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CLAD THICKNESS VARIATION N-REACTOR FUEL ELEMENTS

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N-REACTOR FUEL ELEMENTS

E. A. SMITH

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CLAD THICKNESS VARIATION N-REACTOR FUEL ELEMENTS

INTRODUCTION

The current specifications for the cladding on "N" fuels were established early in the course of process development and were predicated on several basic considerations. Among these were: a) a desire to provide an adequate safety factor in cladding thickness to insure against corrosion penetration and rupture from uranium swelling stresses; b) an apprehension that the striations in the zircaloy cladding of the U/zircaloy interface and on the exterior surface might serve as stress-raisers, leading to untimely failures of the jacket; and c) then existing process capability— the need to maintain a specified ratio between zircaloy and uranium in the billet assembly to effect satisfactory coextrusion.

It now appears appropriate to review these specifications in an effort to determine whether some of them may be revised, with attendant gains in economy and/or operating smoothness.

SUMMARY, CONCLUSIONS AND RECOMMENDATIONS

Experimental studies of pertinent problems were reviewed and from them the following conclusions are drawn:

1. From the standpoint of corrosion resistance in N-Reactor environment, at least a 50 percent reduction in clad thickness is feasible.

2. Owing to the high pressure on the element surface and the swelling resistance of the uranium alloys used, the need for an extremely strong jacket to restrain uranium swelling is diminished, at least for low exposures.

3. While a single longitudinal scratch or valley in the clad surface (either interfacial or external) may act as a stress raiser increasing the tendency to fracture under internal radial stresses, this tendency is counteracted by the presence of a multiplicity of such valleys (as exemplified by the normal striations appearing in the jacket walls).

4. Notch acuity, which largely determines the concentration of stresses at the apex of the defect, becomes less important as the number of notches (striations) increases, since the total stress is distributed among the several notches rather than being concentrated at the single apex.

5. Because of the many variable factors involved in the behavior of notched zircaloy, specific dimension limits for striations cannot be practically applied. Normally striated, relatively hydrogen-free zircaloy clads remain sufficiently ductile to accommodate internally generated strains without rupture during extended periods of irradiation. It is probable that one striation groove having considerably greater than average depth or acuity compared to its neighbors might bring about sufficient stress concentration to cause rupture, especially if environmental conditions were such as to cause considerable embrittlement (e.g., hydrogen pickup). However, such extremes occur only rarely, if ever. They could not be identified by existing nondestructive test methods even if encountered.

6. More important than notch depth in determining fracture tendency of zircaloy clads is the residual thickness; i.e., the thickness of metal lying between the apex of the notch and the opposite surface or interface.

7. Fracture tendency of zircaloy clad appears to be unaffected by temperature within the range of 290° - 515°C.

8. The feasibility of reducing the nominal clad thickness on N-fuel elements depends principally on three factors:

   a. Ability to control in the process of fabrication, the uniformity of clad thickness (including uniformity of striations on the surface and at the U/zircaloy interface).

   b. Ability to preserve ductility of the zircaloy during fabrication and irradiation.

   c. Ability to control the magnitude of internal stresses and strains developed within the uranium core during irradiation.

   With these factors controlled to the degree obtaining normally in current production and for irradiation to normal exposures, it appears that the clad thickness could safely be reduced to .020 inch nominal for the outer clad of the inner element. This is a thickness reduction of $\frac{.012}{.032}$ or 37 1/2 percent.

9. To maintain the balance of restraint between the external and internal claddings, the latter should presumably be reduced proportionately, giving about .012 inch for the nominal internal clad thickness.

10. Any applied reduction in clad thickness will automatically increase the severity of the effect of normal clad thickness variation since the variation will represent a larger fraction of the total clad thickness. This makes more urgent the development of devices and techniques for nondestructively measuring these variations, and eliminating from production channels those elements exhibiting excessive variation.

The following action is recommended:

1. Experimentally determine the feasibility of altering extrusion tooling and procedures, and the extrusion component dimensions to produce coextruded clad stock with thinner cladding.

2. Develop techniques for nondestructively monitoring the output of this revised extrusion technology for quality and uniformity.

3. Apply thorough ex-reactor and in-reactor evaluation tests to samples of the revised product.

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4. WHether or not efforts are made to reduce the clad thickness, steps should be taken to maintain tighter control over the factors that determine cladding behavior (See conclusion No. 8 above).

5. Accept elements that are marginal with respect to current clad thickness specifications, to be irradiated in the less rigorous zones of the reactor or to lower exposure. This practice can improve the economy of the operation.

