6th International Conference on CANDU Fuel

ABSTRACT

LOOKING BACK AT FUEL THIRTY NINE TO FORTY YEARS AGO

by

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This paper addresses some of the CANDU fuel's behaviour which we did or really did not know back then, in the late fifties and early sixties and how we tackled decisions on the various subjects associated with fuel, both real and imagined.

1.0 What was the maximum rating for UO_2 fuel to be used in a commercial reactor - $\int \lambda d\theta$ 40 to 60 W/cm? What would the resultant fission gas pressure do to the behaviour of the fuel element?

2.0 Was the corrosion behaviour of Zircaloy-2 good enough to last the exposure time required?

3.0 What was the significance of circumferential ridges to the fuel performance?

4.0 Would the bundle be dimensionally stable with .38 mm (.015 inches) thick cladding and strong enough to withstand fueling machine loads to pass through the reactor, thus would the kink-tube test provide enough allowance for future pressure tube sag?

5.0 Would the bearing pads work, as proposed and would braze beryllium alloy, stand the test of time?

6.0 How stable would a defective element be? Drilled holes did not seem to be the answer to the question.

7.0 What were hydride sunburst and blisters?

8.0 Was inter-element fretting life threatening to the fuel?

9.0 Was the quality of resistance welding good enough for the fuel's lifetime?

10.0 What transient high temperatures could the sheath withstand?

These were just some of the questions we faced back then.

A lot of these questions may seem irrelevant today, but back then, they were real and caused many hours or years of soul-searching research, development and irradiation testing to either prove or disprove. These various topics are reviewed and discussed.

I was right in the middle of writing the history of Avro Canada and Orenda Engines, which my publisher had requested, when I received notice that this paper was to be available in June. I thought I had at least until August to produce it! At the same time I was committed for a paper on the history of Gas Turbines in Canada for June for the 5th Air Force Historical Conference. So I had to switch overnight from aeronautical history to Canadian Nuclear Fuel history and dig out my old dusty papers of the early sixties and seventies, some of which I had forgotten I had written.

I am always fascinated when I dig back into my old reports, with the number and diversity of irradiations we did in such a short time span. At the same time, write the reports and papers on them, without wordprocessors! Our problem in those days is we did not realize how little we knew and much we had to learn from doing. Sometimes we made decisions which were wrong.

Sheathing Ductility

A good example of this was early in our development of thin wall fuel, we thought that stronger sheath material should be better. This was in the days of Douglas Point and we used cold drawn sheaths with a fair amount of cold work, tube reducing was not yet fully developed. We started to get some defects in the first charge fuel during fueling. Thanks to John Lipsett who played his magic touch with the Delayed Neutron monitoring system. Ref. 1. This allowed the team to identify the defective channels and sometimes the bundle pairs. Upon examining the fuel in the bays and in the Hot Cells at Chalk River, we found very jagged cracks or splits in the sheaths. Fig.1. On examination and testing of the irradiated material we discovered, that the material had essentially no ductility left. This is graphically illustrated in the Fig. 2. Ref 2. It is simple to look back now and see how obvious it was. But it took a lot of hard work and much puzzling over the results to come up with the solution, i.e., how to increase the circumferential ductility of the fabricated tubing and devise a test to verify the tubing properties. This resulted in a new specification for 'as received tubing'. Fig 3. (Note the change in diametral clearance). This was before the days of the classical stress corrosion cracking (SCC). Fig 4. We had stress corrosion cracking, way back on the major wing bolts on the Dakotas, we were flying in India in 1946/47. Fortunately none of our wings fell off. SCC was probably the initiating failure mechanism of these defects, but that knowledge would come some time later in our understanding of the fuel-sheath behaviour due to an incremental power increase during fueling. We should not forget the debt we owe Mike Notley et al, for saving the Zircaloy tubing industry. He proved that it was residual fluoride in the UO_2 that was the cause of the prototype Dresden fuel failures in the GE Vallecitos VBWR, as a lot of the American industry was thinking seriously of switching to stainless steel, because of the result. Ref.3.

Thin Wall

Though the abstract has fuel rating as one of our concerns, the bigger one initially was, would thin wall sheaths survive the arduous conditions that were required to survive in the reactor? The debate of 0.38 mm(0.015 in) vs. 0.63 mm(0.025 in) thick sheaths went on until we went into production with Douglas Point. We hedged our bets with the loading of NPD[1] by both thick and thin wall. Dr. W. B. Lewis was a great believer in the thinner sheath, as it improved his favorite topic 'Neutron Economy'. I think he would turn over in his grave, if he saw the load of neutron absorbing material, that is now in our reactor cores!

With the various single element and bundle irradiations that were performed in NRX[2] and NRU[3] we became less concerned with the thin wall debate. Ref.4 &5. History has proven us right and I believe we are striving for thinner sheaths yet. Have we used thinner wall sheaths in our reactors?

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[1] Nuclear Power Demonstrator

[2] Nuclear Reactor eXperimental

[3] Nuclear Reactor Universal

Fuel Rating

The other concern we had was, how high could the rating be raised before it became uneconomical and impractical. The experience with UO_2 was limited relative to all the work that had been done on Uranium metal and its alloys. But much to our surprise element ratings up to 60 W/cm and higher were achieved, and in some cases central melting occurred with no major defects. Fig 5. Ref 6.

Fission Gas Release

The effect of fission gas pressure in the longer term from these high ratings was a concern. This was due to the fact that we didn't really know what the normal fission gas release would be during a fuel bundle's normal operating life. We had very scattered results from the early bundle irradiations in NRU, which is illustrated in Fig 6. Ref 2. The dotted line is the compromise we made with Dr. W.B. Lewis for a major paper at a Heavy Water Power Reactors Conference in Vienna. Mike Notley and Archie Robertson et al, did yeoman work in this field and slowly were able to come up with proper models and predictions. But before this we put a lot of unnecessary work and thought in how to prevent the fission gas pressure from exceeding the coolant pressure and cause fuel element diametral creep. This I assume is still a concern for high burnup bundles for future fuel cycles, though the irradiation hardening of the sheaths does reduce some of the concern. I understand that some bundles have gone to a very high burnup with and without defects.

