Atomic Energy of Canada Limited

THE PERFORMANCE OF ZIRCONIUM ALLOY CLAD UO_2 FUEL FOR CANADIAN PRESSURIZED AND BOILING WATER POWER REACTORS

by

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SYNOPSIS

Canada's power reactor fuel industry has until recently been based on the production of fuel for the CANDU-PHW reactor concept. This paper presents the most recent information on the performance of this type of fuel, and goes on to summarize the results of an extensive irradiation program to develop fuels for boiling water cooled reactors. In particular, data are presented on the behaviour of the reference fuel for the CANDU-BLW-250 reactor, and the many single element irradiations undertaken in the X-4 and X-6 loops in NRX.

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<u>Performance du combustible UO_2 gainé d'alliage de zirconium destiné aux réacteurs refroidis par eau lourde pressurisée ou par eau légère bouil-</u>lante des centrales canadiennes

par R.D. Page et A.D. Lane

Résumé - L'industrie canadienne des combustibles nucléaires s'est surtout préoccupé, jusqu'à présent, de fabriquer le combustible des réacteurs de type CANDU-PHW (refroidis par eau lourde pressurisée). Les auteurs du présent rapport fournissent des renseignements extrêmement récents en ce qui concerne la performance de ce type de combustible et ils donnent un aperçu des résultats obtenus lors d'un important programme d'irradiation visant au développement de combustibles pour les réacteurs dont le caloporteur sera de l'eau légère bouillante. Les données fournies concernent, en particulier, le comportement d'un prototype de combustible destiné au réacteur CANDU-BLW-250 et les nombreuses irradiations de simples éléments entreprises dans les boucles X-4 et X-6 du NRX.

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1. INTRODUCTION

Canada's power reactor fuel industry has, until recently, been based on the production of fuel for the CANDU-PHW[1] reactor. Two reactors of this type are now operating, NPD[2] [1], [2] since 1963, and Douglas Point [3] , [4] since 1967. Seven more reactors of this type are now under construction, two at Rajasthan, India [13] , another at Karachi, Pakistan [13], and four more at Pickering, Ontario. The four Pickering reactors require a 10 cm diameter fuel bundle instead of the 8 cm bundles used in all of the others [5] , [6] .

A second type of reactor, called a CANDU-BLW[3], is now being developed and built in Canada which requires a different design of 10 cm diameter bundle. This is a vertically oriented reactor which uses boiling light water as its coolant. The first reactor of this type, a 250 MWe station, is being built at Gentilly near Trois Rivières, Quebec, and is expected to be in operation during 1971 [7] , [8] .

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1. CANada Deuterium Natural Uranium Reactor - Pressurized Heavy Water
2. Nuclear Power Demonstration
3. CANada Deuterium Natural Uranium Reactor - Boiling Light Water

This paper presents the latest information on the performance of the CANDU-PHW fuels, and summarizes the results of an extensive irradiation program to develop fuels for CANDU-BLW reactors.

2. PRESSURIZED HEAVY WATER FUEL

2.1 Design and Development

The design and development of fuel for the CANDU-PHW type reactors have been well documented [9], [10], [11], [12], [13]; therefore, it is only necessary to outline briefly the salient points.

The original fuel charge for NPD contained wire wrapped 7-element bundles in the outer zone and 19-element wire wrap bundles in the centre. Element closures and bundle assemblies were done by Tungsten inert gas welding. The 7-element bundle has not been developed further and is being phased out of the reactor. The 19-element bundle design was modified for Douglas Point and resistance welding [14] was used for both the end cap to element closure and bundle assembly. Because of the concern of possible sheath fretting by the wire wrap which spaces the elements apart, the replacement fuel for this reactor utilizes a brazed skewed split spacer design [11], [15]. A close-up photograph is included as Figure 1. This fuel is in production at Canadian Westinghouse Company, Port Hope.

The fuel for the Pickering reactors uses the same length and diameter of element (50 cm and 1.53 cm) and method of fabrication, but the number of elements has been increased to 28 to fill the 10 cm diameter pressure tube. These bundles are shown in Figure 2.

2.2 Fuel Costs

The total price of fabricated fuel for Canada's power reactors has decreased significantly over the last eight years from $121/kg U for the original charge for NPD to the averaged price of $44/kg U (Canadian 1967 $) for 19,000 bundles for the first four Pickering charges. This decrease in price is illustrated in Figure 3, which plots the cost of various fuel charges. Also shown is the suggested worth of the spent fuel from these reactors due to contained plutonium. If this price is realistic, then we may have negative fuelling costs during the early part of the 1970's and very low fuelling costs thereafter, viz: less than 0.5 mills/kWh [16]. Canada's fuel costs will probably decrease still further as the volume of UO_2 required per year increases. This is illustrated in Figure 4 - a plot of UO_2 throughput in tonnes per year - which indicates that after 1971 there is going to be a continuing demand, rising to a thousand tonnes per year by 1980.

2.3 Irradiation Performance

The latest performance data of CANDU-PHW fuel, irradiated in NPD, Douglas Point and the U-2 loop [17] are summarized in Table 1.

Approximately 6,000 bundles have been irradiated in NPD and Douglas Point up to heat ratings $\int\lambda d\theta$ of 43 W/cm. Of the 2,157 bundles irradiated in NPD, 34 have been taken to burnups in excess of design 6,500 MWd/TeU. Eight of these have been taken to burnups greater than 10,000 MWd/TeU and one enriched bundle to 20,000 MWd/TeU. Some of this fuel has now been in NPD for about 1,500 coolant days and seen 132 power cycles. All bundles have achieved this level of performance without significant deterioration [18]. The corrosion and hydriding of the zirconium alloy sheath is discussed in detail in a companion paper by A. S. Bain and J. E. LeSurf [19] and therefore will not be repeated here. Dimensional changes of bundles tested in both NPD and the U-2 loop have been small, less than 0.1% length and 1.3% diameter increases up to $\int\lambda d\theta$ of 50 W/cm.

The Douglas Point fuel charge (containing 12 bundles of the brazed split spacer type) has been in reactor for 100 coolant days and obtained a maximum burnup of 2,400 MWd/TeU at maximum rating of $\int\lambda d\theta = 43$ W/cm. The fuel is operating well.

The 28-element fuel for Pickering reactor has reached an outer element burnup of 9,150 MWd/TeU in the U-2 loop at CRNL at an average rating of $\int\lambda d\theta$ = 47 W/cm. The design satisfies the reactor requirements and the fuel is in production at Canadian General Electric Company, Peterborough.

2.4 Fuel Defects

The fuel defects in NPD can be classified into three groups. The first results from mechanical damage. Three bundles have been damaged during loading and back refuelling into the reactor. The end plug was gouged and the end plug to sheath weld was punctured. Some of these defected bundles were left in the reactor for three years without marked deterioration of the fuel element assembly [20]. One bundle was structurally damaged (end plate welds broken) and was left at the zero flux position in the end fitting for two years before removal. Some of the element sheaths had been fretted through due to their loose arrangement.

The second group consists of purposely defected fuel [21], [22]. Two bundles have been irradiated for two months with one element punctured in each bundle at heat ratings up to $\int\lambda d\theta$ = 40 W/cm without significant deterioration of the fuel.

The last group consists of natural defects and so far only one bundle has fallen into this category. The defect consisted of a crack in the sheath of one element and was left in the reactor for one month after the onset of the defect signal. Examination of this element is still in progress.

The defects encountered in the experimental bundle irradiation program at CRNL are also classified into three groups.

The first again results from mechanical damage caused by the U-2 loop fuel carriage: only a simple redesign was required to cure it.

The second group of defects was caused by gross overpower of two bundles due to local flux distortions. This suggests that the Pickering fuel design has an overpower capability of 180%, but failure of the sheath may result if central melting of the UO_2 occurs. A cross-section of one of the defective fuel elements is shown in Figure 5. The diametral strain at the failures ranged from 1.5 - 2.5%.

The third group has been found to be due to internal contamination of the fuel elements by moisture. The excess hydrogen available from this residual moisture in the pellets has caused local hydride blisters to form in the sheath. Cross-sections of typical blisters are shown in Figure 6. To date, this type of defect has only occurred in enriched experimental fuel elements and precautions are being taken to prevent it occurring in natural production fuel.