DISCUSSION

The basic function of the cladding on uranium fuel elements is, as is well known, to protect the uranium core against corrosion and destruction by the coolant liquid. For this function, zircaloy-2 clad must have a certain minimum thickness. It is also known that the cladding helps to restrain the uranium against swelling during irradiation, although the additional thickness required to serve this function cannot be determined from available data.* Early criteria[1] for coextruded zircaloy-2 clad fuel elements specified a minimum thickness of 0.040 inch for the outer clad of the inner tube and 0.025 inch for all other cladding, along with the recommendation that these values be reduced as soon as accumulated experience showed such reduction to be feasible. The engineering limits for clad thickness have been revised several times; currently the specified minima are as follows:[2]

Inner tube	outer clad:	0.032" with 0.010" allowable variation
	inner clad:	0.017" with 0.008" allowable variation
Outer tube	outer clad:	0.017" with 0.008" allowable variation
	inner clad:	0.017" with 0.012" allowable variation

Inner clad variation of between 0.012" and 0.021" qualifies as Class II.

A powerful incentive to the reduction of clad thickness is the cost of zircaloy, which at present accounts for over half the total cost of all the materials used in fabricating coextruded fuels. On the other hand, the high cost of an element failure[3] militates against taking any steps that will increase the risk of incurring failures. Three questions arise:

1. Are present dimensions adequate for operating contingencies?

2. Are present specifications adequate to prevent acceptance of deficient elements?

3. Can steps be taken to reduce clad thickness without jeopardizing the operating life of the elements, and without upsetting the coextrusion operation as the zircaloy/uranium ratio is changed?

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* These considerations are discussed in the Appendix.

1 R&E Section, IPD, "NPR Fuel Element Criteria, etc.," HW-65023, May 3, 1960.
2 Coextrusion Process Specifications, CX-660.0, HW-69941 Rev.
3 Curtiss, D. H., "Projected Cost for Fuel Ruptures," RL-NRD-65 (12/28/64).

[redacted]

Zircaloy-2 was chosen for cladding material because, among other favorable features, it is normally highly resistant to corrosion in pH-10 water at a 300-360°C temperature. When properly coated with the black hydrated oxide film supplied by suitable autoclave treatment, the corrosion rate is only about .00035 inch per year* under the above conditions.

Thus, from the standpoint of corrosion alone, a zircaloy-2 cladding thickness of 0.010 inch would be adequate to protect the substrate uranium for a period of approximately 28 years under conditions anticipated for N-Reactor.

The rupture resistance of the cladding, however, is a different matter, being affected by a variety of conditions. Pure zircaloy-2 is quite ductile at ordinary temperatures, but the presence of relatively small concentrations of hydrogen and/or other impurities can cause embrittlement of the zircaloy[4] to such an extent that the expanding or distorting forces of the uranium core during irradiation may lead to splitting of the jacket. Such tendency to rupture under radial internal forces may be augmented by the presence of "stress-raisers" in the form of extrusion striations on either the exterior or zircaloy/uranium interfacial surfaces of the cladding. There is a further embrittling action resulting from irradiation[5,11] which tend to increase the tensile strength and notch sensitivity of the zircaloy with increasing exposure in the reactor, although this effect is partially offset by the increased ductility due to high temperature operation.

In an effort to obtain quantitative values for some of these factors, the writer made a search through literature on the subject. Some data useful in calculating quantitative values was found,** as well as considerable enlightening information on the subject. The following are pertinent points:

1. Zirconium and some of its alloys, including zircaloy-2 are quite ductile in the pure state, but are markedly embrittled by the addition of even small concentrations of certain impurities, including oxygen, nitrogen, and particularly, hydrogen. The latter may be absorbed from a hydrogen-bearing environment at a significant rate beginning at about 235°C and at increasing rate with increasing temperature, to a maximum rate at about 375°C.[6] The rate of absorption is also influenced by the concentrations of hydrogen in the environment and in the absorbing body.

2. Pure zirconium (and zircaloy-2) does not work-harden rapidly.[7] Thus, in contrast to the behavior of most metals, a neck-in occurring under tensile

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* Calculated from data given in E. Hillner, "Hydrogen Absorption in Zircaloy During Aqueous Corrosion," WAPD-TM-411, November, 1964.

** See Appendix.

4 Kirk, W. W., et al., Zirconium Highlights - WAPD-ZH-15, WAPD-ZH-21, Mudge, W. L., Jr Effect of Hydrogen on Embrittlement of Zirconium and Zr-Sn Alloys, WAPD-T-20, 11/15/52.

5 Bement, A. L., et al., Quarterly Progress Report, Metallurgy Research Operation, Third Quarter, 1963, HW-78962, pp. 4, 58.

6 Lustman and Kerze, "Metallurgy of Zirconium," McGraw-Hill, 1955, p. 283.

7 Ibid, p. 502.

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stress is not arrested because of work-hardening, with a shift of the strain to adjacent regions. Rather, continued application of stress causes continued necking in the same region, with eventual rupture.