Zircaloy Corrosion

A subject that took a lot of attention world wide when Zircaloy became widely used in Power Reactors was the corrosion behaviour of the material in the hot reactor coolant. It was another nonissue after many years of worry. Once the material was made consistently and of high quality no real corrosion problems were observed. We spent a lot of time and money trying to predict corrosion rates and hydrogen pickup. All the fuel bundles were originally autoclaved before use in the reactors. We found that the reactor coolant could do a better job of producing a good oxide layer without adding the extra hydrogen pickup before service, with good coolant chemistry control. Out of reactor tests were not very helpful as the effect of neutron irradiation was important in understanding the behaviour of the oxide and hydrogen pickup and in our case deuterium pickup. Fig 7. (Ref 2) The Russians were very upset with me in particular when we published that we were dropping autoclaving. They read to me chapter and verse, all the papers I had presented, during one of their Cold War visits! It should be mentioned that we left 13 fuel bundles in NPD during its whole operating life and they survived.

Circumferential Ridges

We were initially fascinated by the circumferential ridging as seen on reasonably rated bundles and elements. We called it the 'Bamboo effect' much to the disgust of Archie Robertson. He was very critical of my engineers for coming up with colloquial terms to graphically describe what we saw. Such as 'snowball corrosion'[4], 'banana effect'[5], 'racoon-tail'[6] Canadian or tiger-tail corrosion UK one, 'split-warts which became the split spacer', 'dryout', and so the list went on. This is where the cartoons came from for our Fuel Engineering Coat of Arms. The alternating bands of corrosion, the effect of grooves in the pellets and the cracking pattern of the pellets producing marked corrosion difference, have never to my knowledge been properly explained in detail. Fig 8. I often wonder if the ridges actually strengthened the sheathing, as corrugations of metal plates produce an increase in rigidity strength. It was used in the earlier Junkers Tri-motor aircraft and in SR-71 aircraft Black Bird program and corrugated roofing. The ridges as far as my knowledge goes, have never been a major performance problem.

Bundle Strength

As our bundles had to be fueling machine loaded and moved under power, we had to prove to everybody's satisfaction that the bundles were strong enough. We did out of reactor tests and on irradiated bundles. What we found was that the bundle was very strong, as long as it was inside the pressure tube. After irradiation the bundles were so strong that the strength test machine was found not to have enough force to obtain a significant deflection.

Kink Tube

Back then, the kink tube test was our best guess of what the bundles may see many years in the future, due to pressure tube sag. At the same time it was a compromise, not to restrict the maximum allowable bundle diameter. I assume it is still in place today? I understand this is a current topic of discussion.

Bearing Pads

Originally the NPD fuel bundles had no bearing pads per say, but slid on the wire-wrap. For Douglas Point an extra thick wire was added to the outside elements and machined with a flat surface. Because of the concern with inter-element fretting of the wire-wrap, we developed the split-spacer fuel with designed bearing pads, even though we never had a normal inter-element fretting defect. A large number of various designs of pads were investigated and tested, including graphite and roller bearings on the flexible wire spacer bundles. The only candidate which was successful was the design used today. Fig 9. The control of the wear rate was found to be most dependent on the width of the pad frontal area. It did not seem to make much difference whether the whole pad was in contact with the pressure tube or not, but there was some relationship to contact area.

Defects

The Americans (Bettis) were very keen on drilled hole defect testing. Unfortunately, they used rather large drill holes in their sheaths. This allowed the water and steam to escape easily to the coolant and produced minimal damage to the sheath. However, when the defect is very small,

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[4] The increasing oxide thickness insulates the Zircaloy and produces higher temperatures, with resultant corrosion.

[5] The description of a bowed element.

[6] The alternating bands of dark and white Zirc oxide

like a pinhole or a small crack, then the water penetrates the inside of the fuel sheath, most likely during shutdowns or a thermal cycle. The resultant steam produced corrosion of the inner surface and the resultant hydride entered the sheath and migrated to the cooler parts of the sheath. If the fuel was left in the reactor for a significant length of time after a defect had occurred, then the hydriding can becomes very severe, producing 'sunbursts' on the inside and on the outside 'blisters' due to the volume change of the hydride. Fig 10. Ref.8 . If the defect is very small, it becomes plugged by both corrosion and fission products, then the mechanism for deterioration is accelerated. This is why we always recommended that the defective bundles should be removed from the reactor within 24 to 48 hours. The sooner the better, if you want to keep a clean reactor and coolant, to minimize your maintenance exposures.

Resistance Welds

The resistance welded endcaps have worked extremely well, but the secret to them was good quality control of the welding, to minimize the weld line and good housekeeping cleanliness. Roman Sejnoha and others are far better qualified to speak on this subject then I am. My major experience on this subject, was with the initial fuel charge for the Bruce A reactors. There was some concern that a number of bundles had suspect welds but the exact bundles in the batch were never identified. I had the whole batch quarantined. A number of years later Hydro decided to use them without my knowledge. They quickly found out why they had been quarantined. The bundles with defective welds soon became defective when they were loaded in the reactor!

Fuel Sheath Transient Temperatures

This subject has always been a favourite of mine, as it produced the most excitement in the programme. When you are investigating the unknown with rather dire consequences of serious fuel damage it gets the team's attention, and senior management a bit nervous. The programme started with the conversion of the X-4 Loop in NRX reactor to fog (steam/water mixtures), boiling and superheated steam cooling. Ref. It was part the Canadian/UKAEA cooperative programme when they (the UK) were building and developing SGHWR (Steam Generating Heavy Water Reactor). Canada pioneered the first in-reactor heat transfer tests with Zircaloy fuel elements, in the early 60s. These tests produced the Canadian definition of 'Dryout heat flux', where the sheath is no longer cooled by a film of water but by the velocity of steam in the coolant.

A series of experiments was undertaken to see how long the fuel would last at various temperatures above normal. This resulted in the classical graph of time to defect vs. sheath temperature. Fig.11 Ref 7 & 8. The single element tests and pump rundown experiments led us to investigate the behaviour of full scale fuel bundles in the U-1 Loop in NRU reactor culminating in U-111 tests.