3. BOILING LIGHT WATER FUEL

An increasing program has been underway for the last six years investigating potential problem areas associated with fuel operating in a boiling water environment. This program now utilizes four in-reactor loops and can be divided into two parts. The first of these is a general program utilizing single fuel elements or small clusters of two or three elements to investigate problem areas over a very wide range of coolant conditions. The second part which began in late 1964 is aimed at the development of specific fuel bundle designs.

3.1 Bundles

3.1.1 Gentilly Fuel Design

The basic design philosophy for this fuel has been to use, where possible, the technology that has already been developed in the PHW program. However, a number of departures from PHW practice have been necessitated by the particular requirements of the BLW type of reactor.

The most significant of these modifications - a change in both element and bundle design - is due in large part to the fact that in a boiling reactor, the maximum heat flux on the fuel is limited by dryout[1]. Another important factor in this change is the requirement for BLW reactors to keep the amount of light water in the reactor core to a minimum by means of boiling to high qualities and limiting the coolant flow area within a bundle. Although the Gentilly reactor is based on a 10 cm channel diameter, it was felt that the above requirements could best be met by a 19-element radially pitched bundle rather than the 28-element 10 cm diameter bundle already under development for the Pickering reactor. The specific reasons for this choice were:

1) The better general understanding of the thermal and hydraulic performance of the 19-element geometry [23].

2) The greater amount of critical heat flux data available for the 19-element geometry [24

3) The smaller coolant cross-sectional area in a 19-element geometry than in a 28.

In the case of the design selected, the coolant cross-sectional area was reduced even further by the use of a 1 mm inter-element spacing rather than the 1.27 mm used to date in the PHW program.

A second major change from PHW practice resulted from the need in the Gentilly reactor to have all of the fuel bundles of a channel connected together to permit on-power refuelling from the bottom end of the reactor. To satisfy this requirement, the central element is removed from the basic 19-element configuration and this central vacant site is then used for a structural member which holds the bundles together in a string. This structural member is in the form of a gas filled tube with a spring at its lower end which applies a compressive load to the bundles in the string thus preventing relative rotational movement.

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1. Dryout (or critical condition) may be defined as the breakdown of the water film on the surface of a heated fuel element. This breakdown is accompanied by a sudden decrease in the local heat transfer coefficient, and a resultant sharp increase in sheath temperature.

3.2.1 Gentilly Fuel Bundle Development Program

The development program set up for Gentilly fuel is based on finding solutions to a number of envisaged problems, as well as general proof testing. The major part of this program is based on the irradiation testing of prototypical fuel bundles in the U-1 loop [17], but there is also a significant out-reactor testing program. Only the in-reactor portion of this program will be reported here.

The major anticipated problem areas on which the Gentilly fuel development program has been based are as follows.

1) The critical power capabilities of the bundle design at anticipated coolant conditions[1].

2) The sheath strain in the highly rated, large diameter elements (20 mm) which are a feature of the 10 cm diameter 18-element bundle design. The sheath strain generated by the volumetric expansion of the UO_2 and pressure from released fission gases should be limited to a reasonable level. Less than 0.8% uniform strain is considered a reasonable level for the following reasons:

   a) This strain reduces the flow area by $< 4\%$ which limits the increase in Δp to an acceptable level.

   b) On a statistical basis, the number of fuel failures in irradiation testing starts to increase above $\sim 1.0\%$.

   c) The stickage of fuel bundles in a channel becomes a possibility with strains between 1.5 - 2.0%.

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1. The central channel conditions in the Gentilly reactor are: $X_o = 16.5$ wt%, $G = 440$ g/cm^{2}s, surface heat flux = 100 W/cm^2, outer element rating $\int \lambda d\theta = 48$ W/cm + 10% over power margin, outlet pressure = 55 bars.

3) The fuelling scheme proposed for Gentilly and the method of fuel string assembly, subjects fuel bundles to temperature cycles, and pressure cycles not previously experienced.

4) The extent of sheath corrosion under boiling conditions, and the effect that the deposition of iron oxides has on this corrosion.

5) The possibility for the bundle assembly to fret at high steam velocities.

The irradiation program undertaken in the U-1 loop has provided answers in most of these areas and posed a few new problems.

3.1.3 Irradiation Program

The results of the boiling water cooled fuel bundle irradiations undertaken by CRNL are listed in Table 2. This table includes irradiations to develop a prototype fuel bundle for Gentilly and other bundle irradiations of a more general nature.

The Gentilly irradiations have demonstrated the following

1) Correlations based on both U-105 results [11 and out of reactor tests at Columbia University, predicts a critical power ratio of at least 1.5 for the maximum rated channel in the Gentilly reactor.

2) The bundle design can operate at outer element ratings of up to $\int\lambda d\theta$ = 66 W/cm, and to exposures of $\sim$ 300 days, or burnups up to 7,000 MWd/TeU. A metallographic section from one of the more highly rated bundles is shown in Figure 7. These bundles have also been subjected to estimated end flux peaking of

$\sim$ 22% and a gradient across a bundle diameter of up to 20%. The majority of the bundles tested contained a 2.5 volume % pellet dish to accommodate the volumetric expansion of the plastic UO_2 at high ratings, and consequently showed element diameter increases less than 0.39% for ratings up to $\int\lambda d\theta = 66$ W/cm. Other bundles with pellet dishes of 1.5 volume % have sustained average element diameter increases less than 0.5% at ratings up to $\int\lambda d\theta = 54$ W/cm. It is thus concluded that the 2.5 volume % pellet dishes specified for Gentilly are more than adequate to limit sheath strain to $< 0.5\%$.

3) Most bundles irradiated in the Gentilly program have included one or more of the following methods of providing a cold volume for released fission gas, thus limiting the internal pressure build up from this source to less than coolant pressure.

a) Hollow end plugs protected by delineating discs.

b) Circumferential grooves on the pellets.

c) Cylindrically shaped pellet dishes.

The amount of volume required at any given operating temperature has been estimated by the procedure outlined by Notley [24]. We have now demonstrated that each of these methods works, in that its ability to contain a specified volume of gas is maintained. For economic reasons hollow end plugs similar to that shown in Figure 8 have been chosen for the Gentilly fuel. Other bundles are being tested without extra void volume, but sufficiently high burnups have not yet been achieved on them to establish whether or not the Gentilly fuel could be designed without fission gas plenums.

4) Some Gentilly bundles have been removed from the loop, disassembled from their string, measured, and reassembled in different positions four times. No difficulties have been encountered in the handling. There have been no dimensional changes in bundles of the Gentilly reference design greater than had been anticipated, in consecutive loadings some bundles have been moved from low to high ratings, and others have been changed from high to low. (Element diameter changes have been less than 0.39% and bundle length changes less than 0.05%.)

5) In the Gentilly reactor it is planned to add ammonia to the coolant in sufficient quantities to maintain a concentration of 7 ppm in the water phase at exit [25]. This is necessary to reduce oxidizing conditions sufficiently to avoid corrosion of the mild steel piping used in the external circuit. However, Gentilly prototype fuel irradiations have shown that coolant radiolysis products persist in the core. The effect of this "oxidizing environment" and fast neutron flux has caused enhanced corrosion of the Zircaloy-4 fuel sheathing compared with cladding exposed to pressurized water coolant [27

   Oxide thicknesses obtained from two test bundles in U-106 are compared with sheath and corrosion coupon thicknesses obtained in other boiling irradiations and in American boiling water reactors, Figure 10. Extrapolation of this thickness to 1,000 days exposure gives a weight gain of 1,650 $\mu g/dm^2$ equivalent to a uniform penetration rate of 32.5 μm/year Effects of higher ammonia concentrations on sheath corrosion are being investigated.

6) Iron oxides released from corroding surfaces into the coolant, deposit on the fuel sheath surfaces reversibly. Deposits formed at high concentrations of particulate iron decrease in thickness as these concentrations fall. Deposits also increase in thickness as the heat flux is increased. In Gentilly the rate of iron oxide transported into the coolant by the feed water is balanced by removal in the purification circuit so that the iron concentration will average 0.1 mg/kg. The maximum steady state deposit thickness expected is about 25 μm (4,000 μg Fe/cm2). Although for short periods, following system disturbances high iron concentrations will produce transient thicker deposits. Deposition is not expected to cause any problems. Copper and nickel oxides form thicker, non-reversible deposits [26] but alloys containing them are not used in the Gentilly system.

7) Essentially no inter-element fretting has been detected in any irradiated bundles.