3. Absorption of hydrogen by zirconium and zircaloy-2, with formation of hydrides principally at grain boundaries, rapidly reduces ductility and tends to increase ultimate tensile strength as illustrated in Table I.[8] Under certain conditions, the presence of notches in the specimen also greatly reduces the possible amount of elongation under tensile stress.

4. The effect of stress-concentrators[9,10] in causing failure of a specimen under tensile stress depends on many factors, and is variable with different materials, different rates of loading and different sizes and geometries. In general, the concentration of stresses increases with increasing acuity* of the notch and with decreasing ductility of the specimen. The notch effect may ordinarily be lessened by removal of material from the specimen to reduce the acuity of the notch, or by introducing additional notches, whereby the total stress is distributed among several locations rather than remaining focussed at one point.

5. There is some disagreement as to the notch-sensitivity of zircaloy-2. Wheeler[11] decided it is not notch-sensitive because the striations (extended notches) in his test specimens did not promote rupture under heavy hoop-stress, and because puncturing the specimen with a sharp instrument while under heavy hoop-stress did not cause propagation of the breach initiated by the puncture. On the other hand, Chalk River personnel[12] state that zirconium and its alloys are very notch-sensitive. This conclusion is based ~~on the results~~ of Weber's studies, discussed below, as well as those of Armour Research Foundation.[8]

6. Wheeler[11] and others[13] found that zircaloy-2 is further embrittled and strengthened by irradiation above approximately 10^{20} nvt exposure.

7. Studies by several individuals, including Wheeler,[11] Leggett,[14] and Weber,[15] show clearly that uranium swelling during irradiation is restrained by the combination of external applied forces, including cladding tensile restraint, and coolant pressure. With high coolant pressure, the role of the cladding in restraining the swelling is lessened.**

8. Weber[16] made an extensive study of the effect of clad thickness variations on the tendency to rupture during irradiation. Although his specimens were small

* See Appendix, Item 1.
** See Appendix, Item 5.
8 Weinstein, D., and Holtz, F.C., "Delayed Failure Hydrogen Embrittlement of Zirconium," ARF-2230-12 (10/10/62) pp. 21, 22.
9 ASTM Standards, Part 3, 1958, p. 70.
10 Lipsen, Chas., and Juvinall, Robert C., "Handbook of Stress and Strength," MacMillan, 1963, pp. 28, 29, and 38.
11 Wheeler, R.G., "Burst Test of Zircaloy-2 Fuel Element Cladding," HW-64803 (4/15/60), p. 3.
12 Biefer, Howe, Sawatzky, and Krenz, "H-Pickup in Zirconium," AD232022, CRMet 849, p. 4
13 Metallurgy Research Quarterly, Third Quarter 1963, HW-78962, pp. 4.61 ff.
14 Leggett, R.D., Data appearing in Quarterly Report, Metallurgy Research, 10,11, 12/64
15 Weber, JW, "Final Report, Irradiation of Zircaloy-2 Clad U-Rods," HW-67072 (10/62).
16 Weber, JW, "Quarterly Progress Report", Metalurgical Development Operation, Beginning First Quarter 1962, HW-72347, p. 4.6 and ending First Quarter 1964, HW-81484, p. 2.62.

[redacted]

TABLE I[8]

DYNAMIC TENSILE PROPERTIES OF ZIRCALOY-2
AT ROOM TEMPERATURE

Condition	Ultimate Tensile Strength, psi	Yield Stress psi (0.2% Offset)	Total Elongation, %
	Lot W		
Unnotched, vacuum-annealed	45,300	32,000	41.5
Unnotched, 200 ppm H_2	47,100	31,800	34.5
Unnotched 500 ppm H_2	63,200	39,600	33
Notched, vacuum-annealed	64,200	44,300	(10.9)*
Notched, 200 ppm H_2	62,200	41,800	(6.2)*
Notched, 500 ppm H_2	58,800	47,300	(4.2)*
	Lot B		
Unnotched, vacuum-annealed	60,300	-------	------
Unnotched, 200 ppm	64,600	44,100	33.7
Unnotched, 500 ppm	71,800	49,100	28.9
Notched, vacuum-annealed	84,400	-------	------
Notched, 200 ppm	78,300	62,800	(3.5)*
Notched	63,400**	-----**	(2.7)*

* Values taken from load-extension curve; deformation confined to area around base of the notch.
** Load-extension curve showed straight line to fracture.

8 Op. Cit.

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TABLE I (Continued)

Condition	Ultimate Tensile Strength, psi	Yield Stress psi (0.2% Offset)	Total Elongation %
	Lot R		
Unnotched, vacuum-annealed	74,100	65,000	30.6
Unnotched, 500 ppm	77,900	59,900	26.6
Notched, vacuum-annealed	96,700	86,800	(4.7)*
Notched, 500 ppm	59,800**	------**	(1.8)*

* Ibid)
** Ibid) See Footnotes, Page 7.

