimately it may be necessary to instrument fuels for in-reactor measurements.

Dichromate Autoclave Film Tests

Previously reported test fuels prepared to evaluate the corrosion resistance of dichromate autoclave films were found to have a high incidence of surface pitting and blemishes. Approximately 75 percent of the fuels prepared for irradiation testing exhibited one or more pits averaging approximately two mils in depth but extending to seven mils in depth. There was no preferential location on the fuel and the pitting did not appear to be grossly biased to any one autoclave treatment. Investigation of the pitting included rerunning one of the cycles in a laboratory autoclave. Six fuels were processed as follows:

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FIGURE 11

Original rod and tube fuel elements removed from a caustic test solution after an estimated 40 minutes. Note the protective properties of a good autoclave film and how those properties are locally destroyed by roll sizing (wide bands) and normal handling techniques. As seen on irradiated corrosion samples, groove corrosion patterns on the test pieces are formed by a succession of pits, sometimes linear axially, sometimes linear tangentially, and other times randomly oriented.

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FIGURE 12

Evaluating the role of residual stress as a cause of the corrosion patterns seen, certain fuels were selected after autoclaving and roll sizing and subjected to a stress-relieving process (600 C for 10 minutes). The relieved piece (upper) reacted in the bath immediately and produced a satin etch finish. The other fuel shown displays corrosion patterns caused by a partially destroyed protective autoclave film.

FIGURE 13

Degreasing of the fabricated fuel element (after rail attachment and before autoclaving) is not normally employed except where excessive hydrocarbons are seen. A comparison of degreased (left) and non-degreased (right) shows very little difference topographically except in the immediate rail area where residual oil has prevented formation of an autoclave film thus allowing more severe test attack.

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— AUTOCLAVING —
NORMAL CYCLES

WEEKEND
70 hrs

WEEKDAY
20 hrs

FIGURE 14

Reflects resistance to corrosion as a function of autoclave cycle time. Although more metal has been etched away on the short cycle piece, the over-all corrosion is uniform in nature. Thus far preliminary tests indicate that there may be an optimum cycle time beyond which no additional benefit is realized. Tangential irregularities approximately mid-fuel are attributed to film damage caused by the autoclave liners.

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CrO₂ COATINGS

FIGURE 15

Elements of this test were subjected first to a demineralized water autoclave cycle and then a K₂Cr₂O₇* solution autoclave cycle at temperatures shown below:

	<u>Water Cycle, °C</u>	<u>K₂Cr₂O₇ Cycle, °C</u>
Group A	170	170
Group B	290	170
Group C	290	230
Group D	340	230

*8-10 w/o K₂Cr₂O₇, pH 6.6-6.8

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Deionized water, 25 to 30 hours at 290 C

Dichromate solution, 25 to 30 hours at 170 C, at pH 6.65

Fuels were held in 8001 alloy aluminum sleeves (which had been autoclaved at 290 C) and were contained in a stainless steel bucket. Chemicals (potassium dichromate and potassium hydroxide) were the same as used for preceding pilot plant runs. No unusual corrosion was observed following the water autoclaving portion of the cycle. Examination after dichromate autoclaving revealed three pieces out of the six with black spots on the lateral surface similar to those from the preceding pilot plant run. Portions of the closure welds were blackened, and also exhibited shiny areas, resembling active corrosion. The fuels which show no evidence of surface irregularities are being segregated for irradiation testing. All autoclave conditions will be represented in the irradiation test.

High Silicon-Aluminum Alloy Test

Evaluation of data from production test PTA-010, designed to evaluate the performance of fuels clad in 10 w/o silicon-aluminum alloy, indicates that the high silicon-aluminum alloy did not have a greater resistance to ledge and groove corrosion than the standard 8001 aluminum alloy. It was reported previously that the test was discharged from D Reactor on September 4, 1966 when the second of two ruptures was incurred by the high silicon-aluminum-alloy-clad fuel. Two other high silicon alloy-clad fuel elements showing evidence of cladding penetration were found during examination of the irradiated fuel. The test design included individual, segregated columns of hot-die-size process 8001 cladding, hot-die-size process high silicon alloy cladding, and AlSi process 8001 cladding fuels. Alternate piece striped columns permitted comparison of the high-silicon alloy with 8001-clad hot-die-sized fuel and high silicon alloy with 8001 clad AlSi process fuel.

It should be noted that the early laboratory tests which showed the 10 w/o silicon-aluminum alloy to have a greater corrosion resistance than 8001 (by about a factor of two) were based on uniform corrosion and pitting, whereas the failures and test data are based primarily on ledge or groove corrosion. Statistical tests (t-Test with five percent level of significance) performed on the non-uniform corrosion data obtained from the test show no significant difference in frequency or severity of ledge and groove corrosion between the three types of elements tested. However, there was a significant difference noted in the weight loss per element, with a greater weight loss occurring in both 8001 alloy and the high-silicon alloy clad by the hot-die-sizing process than in the AlSi elements. This difference was significant in both the striped and segregated columns. Differences noted on the striped columns are possibly more meaningful than those in the segregated columns since tube variables (tube power, flow, Δt) are essentially eliminated in the striped column comparisons. There is the possibility, however, of interaction between fuel types in the striped columns.

The following table contains data from the 25 test columns which were discharged in September, and did not include four columns which were discharged early for a preliminary examination.

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TABLE VIII

PTA-010 - Ledge and Groove Corrosion Summary

	Hot-Die-Size Process	ALSi		
	High Silicon	8001	Process	
	Alloy	Alloy	8001 Alloy	Total
<u>All Test Columns</u>				
Number of Elements in Test	150	110	110	370
Number of Elements Examined	141	106	106	353
Primary Ledge or Groove	76	48	57	181
Corrosion				
Corrosion Severity > 10 Mils	30+	21	16	67
<u>Segregated Columns Only</u>				
Elements in Segregated Columns	70	70	70	210
Elements Examined	63	68	67	198
Primary Ledge or Groove	36	32	39	107
Corrosion				
Corrosion Severity > 10 Mils	17+	14	13	44
<u>Striped Columns Only</u>				
Elements in Striped Columns	80	40	40	160
Elements Examined	78	38	39	155
Primary Ledge or Groove	40	16	18	74
Corrosion				
Corrosion Severity > 10 Mils	13	7	3	23

NOTE: Unless otherwise stated, ledge or groove corrosion > 10 mils is in the range 10 to 20 mils.

+: Includes three elements with severity > 20 mils.

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Corrosion Program

A routine monitoring of the irradiation performance of both hot-die-size and AlSi canned fuel was started in mid 1966. One of the principal objectives of the program is to compare the corrosion behavior of the two fuel types. To date, nearly 2,000 elements have been examined; approximately 1,500 of these were irradiated in the K Reactors and 320 in B Reactor.

The data from B Reactor showed no difference between the two fuel types with respect to ledge and groove corrosion; 30 percent of the AlSi and 33 percent of the hot-die-size fuels showed ledge or groove corrosion, none of which was over ten mils deep.

The data from the K Reactors show a difference between the AlSi and hot-die-size fuel. Of the hot-die-size pieces examined, 16 percent had primary groove corrosion and 10 percent primary ledge corrosion. Of the AlSi pieces examined, three percent had primary groove and seven percent ledge corrosion. None of the AlSi-canned fuel examined had non-uniform corrosion deeper than 10 mils, and six percent of the hot-die-size pieces have shown groove corrosion deeper than 10 mils. On one hot-die-size piece the cladding was completely penetrated and the element was considered an incipient failure.

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