3.2 Fuel Elements

3.2.1 Types of Irradiation

A wide variety of tests with strings of single elements have been done in the boiling program. The majority of these tests have been collected into groups, representative results for each are given in Table 3, and the broad objectives are outlined below.

1) Low quality boiling tests. These tests for the most part have been done in the X-6 loop [17]. The general objective has been to investigate some element fabrication variables, and to study the chemistry of both ammonia dosed and neutral coolants, together with their effects on corrosion product deposition and the corrosion of various zirconium base alloys.

2) High quality boiling tests. These tests have all been done in the X-4 loop [28] and have in general used a steam water mixture at the test section inlet in order to achieve the desired quality. These tests have generally the same objectives as the low quality boiling tests, but place a greater emphasis on the behaviour of different fuel cladding materials.

3) Dryout tests. These tests have all been done in the X-4 loop using a steam water mixture at inlet. Their main objective has been to delineate the dryout or critical heat flux characteristics of different test geometries in order to set the operating conditions for the type of test outlined in either 2) or 4).

4) Time to defect tests. In these tests, done in the X-4 loop, Zircaloy clad fuel elements have been operated over a range of temperatures beyond dryout, to determine their time to defect, and to get a "feel" for the severity of such defects.

5) Defect tests. These tests, done in both the X-4 and X-6 loops, have been undertaken to determine the effect different flow regimes have on the behaviour of defected elements, both from the viewpoint of the defect signal and the rate of deterioration of the element.

3.2.2 Low and High Quality Boiling Tests

Other than the chemistry and deposition data which were generated in these tests [27], and some unexpected results reported in section 3.2.5, the main result from the fuel portion of these tests was to confirm that the fuel operated satisfactorily under these conditions. From the viewpoint of the low quality boiling tests (exit quality $< \sim 15$ wt% steam) this type of data are of increasingly less relevance as more data becomes available from the bundle irradiation program. However, some unique fabrication features tested in these experiments have yielded useful information such as:

1) The satisfactory irradiation behaviour of zirconium-2½% niobium clad fuel elements (with various post weld heat treatments) have led us to the following conclusions [29].

   a) Corrosion is less than on Zircaloy alloys.

   b) Corrosion at weldsis difficult to control with heat treatment, but such corrosion should not limit use of this alloy as a fuel sheath material.

   c) There is no evidence that low ductility in welds will preclude the use of cold worked fuel sheathing.

2) The satisfactory irradiation behaviour of long (1.8 m) fuel elements, as single elements and in trefoil clusters.

In the high exit quality tests (exit qualities in the range 27 to 60 wt% steam) the objective has been to use high steam qualities but to keep the surfaces of the fuel elements wet (i.e., not exceed the local critical heat flux condition). In order to achieve this roughened test sections have been used extensively. In such test sections the inside surface is equipped with 1.27 x 1.27 mm stripping rings at a 38 mm pitch. These strippers break up the otherwise thick water film on the cold test section wall, and throw it on to the surfaces of the fuel elements thus keeping them wet at higher qualities than would otherwise have been possible.

One general area of information coming out of the high quality tests is the effect of stripping rings on both coolant chemistry and the deposition pattern. Uncertainty regarding the water hold up within the test section complicates chemistry calculations and the "racoon tail" deposition pattern with its wide variations in thickness results in a very poor simulation of deposition at high qualities. However, based on the average deposition in such tests, it would appear as if the concentrating effect in the water phase has little significance, as the deposition rates are no higher than in low quality boiling tests, with the same iron concentrations.

An indication appears to be emerging from the duofoil tests that inter-element fretting is significantly higher in these assemblies after long periods of operation at high steam qualities. This may be due to hydraulic noise from the inlet mixer used to obtain the high qualities.

3.2.3 Dryout and Time to Defect Tests

Dryout tests have shown that even small geometrical changes can greatly affect the dryout performance of simple geometries. For this reason dryout tests using instrumented fuel elements have been undertaken to map out the operating envelope for most geometries before they are used for long term experiments. Dryout tests have been undertaken on the following geometries:

1) Single elements in a smooth flow tube [30]

2) Single elements in internally roughened flow tubes with various element to flow tube clearances.

3) Duofoil (2-element) assemblies in an internally roughened binocular shaped test section.

A dryout test on a trefoil geometry is in the final stages of preparation.

The techniques used in these tests and some of their results have been described previously [30], [31]. It has been found that the internally roughened flow tube geometries are more susceptible to up stream dryout[1]. This makes it difficult to ensure that very localized dryout similar to that shown in Figure 9 does not occur at some location where it is not detected by a thermocouple.

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1. Dryout that occurs up stream of the point of closest approach of the indicated dryout locus to the heat flux curve.

The duofoil geometry with an internally roughened flow tube was introduced as a means of increasing the number of elements in each test and subjecting pairs of elements to high coolant qualities while ensuring that each element in the pair was subjected to identical conditions. However, these geometries have yielded lower qualities than had been anticipated and the possibility of up stream dryout has further limited their use.

The results from the majority of the time-to-defect tests have been reported previously [31]. In these tests fuel elements were operated in a dried out condition at various sheath temperatures to determine the time necessary to cause defects due to the ensuing accelerated corrosion. These data plus some more recent results have been summarized in a graphical form in Figure 11.

3.2.4 Defect Tests

A number of experiments have been undertaken to investigate the effect that the various flow regimes possible in boiling reactors may be expected to have on defects. These tests have been undertaken at ratings varying from $\int \lambda d\theta$ = 35 W/cm to 58 W/cm with the following conditions at the defect location.

1) Sub-cooled water.
2) Sub-cooled boiling.
3) Low quality boiling.
4) High quality boiling
5) Dryout.

The first four flow regimes showed little effect on either the defect signal[1] or the rate of deterioration of the defected element. The dryout condition did, however, show an increase in signal level. One test also showed that a defected fuel element can not withstand continued operation in dryout with sheath temperatures ~ 700°C for longer than ~ 4.5 hours without risking gross deterioration.

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1. Except for the partition of some of the gaseous fission products between the steam and water phases.

All defect experiments had a standard hole area of ~ 0.1 mm^2, most utilizing a round hole. In one experiment, this hole was made in the form of a long narrow slit (simulating a crack), and it was found that although the experiment ran at a relatively high rating ($\int \lambda d\theta$ = 58 W/cm) the defect hole tended to plug up during operation.

3.2.5 Unexpected Results

A number of incidents have occurred in the irradiation program which, while not planned, have contributed to our general understanding of the operation of fuel elements in a boiling environment. Historically, the first such incident occurred during the in-reactor commissioning of the then new X-6 boiling loop. The fuel for this commissioning consisted of a string of single wire wrapped elements of the type used in the bundles for the Douglas Point reactor. Several fuel defects occurred after short periods (~ 50 days) in a boiling environment. As can be seen from Figure 12, the cause of the defects was due to poor local heat transfer immediately down stream from the wire wrap which led to accelerated corrosion. It would appear that in a simple annular geometry, the wire wrap removes the water film from the fuel element in much the same way as stripping rings do from the wall of internally roughened flow tubes[1].

Another incident occurred during an experiment intended to investigate the drastic reduction in coolant flow that might be anticipated during a pump failure in a CANDU-PHW reactor. Although the experiment was set up to determine the seriousness of the fuel failure that would occur as the flow was reduced in steps[2], the results were made useless due to the effects of a loop flow instability. A liquid deficient condition occurred over a large portion of the fuel string length resulting in very high sheath temperatures and the disintegration of the fuel string as

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1. It should be noted that this effect does not occur when the wire wrapped elements are part of a bundle (see U-201 in Table 2).

2. The power produced by the fuel was maintained constant at a value to produce a peak heat flux of ~ 120 W/cm^2.

shown in Figure 13. Analysis of the experiment indicated that the extensive liquid deficient condition was probably due to an elastic loop condition initiated by boiling in a loop heater. The extensive fuel damage was due in large part to the sheath temperature exceeding the Leidenfrost point and thus preventing re-wetting. It is also believed that much of the mechanical damage was done when the flow was returned to normal while the elements were still hot.

Perhaps a more appropriate answer to this question occurred during the preparation for a dryout experiment (utilizing a thermocoupled element) when the loop pumps were inadvertently cut off. The flow in the loop, which was operating with sub-cooled water, fell rapidly and the sheath temperature on the fuel element rose rapidly to $\sim$ 530°C at which point the effect of the reactor trip, caused by the fall off in loop flow, started to reduce the fuel temperature. Extensive post-irradiation examination of the fuel (which had been operating at a surface heat flux of $\sim$ 76 W/cm^2)showed no signs of any damage.