For some reason our senior management forbade us to do any more in reactor heat transfer tests after those major experiments. Fuel Engineering Branch was instructed not to concern ourselves about Critical Heat Flux, as this was reserved for Safety Branch . This did not make sense to us then, as our bundle designs were there to produce heat to the coolant! The programme suffered from this as they had to pour millions of dollars into electrically heated bundle tests. This was not corrected until many decades later when Alan Lane was able to bring Fuel Engineering and Heat Transfer together which resulted in the significant advancement of the

CANFLEX bundle.

The majority of these subjects and more are described and discussed in Ref. 9.

The next slide has nothing to do with the presentation but it will get your undivided attention. It will not be published in the proceedings and the editor's decision on this is final!

Thank you.

REFERENCES

1. KEE, F.J., LIPSETT, J.J., "CANDU Operating Experience with a Delayed Neutron Failed Fuel System" IAEA, Vienna, 1970, AECL 3628.

2. PAGE, R.D. "Engineering and Performance of Canada's UO_2 Fuel Assemblies for Heavy-Water Power Reactors," IAEA Heavy Water Power Reactor Symposium, Vienna, Sept. 1968, AECL-2949.

3. NOTLEY, M.J.F., ROBERTSON, J.A.L. "Zircaloy-UO_2 Failures Tied to Fluorides," Nucleonics, March 1961, Vol.19, No.3, Pgs. 77-79, AECL-1219.

4. MOORADIAN, A.J., "Economic Fuel for CANDU," Sixth World Power Conference, Melborne, Australia, 20-27 October, 1962.

5. CHALDER, G.H. "Problems in the Use of Thin Zircaloy Cladding for Uranium Dioxide Fuel," Proc. Symp. Fuel Element Fabrication, Vienna 1960; 1, 57-58, IAEA, 1961, AECL-1145.

6. NOTLEY, M.J.F., LANE, A.D., "Factors Affecting The Design of Rodded UO_2 Fuel Bundles for High Output" Presented by Page, R.D. IAEA Heavy Water Power Reactor Symposium, Vienna, Sept. 1968, AECL- 2945.

6. PAGE, R.D., et alii, "Engineering and Performance of UO_2 Fuel Assemblies", Third U.N. International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1964. AECL 2018

7. PAGE, R.D., LANE, A.D., "The Performance of Zirconium Alloy Clad UO_2 Fuel for Canadian Pressurized and Boiling Water Power Reactors." ANS-CNA Conference, Toronto, June, 1968. AECL 3068.

8.COLLIER, J.G. et alii, "The First Experimental Irradiation of Fog Cooled Fuel," AECL-1819, July 1963.

9. PAGE, R.D., "Canadian Power Reactor Fuel", March 1976, AECL 5609.

Paper 125 Figure Captions

Fig 1. Early Douglas Point Fuel Sheath Failure showing jagged straight spilt in sheath.

Fig 2. Zircaloy sheath Total Circumferential Elongation % at 300*C vs Axial Ultimate Tensile Stress MN/m^2 at room temperature.

Fig 3. Comparison of tubing specifications and diametral clearance.

Fig 4. Examples of Zircaloy Sheath Stress Corrosion Cracks In & Out Reactor.

Fig 5. Longitudinal Cross Section of a BLW Fuel Element at high heat ratings. (Note the pellet grooves, cracks and the effect of the end cap on the grain growth).

Fig 6. Fission Gas Release for Prototype Power Reactor Fuel with Different Power Ratings ($\int \lambda d\theta$ W/cm).

Fig. 7. Early results of NPD Hydrogen and Deuterium pick-up vs Hot Coolant days, for different types of Zircaloy-2 Tubing.

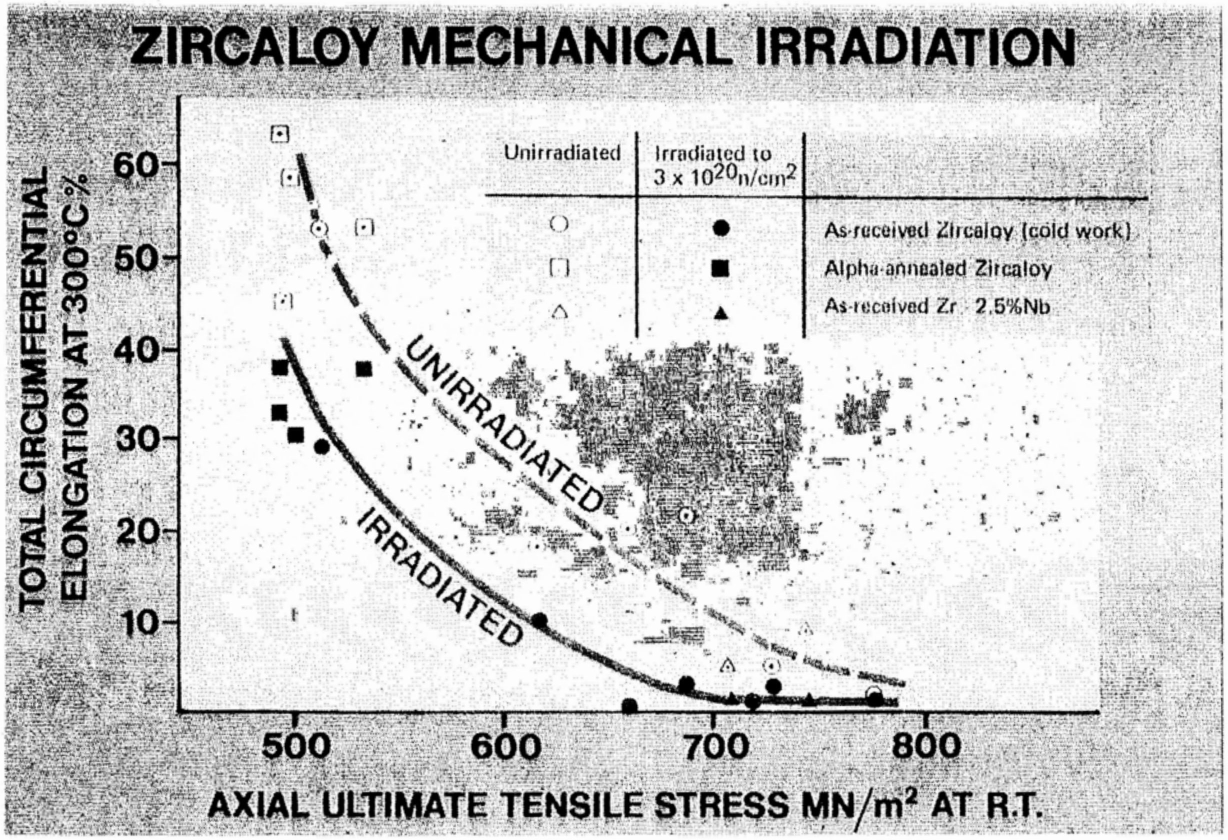
Fig 8. An Example of a BLW Bundle displaying marked corrosion patterns outlying circumferential ridges, pellet cracks, axial gap and circumferential grooves in the pellets.