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in size, were solid rods, and were irradiated in a bath of NaK within a closed capsule, (hence, not certainly representative of fuel element behavior in a water-cooled reactor) their qualitative behavior may nevertheless be considered as illustrative of certain general principles. The writer has taken Weber's published data[17] and worked them up into graph form (Figure 1). From this and Weber's temperature data (not amenable to graphing), the following results are evident:

a. <u>All</u> specimens having less than 0.015 inch residual clad were split failures (five out of five).

b. <u>All</u> specimens having less than 0.018 inch residual clad were either splits or severely necked (29 out of 29).

c. Eighty-nine percent of specimens having less than 0.021 inch residual clad showed some degree of failure (49 out of 55).

d. Eighty-six percent of specimens having less than 0.024 inch residual clad showed some degree of failure (60 out of 70).

e. Thirty-seven and one-half percent of specimens having greater than 0.025 inch residual clad showed some degree of failure (three out of eight).

f. Failures occurred indiscriminately with respect to operating temperature.

g. Specimens which split without necking showed very little ductility, and represented the most severe type of failure. Those which necked before rupturing were considered less severe; those which incurred relatively deep necking were classed as failures although not ruptured. Those with faint necking were considered mild failures; they, as well as those showing no degree of failure showed general circumferential extension of the cladding, indicating good ductility.

From these results the following conclusions can be drawn:

1. Tendency toward element failure increases with increasing sharpness (acuity). of notch (striation).

2. Failure tendency is increased with decreasing residual clad thickness.

3. Severity of jacket failure is increased with increasing brittleness (decreasing ductility).

4. Failure resistance increases disproportionately with increased clad thickness.

5. Failure resistance appears to be more dependent on thickness of residual clad below the notch than on notch depth.

6. Temperature variations within the range of 290 - 515°C appear to have negligible effect on incidence or severity of failure.

17 Quarterly Progress Report, Metallurgy Development Operation, Fourth Quarter, 1963, HW-80112, pp. 2.45 ff.

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Notch sharpness or acuity, defined[18] as the ratio of notch depth to the radius at the root of the notch, is recognized as an important factor in determining the concentration of stresses in notched specimens under tension. Even with obtuse notches (130° included angle) all of Weber's V-notch striated specimens either split or necked. The greater the ductility of the material, the less important the notch geometry becomes, since, in ductile materials, the concentrated stresses may be relieved by plastic flow of the metal. At low plastic strains the cohesive strength of the metal is not exceeded and no fracture, but only neck-in of the metal occurs. The fact that zircaloy does not work-harden readily permits continued necking rather than a redistribution of forces to less-worked regions.

As the ductility of a notched material decreases (i.e., brittleness increases) its tendency toward tensile fracture in a plane through the root of the notch increases. Thus, cladding material such as zircaloy, which tends to become embrittled due to hydrogen pickup and irradiation damage during exposure, is apt to show a decrease in notch-rupture strength, and may fail by brittle fracture under the expansive forces of the fuel swelling.

The foregoing discussion applies to specimens having a single notch or groove. A common method of alleviating the concentration of stress at the root of a notch is to deliberately form additional notches adjacent and parallel to the original one. In this way the total stress is distributed among the several notches rather than being concentrated at the root of the one.*