A further incident which has contributed to our knowledge of fuel element behaviour during accident conditions occurred during a boiling deposition test utilizing a simulated feed train. In this instance, a gasket blew out in the simulated feed train causing a rapid loss of pressure in the main loop (from $\sim$ 70 to 45 bars), and a 17% reduction in coolant flow due to pump cavitation. A thermocoupled fuel element operating at a surface heat flux of $\sim$ 126 W/cm^2 experienced a temperature spike of $\sim$ 70°C at the location of its thermocouple, and no damage to the fuel element was detectable during post-irradiation examination.

In an experiment where it had been planned to irradiate a single 183 cm long element containing a number of specific design features, one of which was the inclusion of only three centering spacers at its center plane, severe loop pressure oscillations were encountered as it was being taken into its boiling mode of operation. The phenomena has subsequently been found to be repeatable with other fuel elements with the above two design features. While analysis is not yet complete, a tentative model has been advanced that accounts for the bulk of the phenomena observed. This model ascribes the violent pressure oscillations observed in the test section to an amplication of resonant pressure pulsations within the loop proper. The amplification mechanism proposed is that the loop pulsations excite the fuel assembly into a transverse vibration which in turn causes local boiling at the point of approach of the fuel element to the pressure tube wall. Visual examination of the fuel elements involved has revealed no indication of local over-heating, however, no definitive statement can be made regarding local over-heating until these elements have been destructively examined.

4. FUTURE PROGRAM

Future CANDU-PHW reactors will probably be larger in size (750 to 1,000 MWe) [32], but will probably use fuel similar to the present Pickering reactors except that it will operate at heat ratings of $\int\lambda d\theta$ = 50 W/cm or higher. These reactors may be designed to take maximum advantage of plutonium recycling [33] and may thus require extensive modifications to existing fuel fabrication procedures. This decision will depend on the demand for the plutonium in our spent fuel by the world market.

Because of the light water load in a natural uranium BLW reactor, there is considerable economic incentive both to decrease the coolant density by increasing the exit quality, and increase the uranium density of the fuel.

As a means of increasing the exit steam qualities at which boiling reactors may operate, two parallel programs are being undertaken. The first of these investigates the use of new bundle geometries in which sub-channels are sized for a balanced enthalpy rise through the bundle. The second will investigate the use of more corrosion resistant alloy sheaths, permitting the critical condition to be exceeded. These include zirconium base alloys such as the Russian alloy Ozhennite [34] zirconium-1% chrome-0.1% iron [35] , and zirconium-2½ niobium aimed at operation at temperatures up to 500°C.

A major development program is now in progress to develop U_3Si as a high uranium density fuel. An irradiation program [36], [37] of single fuel elements has been successful in achieving burnups to 12,000 MWd/TeU,demonstrating that the irradiation swelling of the material can be controlled and that the hot water corrosion rates are acceptable. An effective increase in uranium density of approximately 30% is anticipated with U_3Si fuel, and if the cost of the various fabrication steps can be kept as low as for UO_2 fuel, U_3Si may well be designated as the reference fuel material for the next CANDU-BLW reactor.

Other small improvements are also being made to the present UO_2 bundle design to increase its effective uranium content These include the use of thinner sheathing and the fuelling of the now hollow central structural member.

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TABLE 1

IRRADIATION PERFORMANCE OF FUEL BUNDLES

EXPERIMENT	PURPOSE OF IRRADIATION	BUNDLE TYPE[a]	NUMBER OF BUNDLES	MAXIMUM FULL POWER DAYS	MAXIMUM COOLANT DAYS	RANGE $\int \lambda d\theta$ W/cm	BURNUP[b] RANGE MWd/TeU
			PRESSURIZED HEAVY WATER IRRADIATIONS				
NPD	First Loading	7-element	850	1,392	1,468	25-35	7,960
NPD	First Loading	19-element	643	1,164	1,468	15-20	8,750-11,300
NPD-101 103 1301	Design & High Burnup	A	4	931	956	14-27	10,000-20,000[d]
NPD-1201	Refuelling	A	310	729	748	10-20	7,900
NPD-301 601	Design & Refuelling	B	63	600	609	18-33	7,000-11,300
NPD-701 901 1001 1101 1701	Development Irradiations	B[c]	17	594	603	18-42	3,200-13,000
Douglas Point	First Charge	A	3,661	56	100	43	2,400
	Replacement	B	12	56	100	43	2,400
			PRESSURIZED LIGHT WATER IRRADIATIONS				
U-206 Ph I Ph II	Douglas Point Replacement Prototype & Development Tests	B B[c]	12	228	256	22-53	1,300-8,600[d]
U-209 210 214 219 218	Pickering Development Prototype Tests	C C[c]	25	226	232	25-49	2,000-9,150

LEGEND

a) See Figure 2.
b) Burnups based on calorimetric data, outer element.
c) Development bundles using variations of welded and brazed spacers.
d) Burnups based on isotopic analyses.

TABLE 2

IRRADIATION PERFORMANCE OF BOILING LIGHT WATER FUEL BUNDLES

EXPERIMENT	PURPOSE OF IRRADIATION	BUNDLE TYPE	NUMBER OF BUNDLES	MAXIMUM FULL POWER DAYS	MAXIMUM RATING RANGE ($\int\lambda d\theta$ W/cm)	MAXIMUM BURNUP RANGE (MWd/Te)	COOLANT CONDITIONS: MASS VELOCITY (g/cm^2s)	COOLANT CONDITIONS: EXIT QUALITY (%)	COOLANT CONDITIONS: CHEMISTRY	SHEATH CONDITIONS: MATERIAL	SHEATH CONDITIONS: OXIDE THICKNESS (μm)	STATUS
Boiling Light Water Irradiations												
U-101	Tests to determine critical heat flux capabilities of various bundle designs under a variety of ratings	B	10	N/A	55	1000	135-270	16.5-46	Neutral	SS & Zr-2	5-8	Completed
U-105		D	10	35	70	1000	135-270	30-60	NH_3	SS & Zr-4	1-6	Completed
U-103	Operation under neutral boiling conditions	B	6	135	19-41	5000	250	30	Neutral	5 Zr-2 1 Zr-4	10 25	Completed
U-101 IV	Operation under neutral boiling conditions	B	7	138	45	5800	250	20	Neutral	Zr-2	-	-
U-106 I	Operation under ammoniated boiling conditions and development of various design features for Gentilly and Advanced BLW reactors	D	Total of 19	18	0^1-61	525	440	~11	NH_3	Zr-4	2-11	Replaced by Succeeding Phases
106 II		D		52	0^1-61	2085	440	~11	NH_3	Zr-4		
106 III		D		21	0^1-61	590	440	~11	NH_3	Zr-4		
106 IV		D		½	-	-	-	-	-	-		
106 V		D		112	0^1-61	5350	440	~11	NH_3	Zr-4	22	
106 VI		D		N/A	0^1-61	N/A	274	~17	NH_3	Zr-4		Continuing
U-110		D	5	46	27-61	1450	225-440	0-22	NH_3	Zr-4		Completed
U-112 I		D[2]	5	55	27-61	1252	440	7-9	NH_3	Zr-4	N/A	To be Irradiated
U-201	Operation with NH_3 dosing but in a closed cycle	A & B	6	160	50	6800	500	11	NH_3	Zr-2	N/A	Completed

LEGEND

1. The extreme ends of the fuel string were out of the reactor flux.
2. Fuel string included two bundles with 0.42 mm thick sheaths and one U_3Si fuelled bundle.