Fig 9. Close of the brazed bearing pads and skewed-split-spacers.

Fig 10. Examples of Hydride Blisters (Bumps) in the sheath and Sunbursts, showing the hydrogen or Deuterium diffusion to lower temperatures.

Fig 11. Sheath Temperatures in Dryout vs Exposure Times at Dryout Temperature to Defect.

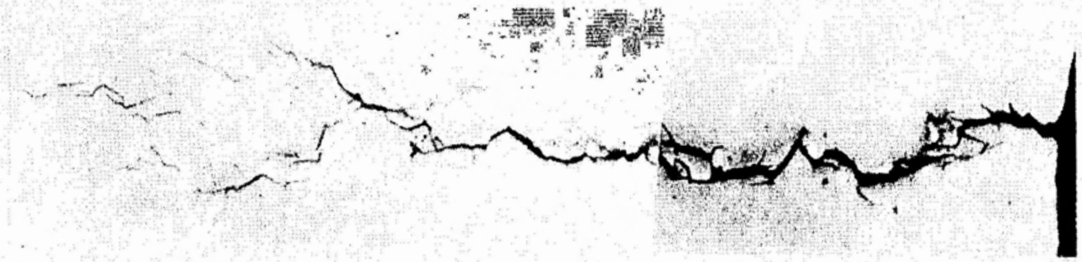

FUEL BUNDLE


ZIRCALOY MECHANICAL IRRADIATION
TOTAL CIRCUMFERENTIAL ELONGATION AT 300°C %
60
50
40
30
20
10
Unirradiated
Irradiated to 3 x 10^20 n/cm^2
As-received Zircaloy (cold work)
Alpha-annealed Zircaloy
As-received Zr - 2.5%Nb
UNIRRADIATED
IRRADIATED
500
600
700
800
AXIAL ULTIMATE TENSILE STRESS MN/m² AT R.T.

TUBING SPECIFICATIONS

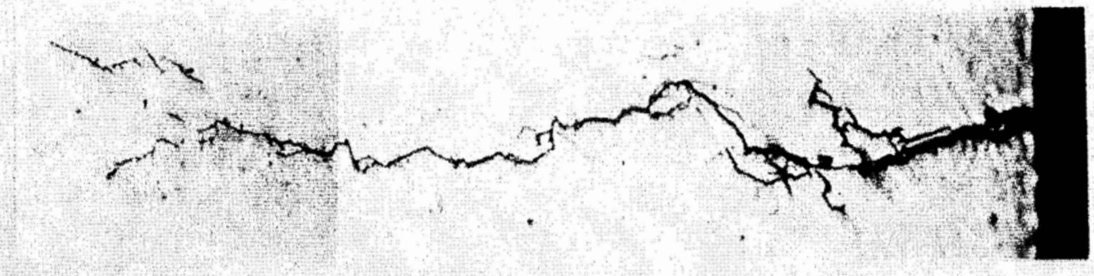
MATL. PROPERTIES AT R.T.	DOUGLAS POINT	PICKERING	NEW SPECIFICATIO
MINIMUM U.T.S. kpsi	75	75	73
MINIMUM 0·2% Y.S. kpsi	65	60	57
MINIMUM TOTAL CIRCULAR ELONG^{N}.%	-	15	18·5
DIAMETRAL CLEARANCE	·0015"-·005"	·0015"-·005"	·0035"-·005"