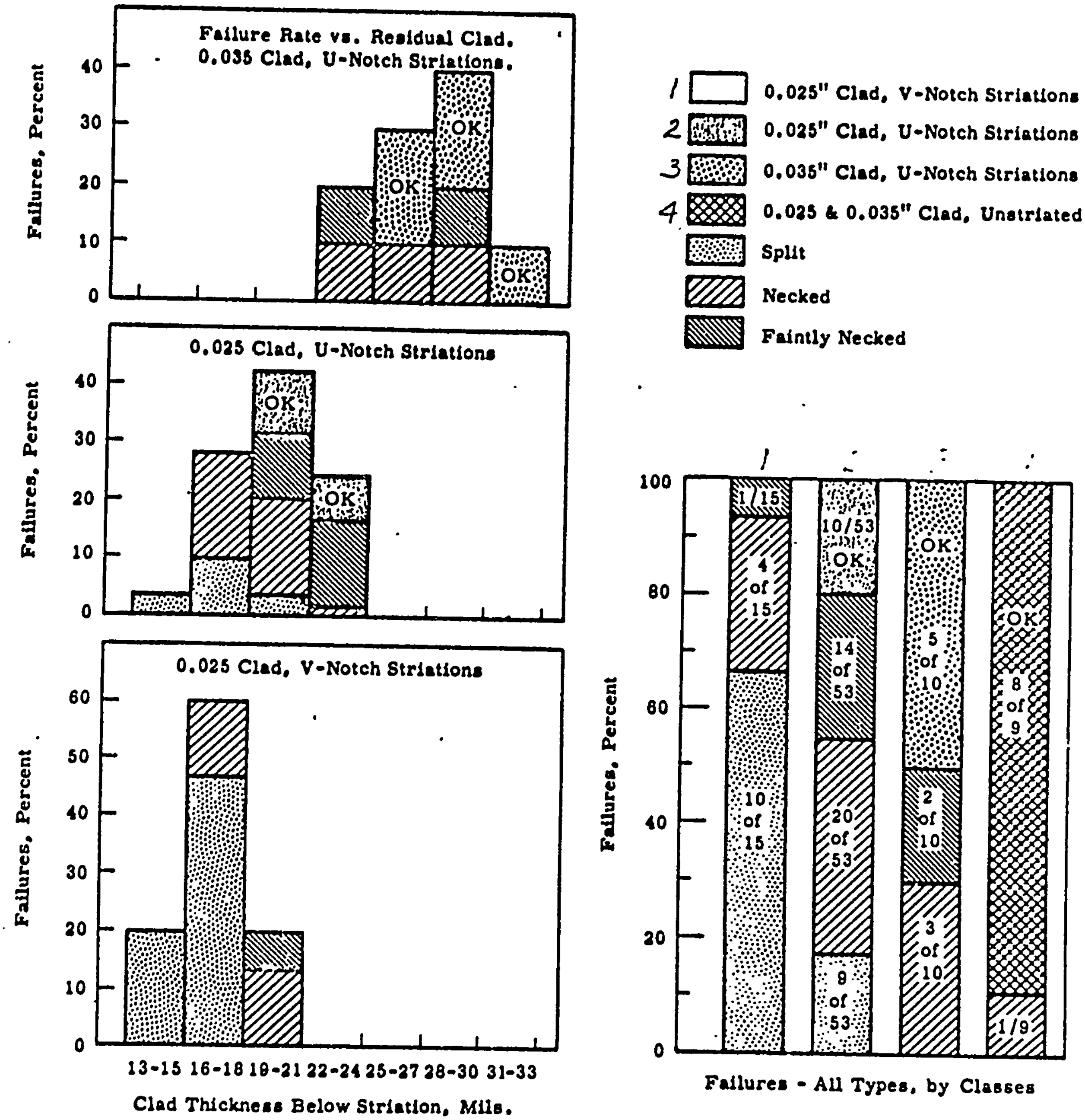
From available evidence, it appears that multiple striations do not have an appreciable stress-concentrating effect, so that the hoop-strength of the jacket is essentially equivalent to the tensile strength of an unstriated sheet having a thickness equivalent to the residual jacket thickness.[11] An exceptionally deep striation would tend to offset the stress-relieving effect of the multiple striations and some stress-concentration would obtain. However, with normal ductility, the clad would stretch and/or neck-in and not rupture under the usual core volume increases up to one percent burnup.

Although increasing the clad thickness increases its overall strength and reduces its tendency to rupture under the influence of forces generated within the uranium core, the increase in rupture resistance is not necessarily proportional to the increase in thickness. About 20 percent of Weber's .025 inch nominal clad U-notch striated specimens having residual clad thickness less than .025 inch showed no failure nor evidence of incipient failure as a result of irradiation, while in the .035 inch clad U-notch striated group, all specimens having a residual clad thickness less than .025 inch suffered some degree of strain damage (necking).

Thus, a decision as to whether it may be possible from the standpoint of the effects of striations, safely to reduce the nominal thickness of the zircaloy jackets depends upon several factors, include the following:

* See Appendix, Item 1.

18 Metals Handbook, 8th Edition, Volume 1, p. 26.
11 OpCit.



<u>FIGURE 1</u>

Distribution of Failures in Zircaloy Clad Fuel Elements Showing Effect of Striations on Tendency to Rupture During Irradiation. Data from Weber, Met. Devel. Op'n. Qr'ly. Rep't., 4th Qr. 1963, HW-80112, pp. 2.45

[redacted]

1. The low thermal conductivity of zircaloy results in an insulating effect where the clad is thick; hence, the uranium tends to be hotter under the thick clad side or spot, and conversely, such irregularity in temperature could lead to warp or bowing of the element with the usual ill effects. The effect of clad thickness variation on element warp has been experimentally verified.[19]

2. Reduction in clad thickness results in reduction of "hoop strength" of the cladding so that it may not be strong enough in the thin region to restrain the expansive forces* generated within the uranium core, and the jacket may split, allowing entry of coolant. Such behavior is favored by embrittlement of the zircaloy due to hydrogen pickup and irradiation damage.

3. Assuming a regular corrosion rate, the thinner spots will be the first to become penetrated due to corrosion. While there is usually an ample thickness of cladding, even in the thin spots, to withstand the normal amount of corrosion, the thinning phenomenon during extrusion is nevertheless an uncontrolled event and could conceivably result in spots too thin to last out the complete irradiation period.

There is, at present, insufficient experience to assess the influence of these factors on continued reactor operation.