N/A - Not Available

TABLE 3

SUMMARY OF SINGLE ELEMENT BOILING IRRADIATIONS

EXPERIMENTAL GROUP	CONSTITUENT EXPERIMENTS	OBJECTIVES	NUMBER OF SPECIMENS	DURATION DAYS	MAX. RATING Q/A (w/cm^2)	MAX. RATING $\int \lambda d\theta$ (w/cm)	MATERIALS	MAX. SHEATH TEMPERATURE NOM. (°C)	MAX. SHEATH TEMPERATURE PEAK (°C)	STEAM QUALITY (%)	CHEMISTRY	DEPOSITION THICKNESS µm	NET H_2 (ppm)	OXIDE THICKNESS (µm)	NOTES
	609 I		6	67	109	40	Zr-2, Zr-Nb, Zr-4	310	-	7	Neutral	0.01-0.2	28-84	2-27	8,9
	610	To verify some element	12	126	135	50	Zr-2, Zr-Nb, Zr-4	310	-	18±1	NH_3	0.01-1.6	0.17-0.72	1-3	
Low Quality	611	fabrication variables and	6	29	98	44	Zr-4	310	-	13	NH_3	0.05-1.7	N/A	N/A	Closed Cycle, BLW Elements
Boiling	612	to study coolant chemistry,	5	33	167	60	Zr-4	285	-	<14.9	NH_3	N/A	N/A	N/A	
Experiments	615 II	deposition and corrosion as	5	17	135	50	Zr-4	315	-	14-20	NH_3	N/A	N/A	N/A	8
	639	a function of coolant	3	~10	147	55	Zr-2	315	-	0	NH_3	N/A	-	-	Oscillations Encountered
	640	conditions.	6(Trefoil)	~120	122	45	Zr-2	310	-	9.5	NH_3	N/A	N/A	N/A	Undergoing Examination
	422	To study the relative	2	61	95	34	Zr-2, Zr-Nb	295	295	25-27	NH_3	1.0	~20	1-6[1]	8,9
High Quality	426	behaviour of various alloy	7	20	95	35	Zr-2, Zr-Nb	295	295	37	NH_3	0.04-0.23	N/A	N/A	8,9
Boiling Tests	427 II	fuel sheathing materials at	7	34	95	35	Zr-2, Zr-Nb	295	295	35	Neutral	0.06-2.6	N/A	N/A	8,9,10
	428	high steam qualities also	5	129	113	42	Zr-2, Zr-Nb, Zr-4	295	295	46.5	NH_3 + Fe	0.5-2.6	N/A	N/A	8,9,10
	430 II	chemistry, deposition and	4	35	120	43	Zr-4	295	295	60	NH_3	N/A	N/A	N/A	2,3,8,10
	432	corrosion as a function of coolant conditions.	10(Duofoil)	50[5]	120	43	Zr-4, Zr-Nb	295	295	51	NH_3 + Fe	0.8-2.8	N/A	N/A	2,7,10
Dryout Tests		To delineate the dryout characteristics of the following geometrics:													
	421	Single element in plain flow tube	2	17	100	36	SS, Zr-2, Zr-Nb	295	-	20-30	NH_3	1.2	20	-	See AECL 1819
	427 I	Single element in roughened flow tube	3	2	125	45	SS, Zr-2	520	557	95	NH_3	0.13	177[7]	1-50	10
	430 I	Single element in roughened flow tube	3	10	103		Zr-4	295	530	90	NH_3	N/A	N/A	N/A	2, BLW Elements
	431	Duofoil in roughened flow tube	6(Duofoil)	60	140	50	Zr-4	295	~350	50	NH_3 + Fe	3.3-4.3	N/A	N/A	
Time to Defect	423	To determine the time to	4	32	62	22	Zr-2, Zr-Nb	380	400	59(Dry)	NH_3	0.04	>200	1-50[1]	See AECL-2016
Tests	424 I	defect Zircaloy clad fuel	5	1½[4]	88	32	Zr-2, Zr-Nb	~600	~700	41(Dry)	NH_3	0.12	500-6000	10-370[1]	See AECL-2016
	II	elements when operated at	4	2[4]	70	25	Zr-2, Zr-Nb	~430	<700	41(Dry)	NH_3	0.02	1000-4000	1-100[1]	See AECL-2016
	III	various temperatures in	4	4½	68	25	Zr-2	460	~550	58(Dry)	NH_3	0.03	100-400	<1-40[1]	See AECL-2016
	431	dryout.	6(Duofoil)	1[4]	140	50	Zr-4	~800	~800	82	NH_3	N/A	N/A	N/A	10
	607		7	2[4]	120	25	Zr-2	310	>1000	0-50	-	-	-	-	
Defect Tests	425 I	To study the behaviour of	3	28	97	35	Zr-2	280	280	26	NH_3	0.02	-	0.5-40[1,6]	8
	425 II	intentionally defected fuel	3	2	58	21	Zr-2, SS	500	~700	44(Dry)	NH_3	0.02	-	3-30(1,6)	8
	606	elements in various flow	3	17	107	39	Zr-2	312	-	0-9	NH_3	4.0-4.7	2.30	0.5-56[1,6]	8
	632 I	regimes.	5	8	177	66	Zr-2	315	-	9.5	NH_3	N/A	N/A	N/A	8
	632 II		4	6	156	58	Zr-2	315	-	8	NH_3	N/A	N/A	N/A	8

LEGEND

(1) With pits.
(2) Contained one U_3Si fuelled elements.
(3) Contained one vibratory compacted UO_2 element.
(4) Defected.
(5) Defected at startup.
(6) On sheath I.D.
(7) Hydrocarbon contamination.
(8) Visual appearance of fuel good.
(9) Corrosion at weld on Zr-Nb specimens.
(10) "Racoon tail" deposition pattern.

N/A Not Available.

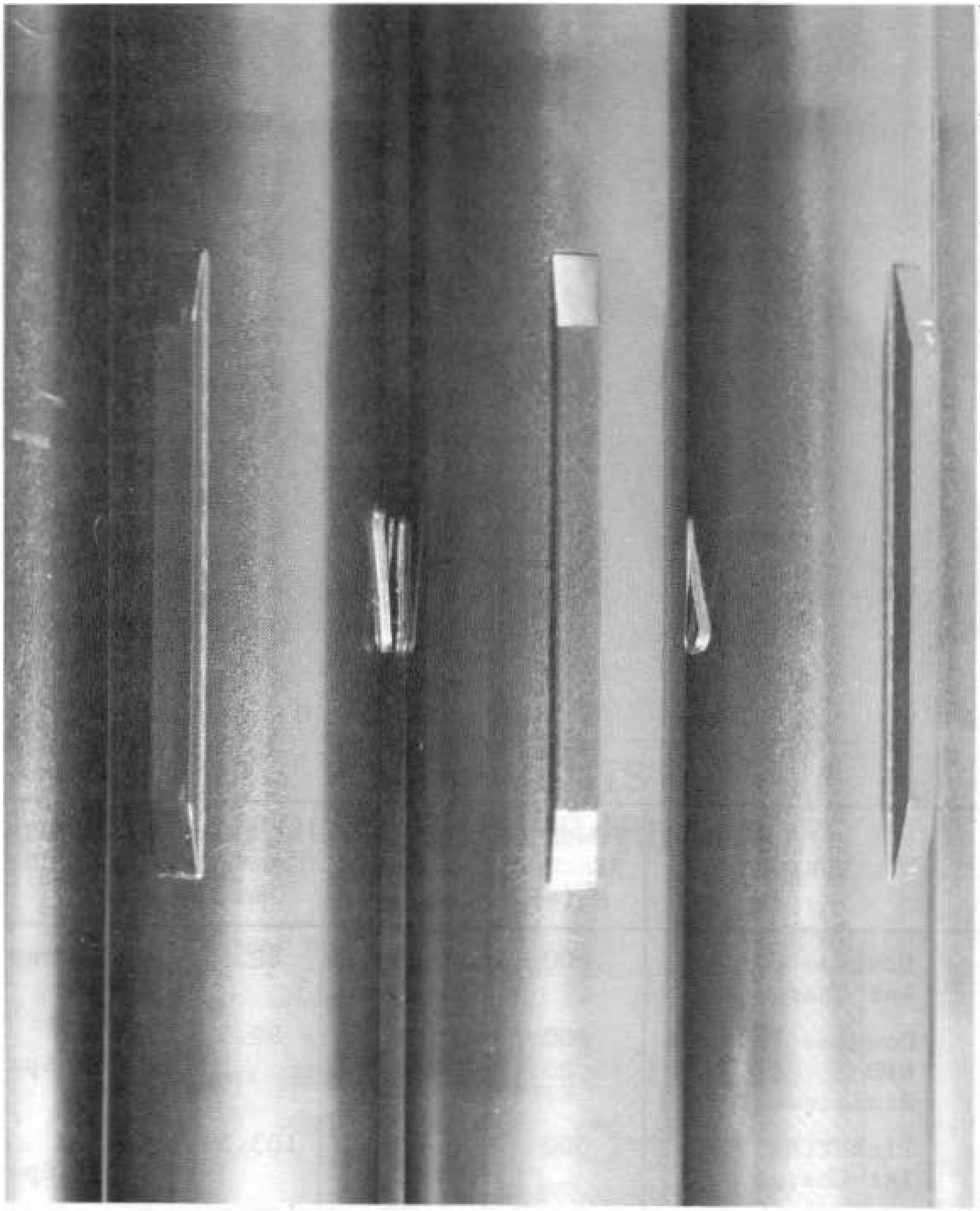


FIGURE 1

Close up of Brazed Spl. Spacer and Bearing Pads

A B C D

TYPE	PURPOSE	POWER OF REACTOR MW(e)	NO. OF ELEMENTS	BUNDLE DIAMETER mm	SPACING METHOD
A	Douglas Point 1st Charge	200	19	82.5	Wire-Wrap
B	Douglas Point & NPD Replacement Fuel	200 25	19	82.5	Brazed Split Spacer
C	Pickering 1st Charge	500	28	103.5	Brazed Split Spacer
D	Gentilly 1st Charge	250	18	103.5	Brazed Split Spacer