STRESS CORROSION CRACKS



Out-Reactor Iodine Test

100μm



In-Reactor Crack

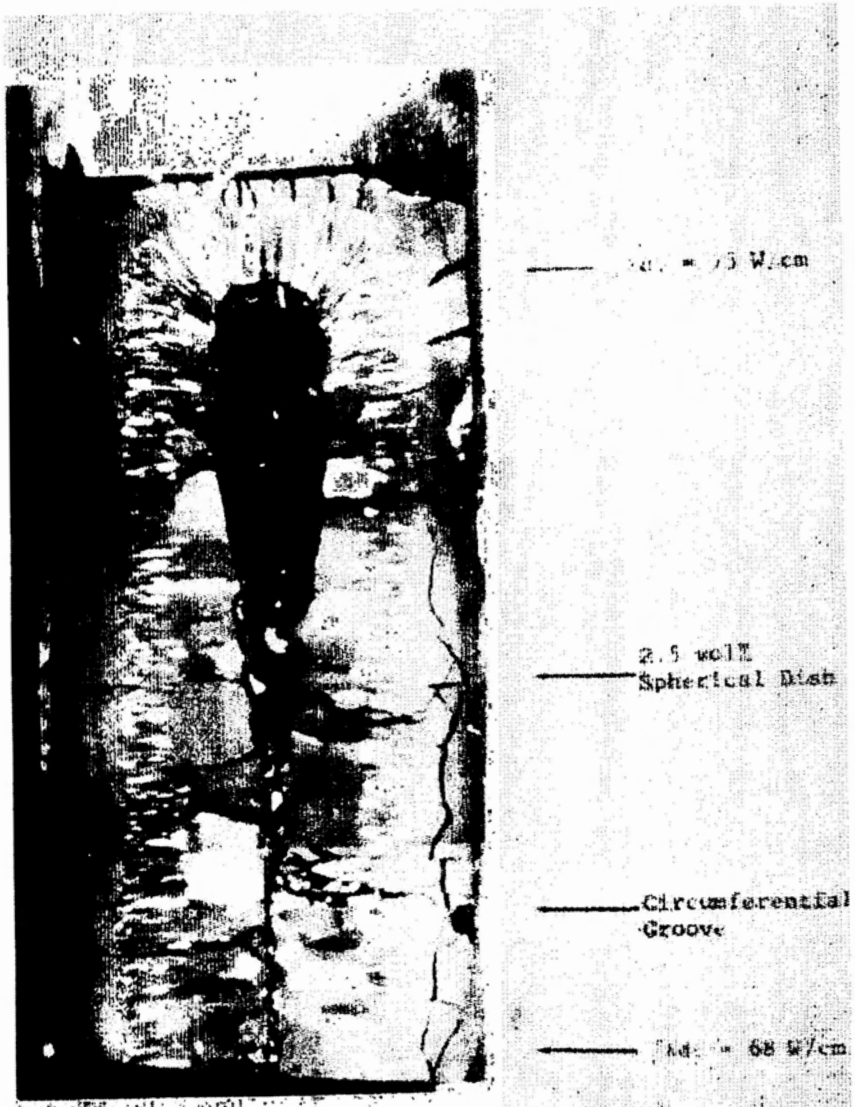


FIGURE 7

Longitudinal Cross Section of [illegible] Fuel Element at High Heat Rating

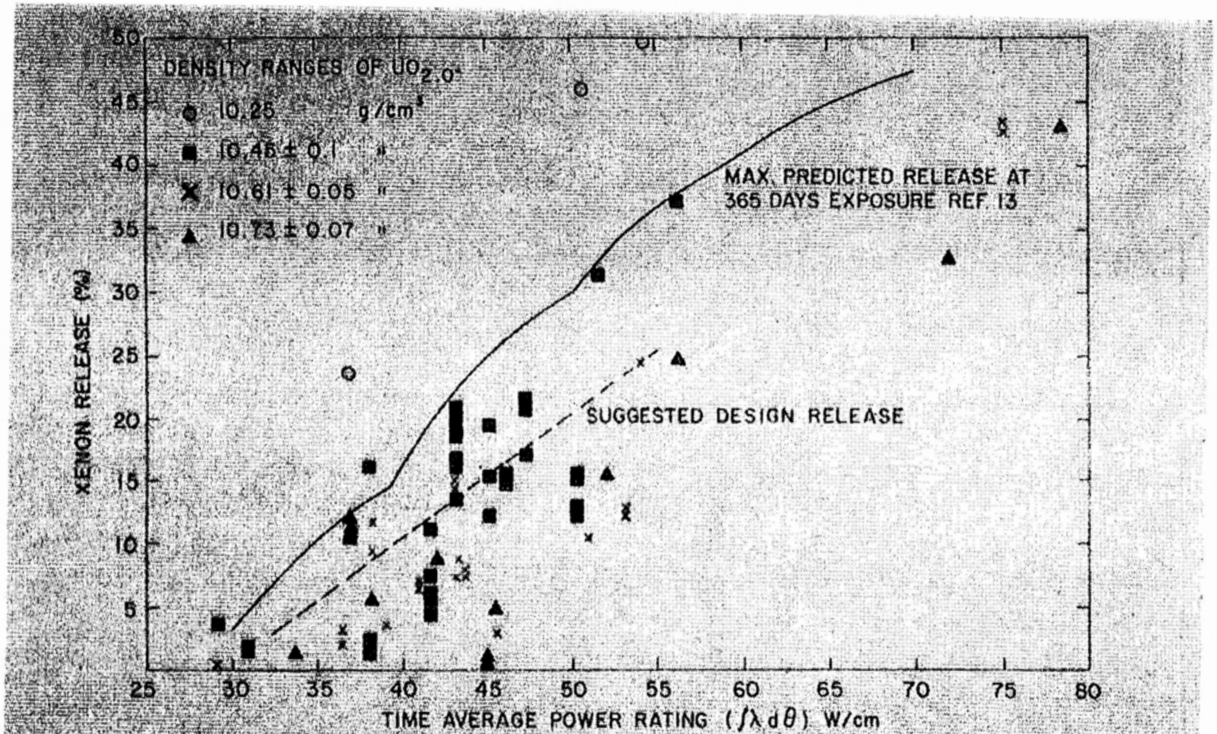


FIG 3 FISSION GAS RELEASE FOR PROTOTYPE POWER REACTOR FUEL WITH DIFFERENT POWER RATINGS ($\int \lambda \, d\theta$)