APPRAISAL OF THE PROBLEM

While test specimens irradiated under simulated and actual reactor conditions have appeared to be not materially affected by the clad thickness variation discussed above, there is sufficient reason to suspect that adverse effects may sometime appear. It is realized that such thickness variations do occur, and it is the purpose of Inspection to eliminate from production channels all individual elements that fail to meet specified standards. However, the present techniques of inspection are not adequate to effect complete elimination of faulty elements nor to permit an accurate assessment of the problem.

The gradually thinning type of clad thickness variation may be detected and measured by the eddy current tester, but the eddy current method has insufficient resolution to portray accurately the condition of the uranium/zircaloy interface with respect to striations. A visual picture may be obtained by autoradiograph, but this method of testing is slow and expensive. Samples of the jacket may be cut from the stock at the end of each element to permit study of the cladding variation, but such methods also are slow and expensive, and the samples are not necessarily representative of the entire extrusion.

For better evaluation of the problem, and to monitor production, a device is needed to measure accurately and nondestructively the dimensions of the striation - the notch depth and acuity as well as its width and length - at the uranium/zircaloy interface over the entire length of the extrusion. Considering that the desired device must rapidly and nondestructively measure the clad thickness under every unit of surface area throughout the length of extrusion, it is readily seen that the number of potentially applicable methods is extremely limited.

* See mathematical treatment, Appendix, Item 2.

19 Smith, E. A., "Final Report, Warp Problem Studies," RL-NRD-840, (3/2/66).

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From a consideration of the foregoing factors it appears that the greatest hazard in the use of elements of current design lies in the potential embrittlement of the zircaloy jacket, coupled with extended burnup and accompanying swelling.

If the depth and acuity of striations do not become more pronounced in the production output, and if there is no increase in the rate of embrittlement due to irradiation and hydrogen pickup, it appears that existing specifications for wall thickness are more than adequate. Currently, there is relatively little jacket embrittlement as a result of hydrogen pickup and irradiation; there is little corrosion and no apparent trouble arising from such variations in wall thickness as pass inspection. However, it also appears that it would not take an extensive departure from the status quo to cause trouble with jacket rupture.

Uranium swelling appears to be a necessary accompaniment of irradiation. It can be reduced by alloying the uranium stock and can be held in check by external restraint - the combined restraining forces of the cladding and the coolant pressure. In this respect the coolant pressure appears to be a much greater factor than the clad thickness at burnups below and approaching one percent and at temperatures below 300°C.

With a coolant pressure of approximately 1850 psi, a cladding strength of approximately 80,000 psi, and a cladding ductility allowing several percent elongation, it appears that use of cladding not exceeding 0.020 inch in thickness is entirely feasible.* Adoption of such a measure, of course, must be based on a thorough testing program, and must involve the certainty that there will be no increase in the rate of jacket embrittlement due to changes in reactor operating conditions. It must be borne in mind also, that changing the ratio of zircaloy to uranium is apt to call for changes in extrusion tooling and practices, and may require a considerable experimental and evaluation period. Furthermore, it has been suggested[20] that a definite ratio (in practice, about 17/32) should be maintained between the inner and outer clad thicknesses of the "N" inner elements to provide satisfactory restraint against the uranium swelling during irradiation. If this argument is valid, the inner jacket thickness should be reduced in proportion to the outer.

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* See Appendix, Item 5.

20 Dickeman, R. L., (Editor) "Coproduct Demonstration and Implementation Plan," RL-NRD-298, p. 8; A-1 and following.

E. A. Smith

E. A. Smith, Engineer
Special Products Engineering

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APPENDIX

1. The tendency of a body to fracture under tensile load due to exceeding the cohesive strength of the material is measured by determining the stress per unit area under which such fracture occurs. The effect of a discontinuity in the body is to increase the effective stress near to, and in the plane of the discontinuity, and thus to decrease the tensile load necessary to cause fracture. The quantitative effect of such a discontinuity depends on its shape and its location within the body as well as on the shape of the body. For a plate with a small hole near its center, loaded longitudinally with a force S, the stress S_1 at the edge of the hole in a plane perpendicular to the direction of the load is given by the formula $S_1 = S(1 + 2\frac{a}{b})$ (see Figure 2)where a and b are the transverse and longitudinal dimensions, respectively, of the small hole. Thus, for a round hole, a = b, and S_1 becomes 3S; for a crack parallel to the load, $\frac{a}{b}$ approaches zero and S_1 is essentially equal to S, the applied load.