FIGURE 2

Fuel Designs for Canada's Power Reactors

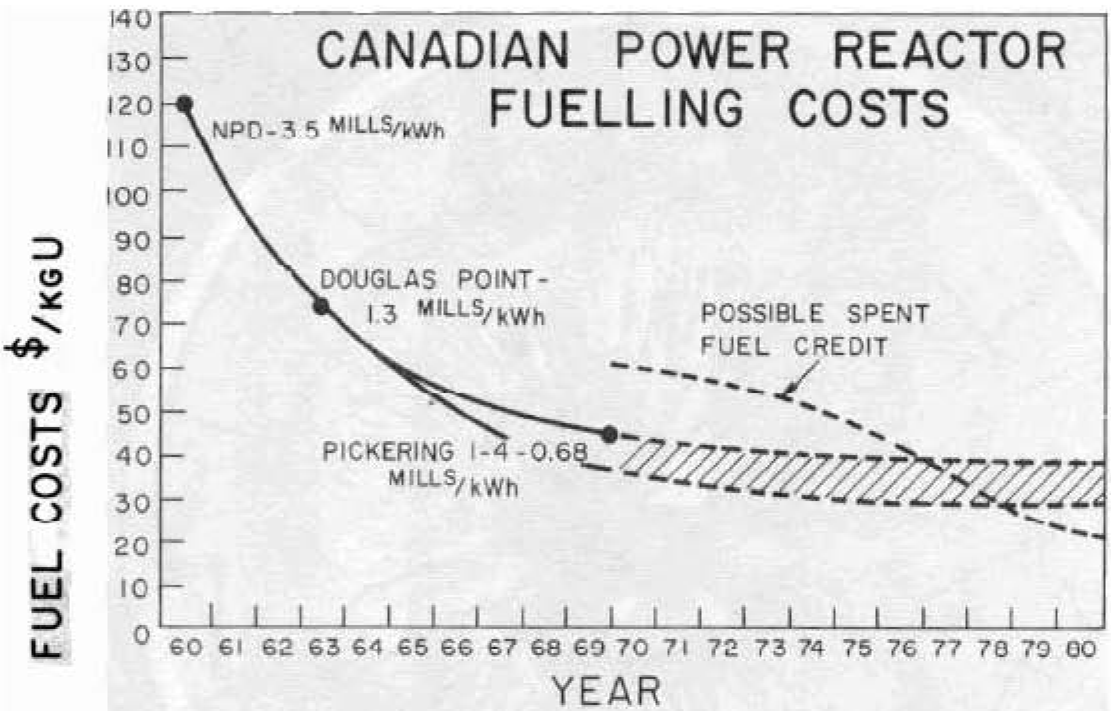


FIGURE 3

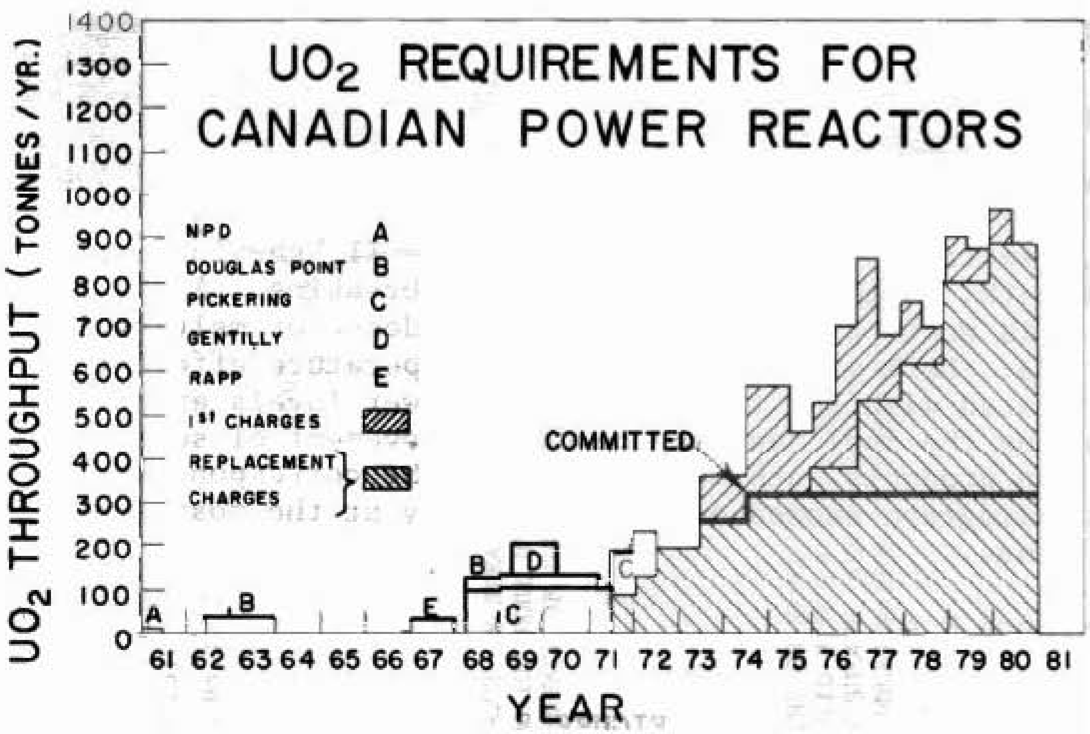
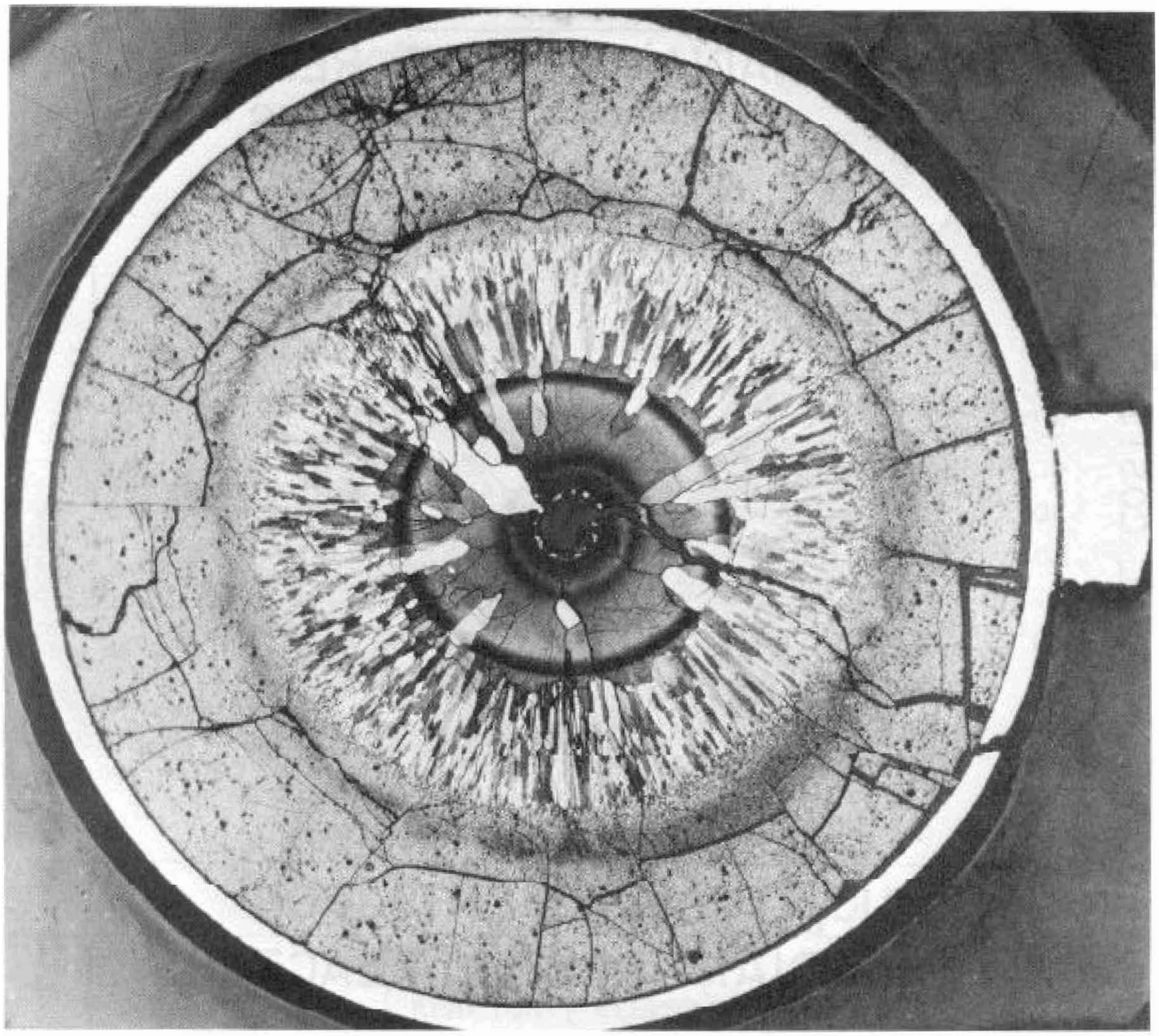