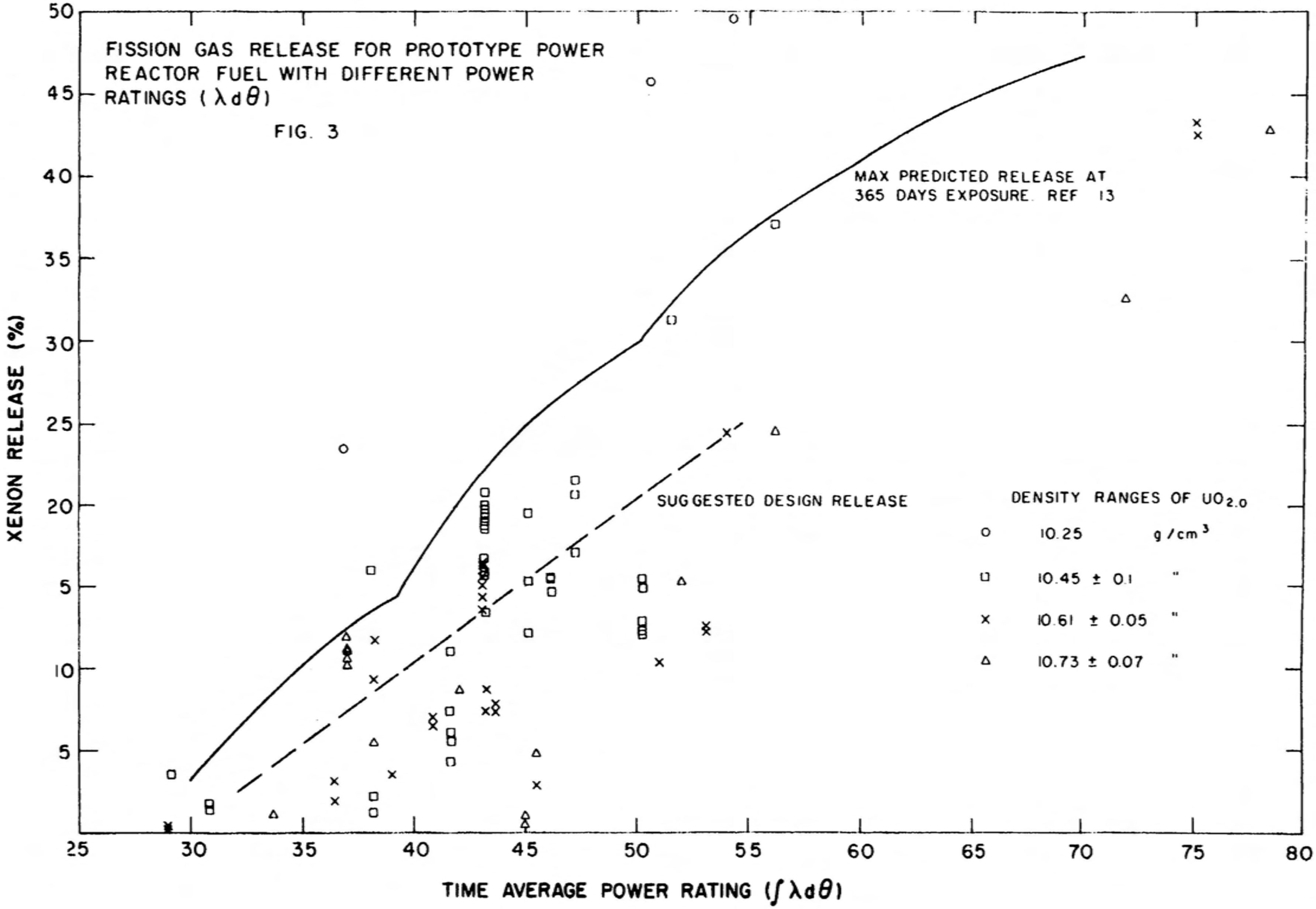
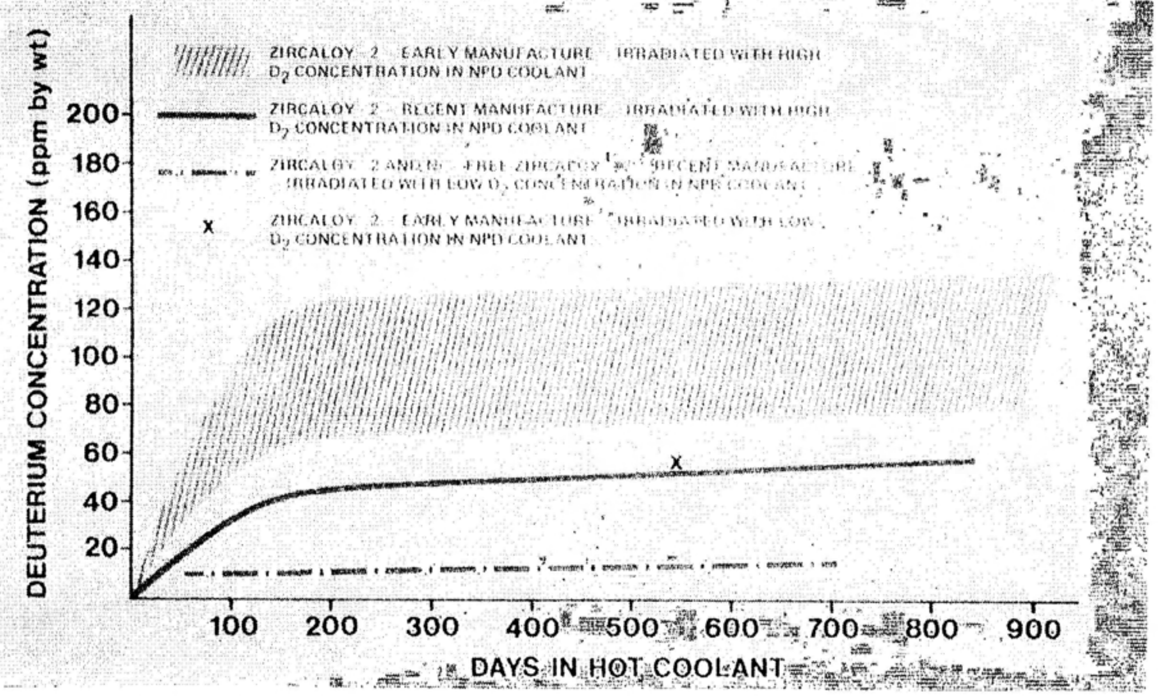