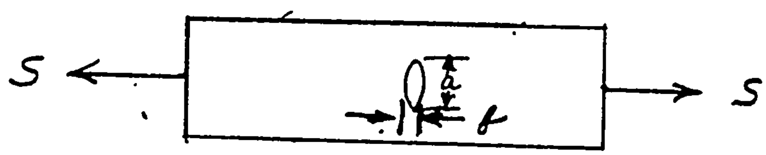

Figure 2

For a transverse crack, $\frac{a}{b}$ approaches infinity and the concentration of stresses at the point of the crack becomes very great. If the material of the body is brittle (i.e., fracures with little deformation), or if the rate of loading is so great that the material exhibits brittle behavior, the concentrated stress can easily exceed the cohesive strength of the material and fracture occurs at and ahead of the point of the crack, which thus is propagated until some barrier is encountered. If the material behaves in a ductile fashion, it elongates ahead of the point of the crack approximately at right angles to the crack, thus relieving the concentrated stresses. A V-notch in the edge of the plate exerts an effect somewhere between that of the crack and that of the round hole. The concentration of stresses at the point depends largely on the sharpness or acuity of the notch, defined as the ratio between notch depth and radius at the root of the V. A radiused or well rounded point on the V-notch favors ductile elongation of the material.

The zircaloy-2 jacket of an N-fuel element normally contains many longitudinal striations in the uranium/zircaloy interface. These striations exist in a variety of sizes and shapes. Between two adjacent striae, the valley may be rounded at the bottom, but often it tends toward a V-shape with only a small radius at the bottom and an acute included angle - the condition which is most donducive to high stress concentration. Although the multiplicity of striations tends to mitigate the concentration of stresses, an exceptionally deep and acute striation conceivably could increase the stress concentration by as much as a factor of ten. Thus, the strength of a zircaloy jacket of normal 80,000 psi UTS would be reduced to 8,000 psi.

2. The "hoop-stress," S_t at any longitudinal section through the wall of a cylinder with internal pressure, is given by the formula:[21]

$$S_t = \frac{PD}{2t}$$

where P = radial internal pressure
D = internal diameter of cylinder
t = wall thickness = .035 inch.

Substituting the value 8,000 psi for S_t, the value of P (the internal pressure to fracture the clad) is found to be approximately 500 psi.

3. The expansive strain of the uranium core is the sum of (a) thermal expansion, (b) irradiation growth, (c) fission gas swelling, (d) cavitational swelling (known locally as "grain boundary tearing"), and (e) expansion due to the development of cracks in the core.

The stresses caused by thermal expansion (and contraction) can be very great — sufficient to overcome the tensile strength of strong material such as the uranium core (resulting in its cracking)— and cannot be restrained by the combined resistive forces of the matrix rigidity, the jacket strength, and the hydrostatic pressure of the coolant, under normal operating conditions. However, the strains imposed by these forces are small and largely reversible; the jacket is ductile enough (unless extremely embrittled) to stretch and accommodate this relatively small swelling. Such swelling is a function of temperature; it stops when the temperature becomes constant, and subsides when the temperature drops. The magnitude of the stress may be derived from Young's modulus[22] to give the formula:

$$S/A = E\alpha\Delta t$$

where S/A = stress in psi
E = Young's modulus (for U = 29,700,000 psi)
α = Thermal expansion coefficient (16×10^{-6} in/in/°C for randomly oriented U)
Δt = Temperature change in °C (approximately 280°)

Thus, the expansive force = 29,700,000 x 16/1,000,000 x 280 = 132,000 psi or 9,000 atmospheres.

But this great force causes only about five mils per inch increase in diameter. The increase in circumference is:

$$\frac{\pi \Delta D}{\pi D} = \frac{.005'' \pi}{\pi \times 1''} = .005 = 1/2\%.$$

The zircaloy jacket normally is readily able to stretch enough to accommodate this strain without rupturing.

21 Boyd, J.E., "Strength of Materials," p. 126 McGraw-Hill, N.Y., 1935.
22 Sears, F.W., and Zemansky, M.W., "University Physics," Addison-Wesley Publishing Co., 1949, p. 256.

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4. Irradiation growth: The distortion due to irradiation growth is an anisotropic phenomenon. Growth in one direction is compensated for by shrinkage in another direction so that there is no net change in density. At the reactor operating temperature, uranium under irradiation behaves like a viscous fluid and undergoes plastic flow.[16] Thus, stresses set up in any direction by crystal growth tend to be relieved by flow of the metal to regions of negative stress, and the jacket-rupture-causing forces contributed by uranium growth remain very low.