FIGURE 4



Cross-section of fuel from U-210, Phase II, showing split in sheath. Note sheath necked before breaking. Also shown is grain growth and possible evidence of melting in the UO_2. Dark etching rings are a temperature effect associated with the three different power levels experienced by the fuel, and may be due to movement of solid fission products such as ruthenium. The corresponding autoradiograph showed depleted activity at the positions of the rings.

Reference R 47 A 1 7.5X

FIGURE 5

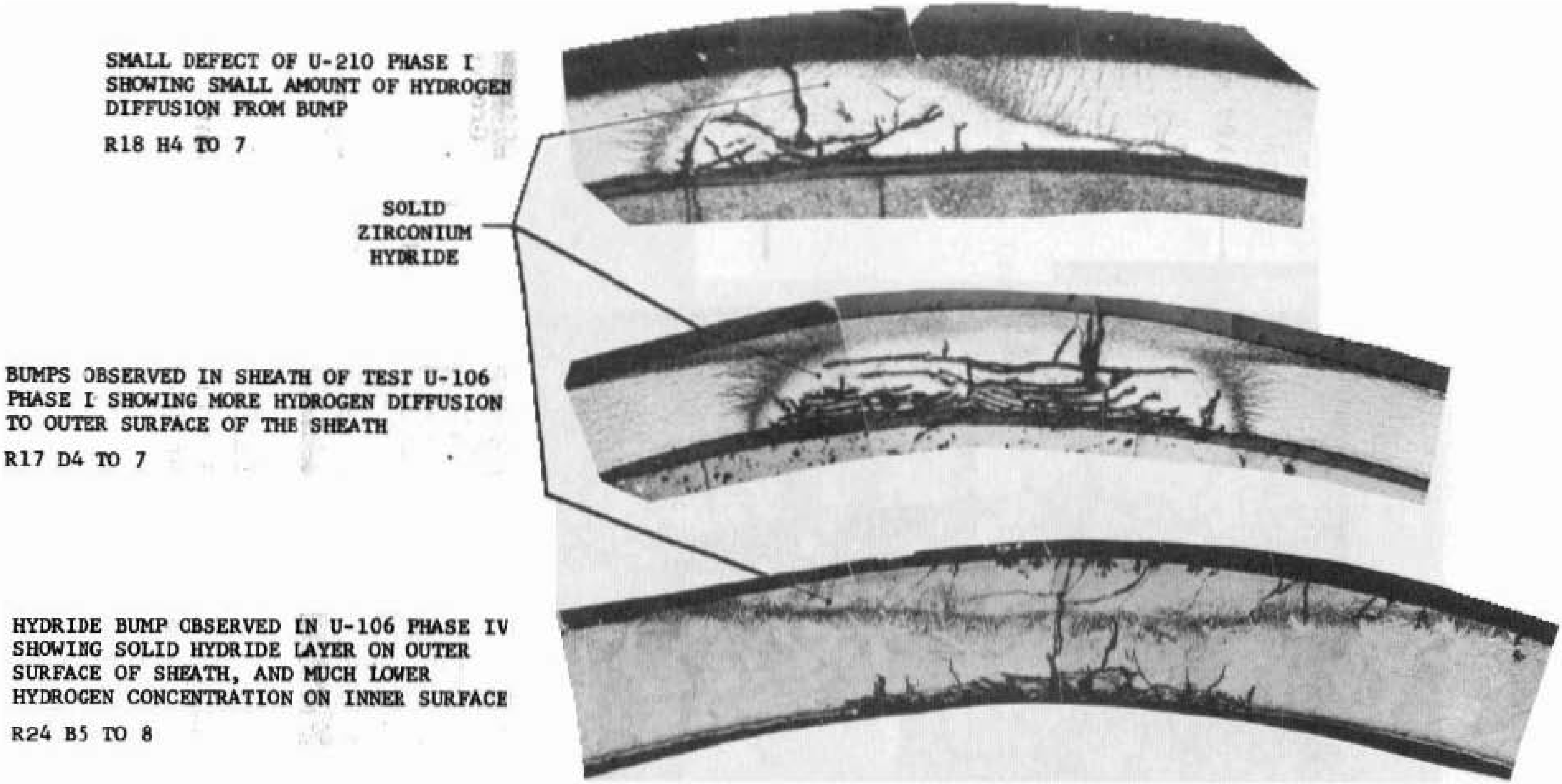
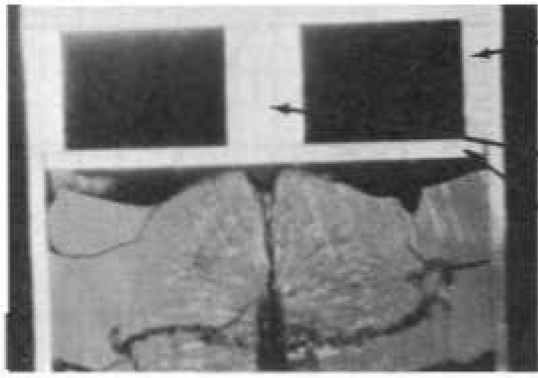


FIGURE 6

Cross Sections of Zircaloy Hydride Blisters in Fuel Sheathing

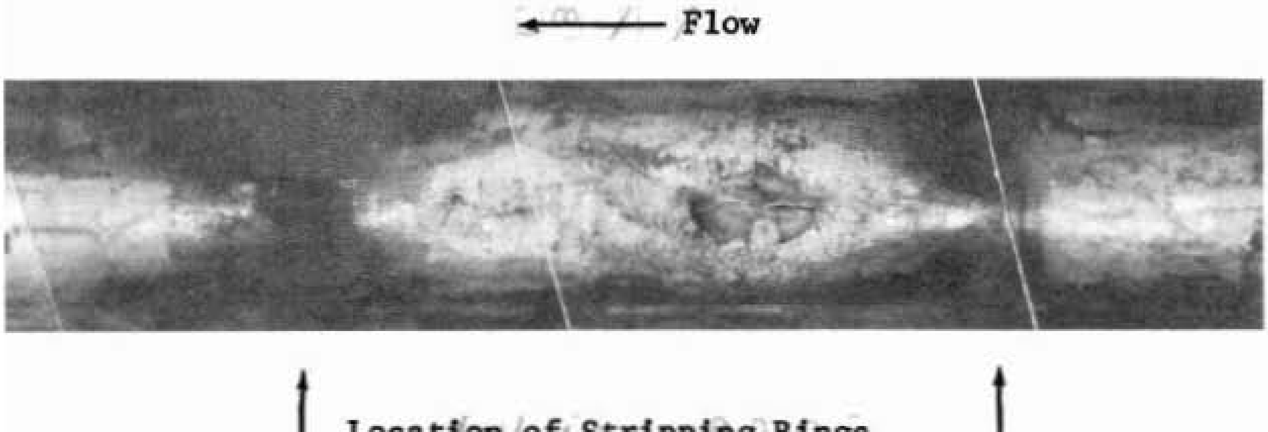
FIGURE 7

Longitudinal Cross Section of BLW Fuel Element at High Heat Rating



Longitudinal Cross-Section of Delineated Hollow End Plug
at $\int \lambda d\theta \geq 50$ W/cm