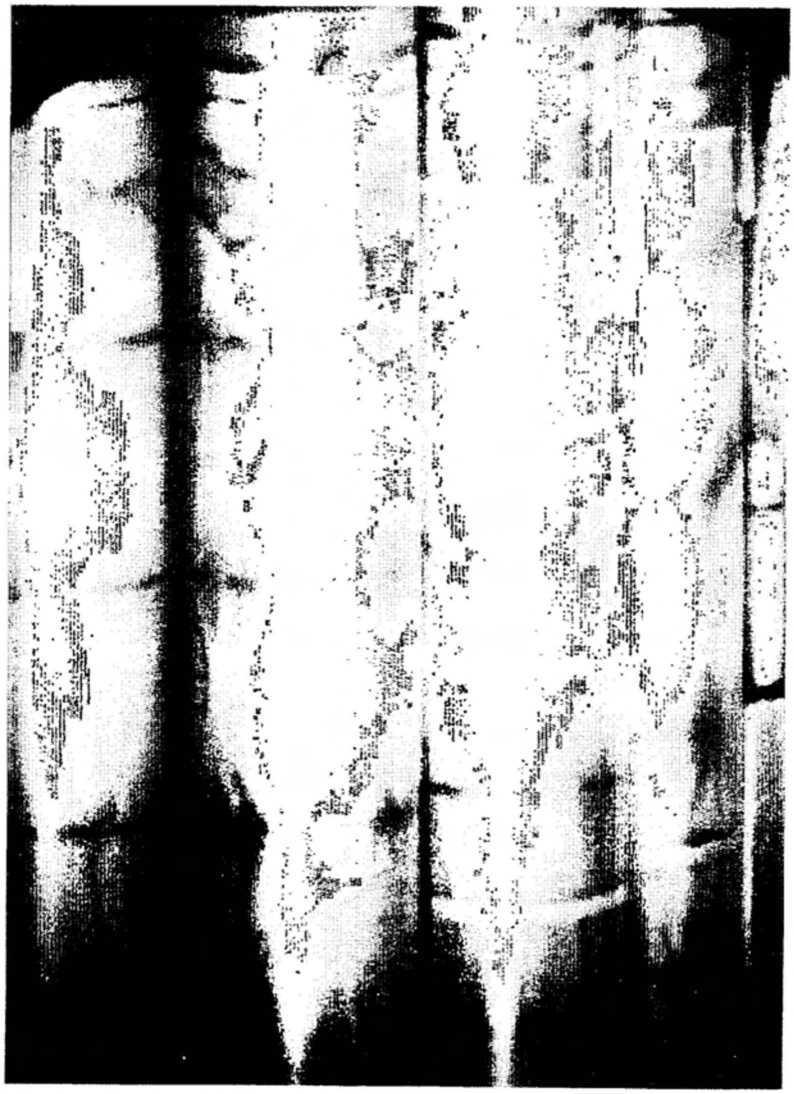
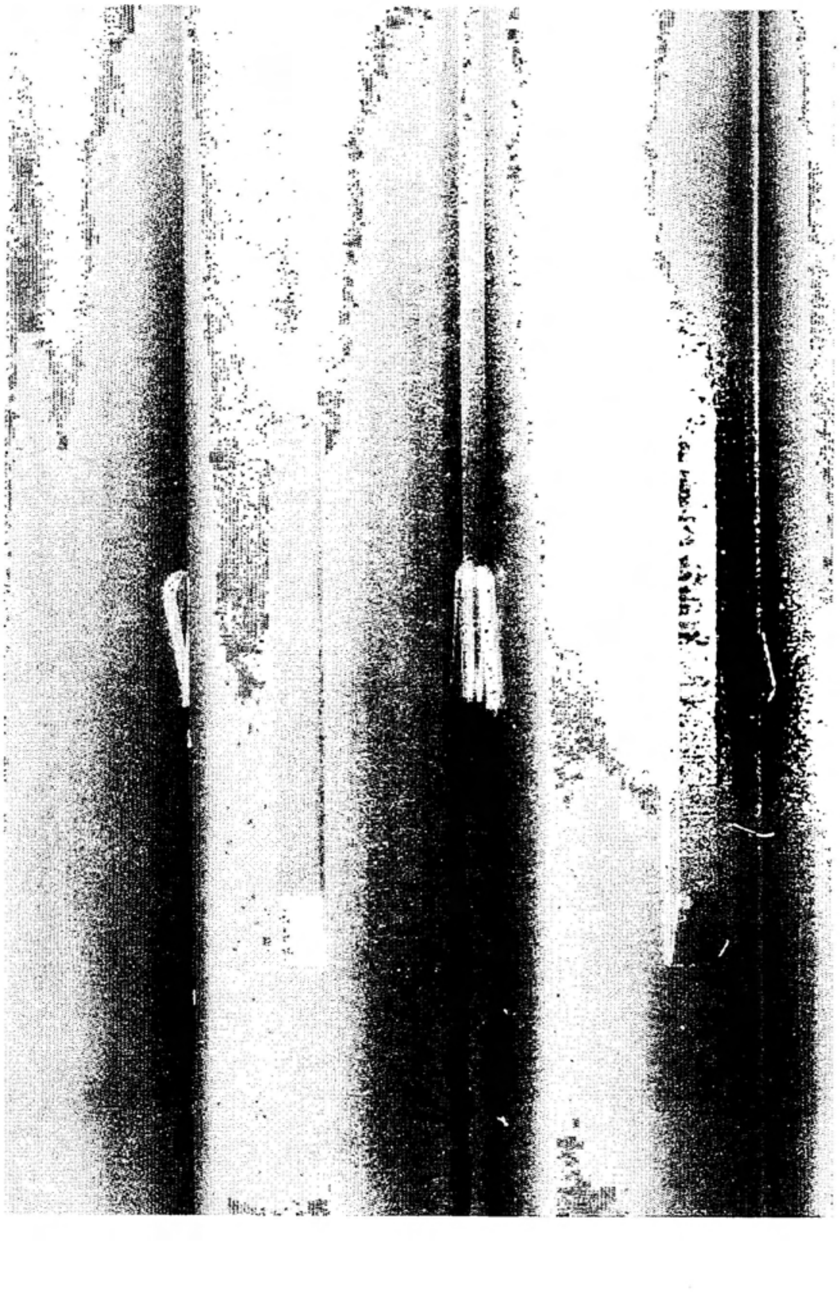