5. Fission-gas swelling: The internal forces, generated by the formation of fission products and transmitted triaxially to the jacket, in keeping with Pascal's law, have been treated mathematically by Willis.[23] He states that the fissioning of one mole of uranium yields 1/4 to 1/3 mole of xenon and krypton gases which occupy approximately 36 cc/mole in the disperse phase. At the one percent burnup level, they contribute 1/2 percent to the total volume of the system. Taking 1/3 mole as the yield of stable gas and assuming a one percent burnup, n, the mole fraction of gas generated = 0.003. Willis gives the following formula for calculating the internal pressure:

$$P = \left(\frac{22,400}{V - 36}\right)\left(\frac{T + 460}{460}\right) \times 14.7 = \frac{9150}{(V/36) - 1}\left(\frac{T}{460} + 1\right) \text{ psi}$$

P = gas pressure, psi
T = Temperature, °F
(V - 36) = gas volume (at pressure)*

This will be recognized as an adaptation of the Boyle's law equation of state, PV = nRT. Remembering that this formula pertains to molar quantities, and that one mole of fission-gas requires the fissioning of approximately three moles of U^{235}, one may calculate:

At a density, $\rho = 18.92$, 1 mole of $U^{235} = 235/18.92 = 12.42\ cm^3$. Then three moles = 3 x 12.42 = 37.26 cm^3 = vol. of U to yield one mole of gas. At one percent burnup this volume of U furnishes .005 x 37.26 = 0.1663 cc., or at 100 percent burnup, 16.63 cc. At a temperature of 280°C (536°F) the volume of gas, (V - 36) = 16.630; V = 52.630 cc.

$$P = \left(\frac{9150}{52.630/36 - 1}\right)\left(\frac{996}{460} + 1\right) = 62,819 \text{ psi}$$, or for one percent burnup, approximately 628 psi.

This gives a hoop stress, $S_t = \frac{PD}{2t} = \frac{62,819 \times 1.125}{.07} = 1,011,385$ psi at 100% burnup.

At one percent burnup, this tangential stress on the jacket would amount to 10,114 psi, approaching the stress at which creep begins in zircaloy-2 at that temperature.[6]

6 Lustman and Kerze, Op cit. p. 529.
16 Weber, Op. Cit.
23 Willis, A. H., Reactor Technology Report #8, KAPL-2000-5 (3/59).

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At a higher temperature (350°C) the stress is raised to 10,989 psi which is definitely within the creep range of the zircaloy at that temperature.

If the wall thickness were reduced to .020 inch, this hoop stress at 280°C, one percent burnup, one atmosphere external pressure becomes $P = \frac{62{,}819 \times 1.125}{.04}$ = 17,683 psi. At 280°C the ultimate tensile strength of zircaloy-2 is about 48,000 psi or about 2 1/2 times the stress generated.

6. Cavitational swelling (grain boundary tearing): While the exact mechanism of this type of irradiation-induced distortion is still imperfectly understood it is thought to involve the following factors:[24]

   a. At approximately 400° - 500°C the anisotropic expansive and growth forces of individual grains become greater than the cohesive forces at the grain boundaries.

   b. There is repeated thermal cycling with pronounced temperature gradients.

   c. The opposing forces set up by the cyclic thermal expansion and anisotropic grain growth cause separation of the metal fibers or grains, leaving small vacant cavities between them. This induced porosity reduces the density of the metal and increases its volume. The cavities may serve as pockets for the accumulation of diffusing fission gases and the mass continues to swell as by fission-gas swelling. Cavitational swelling, like the fission-gas type, is diminished by physical restraint, such as the reactor coolant pressure and the containment of the zircaloy cladding.

In N-Reactor operation, the forces opposing the swelling uranium include the coolant pressure of approximately 1,850 psi which cuts the effective internal gas pressure to

$$\begin{array}{r} -\ 1850 \\ +\ \ 628 \\ \hline -\ 1222 \end{array} \text{ psi.}$$

Thus, at one percent burnup (8000 MWD/T exposure), the swelling tendency and therefore the tendency to rupture the jacket due to internal pressure, is more than offset by the external pressure of the coolant, and the net effect is a tendency to collapse the jacket rather than to expand it.

7. Even if there were no opposing coolant pressure, the enormous internal pressure of the forming gases is rapidly relieved by the stretching of the restraining jacket to increase the gas volume. Assume that the internal pressure of 628 psi developed at one percent burnup, 280°C, succeeds in increasing the volume of the system by 0.1 percent. About 0.7 percent of natural uranium is the fissionable U^{235}, so that the volume of natural U containing one mole of U^{235} is (238/.007) ÷ 18.9? (the approximate density) giving 1797 cm^3. This entire volume serves as a sink into which the fission gases diffuse. Assuming free plastic flow and no porosity to serve as "accumulators" for the forming gas, at one percent burnup and 280°C the pressure transmitted to the jacket wall is approximately the entire 628 psi, but as the volume of the system increases,

[24] Leggett, R. D., et al., "Irradiation Behavior of High Purity Uranium," HW-79559.

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the pressure drops. When the volume has increased 0.1 percent, ($.001 \times 1797 = 1.797$ cc), then

$$P = \left(\frac{9150}{54.427/36 - 1}\right)\left(\frac{996}{460} + 1\right) = 56{,}672 \text{ psi, or for one percent BU, } 567 \text{ psi.}$$

Hoop stress $S_t = \frac{567 \times 1.125}{.07} = 9114$ psi, a value well below the yield strength of most specimens of zircaloy.

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