FISSION GAS RELEASE FOR PROTOTYPE POWER
REACTOR FUEL WITH DIFFERENT POWER
RATINGS ($\lambda d\theta$)
XENON RELEASE (%)
TIME AVERAGE POWER RATING ($\int \lambda d\theta$)
MAX PREDICTED RELEASE AT
365 DAYS EXPOSURE. REF 13
SUGGESTED DESIGN RELEASE
DENSITY RANGES OF $UO_{2.0}$
○ 10.25 g/cm^3
□ 10.45 ± 0.1 "
× 10.61 ± 0.05 "
△ 10.73 ± 0.07 "

FIG. 3

HYDROGEN AND DEUTERIUM PICK-UP

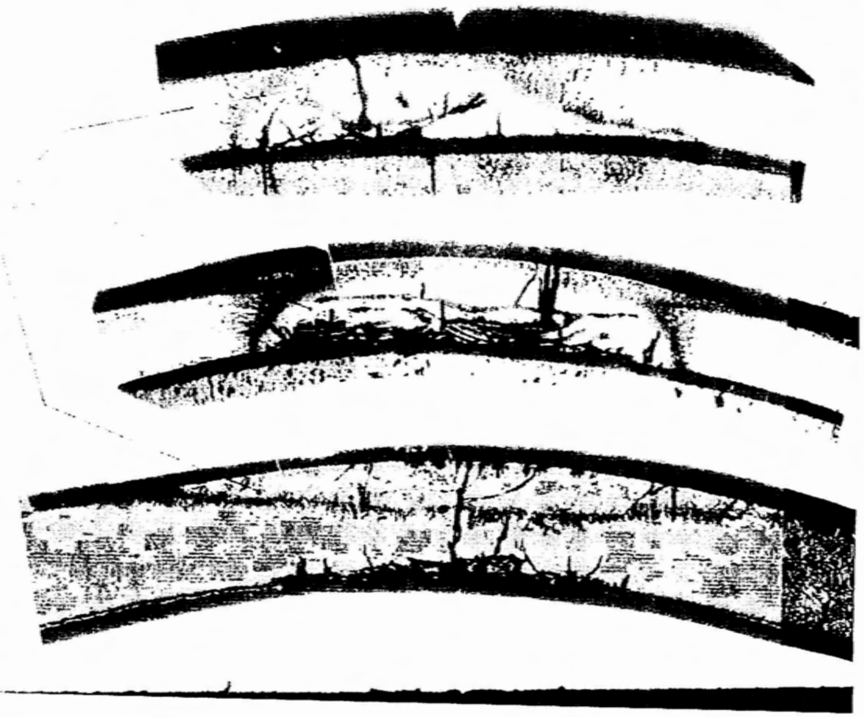






22 SMALL DEFECT OF U-210 PHASE I
SHOWING SMALL AMOUNT OF HYDROGEN
DIFFUSION FROM BUMP.

R16 H4 TO 7



23 BUMP OBSERVED IN SHEATH OF TEST U-106
PHASE II SHOWING MORE HYDROGEN DIFFUSION
TO OUTER SURFACE OF THE SHEATH

R [illegible]

24 HYDRIDE BUMP OBSERVED IN U-106 PHASE IV
SHOWING SOLID HYDRIDE LAYER ON OUTER
SURFACE OF SHEATH, AND MUCH LOWER
HYDROGEN CONCENTRATION ON INNER SURFACE

R24 H5 TO 8

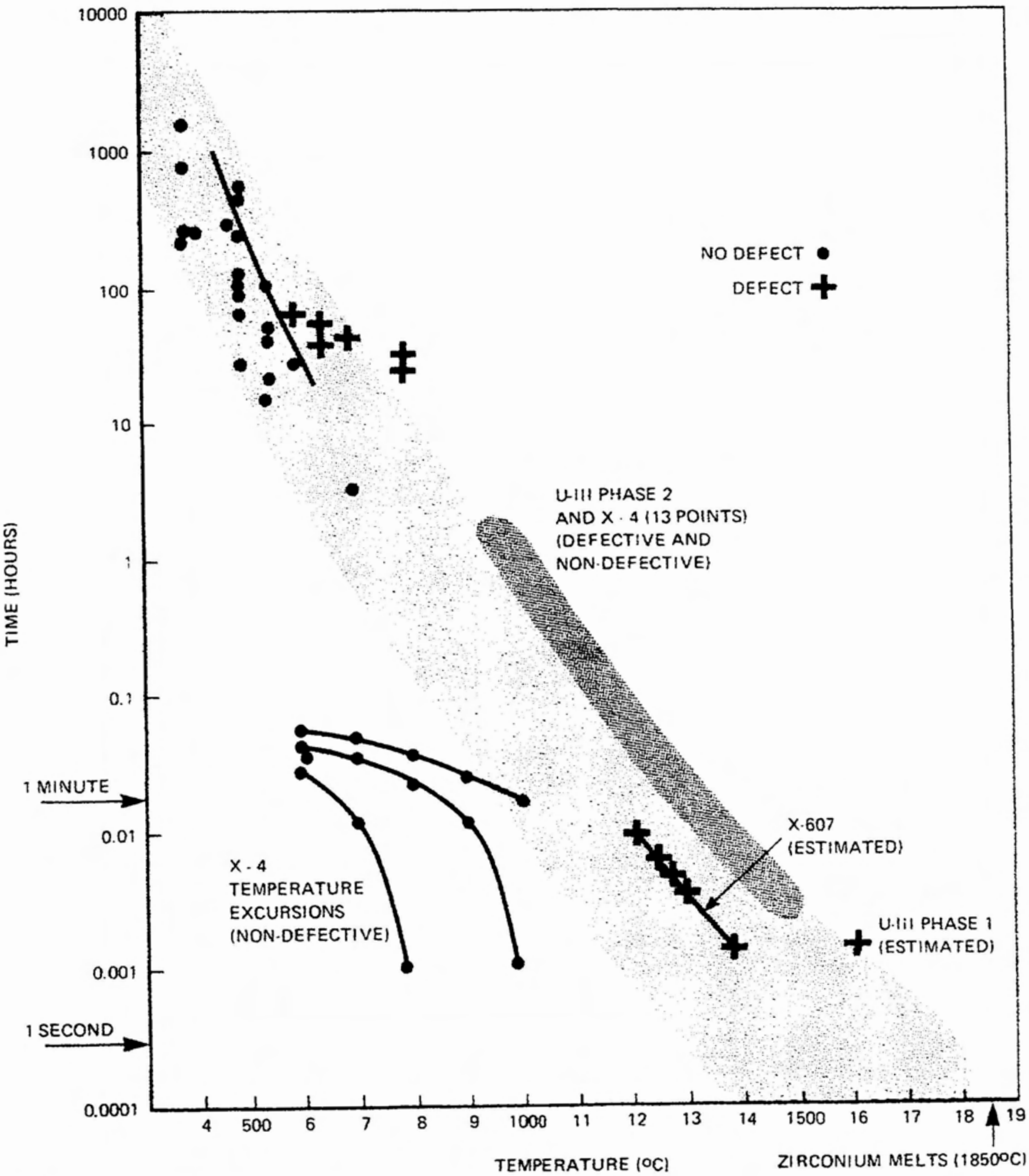


FIGURE 39 *Sheath Temperature versus Time to Defect*

FIGURE 40 *High Temperature Corrosion Failure of Fuel Element*