FIGURE 8



Localized Dryout Pattern with Internally
Roughened Test Section

FIGURE 9

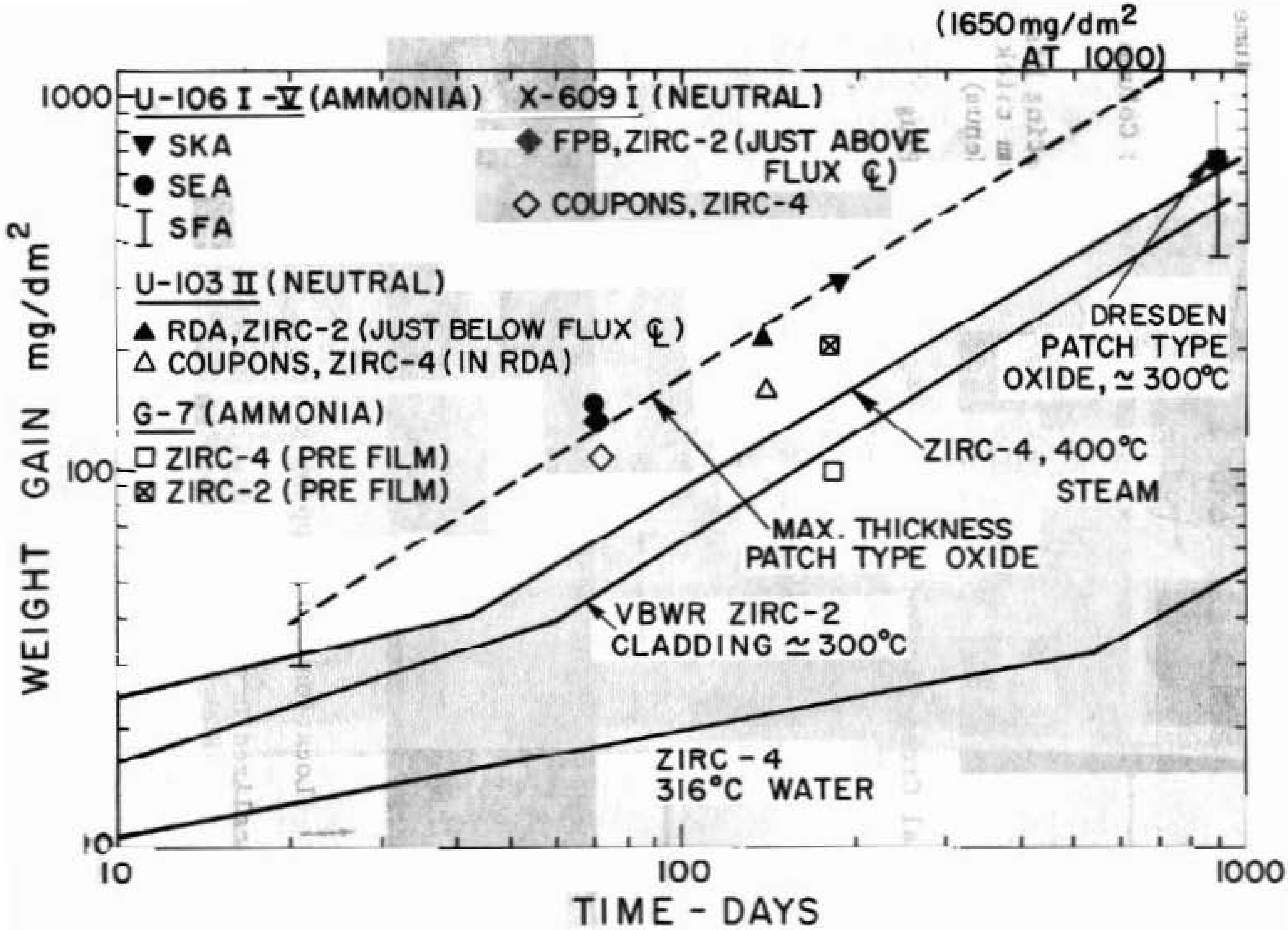
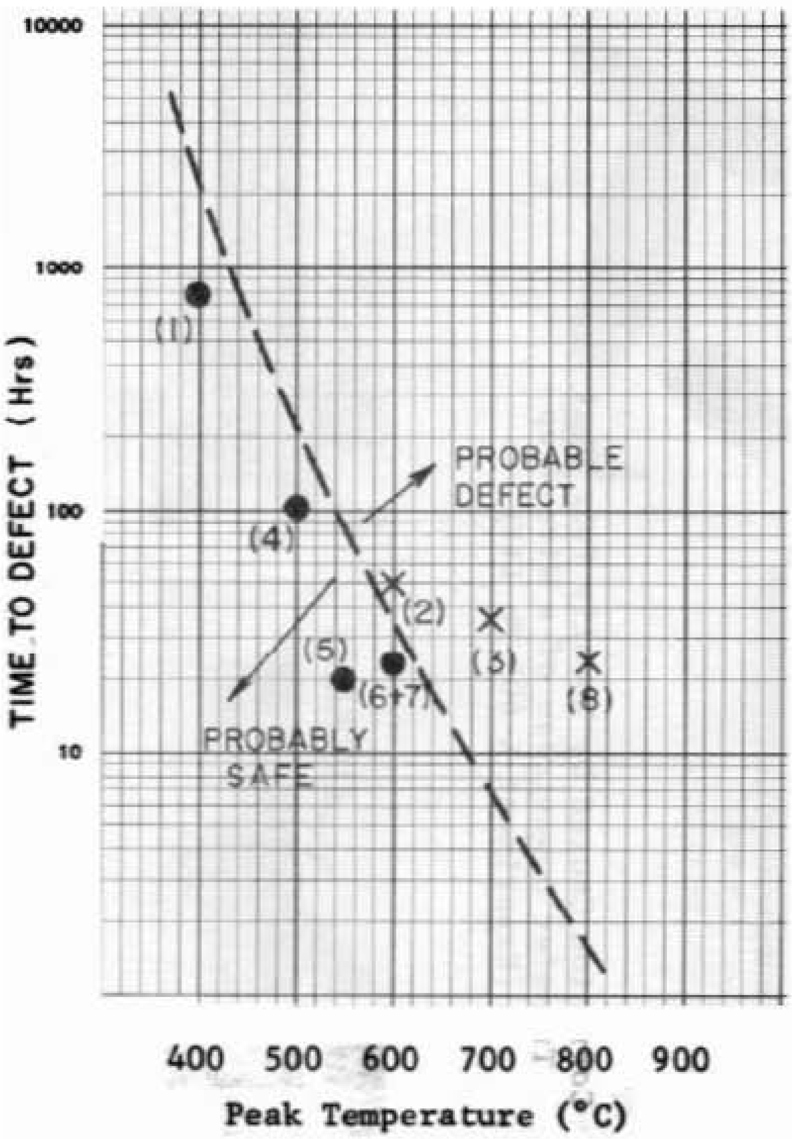


FIGURE 10

Gentilly Prototype Sheathing Corrosion Compared with Data from Neutral Boiling Water Irradiations



LEGEND

X Defected in time shown
● Did not defect during time shown
(1) X-423
(2) X-424 Ph I
(3) X-424 Ph II
(4) X-424 Ph III
(5) X-427 Ph I
(6) X-431 Element ASB
(7) X-431 Element ASE
(8) X-431 Element ASF

Results from Time to Defect Tests Undertaken on Zircaloy-2 Clad Elements in Dryout

FIGURE 11

Surface Heat Flux ≈ 140 W/cm

Corrosion Caused by Locally Poor Heat
Transfer Downstream of a Wire Wrap
(Experiment X-605 Ph III)

FIGURE 12

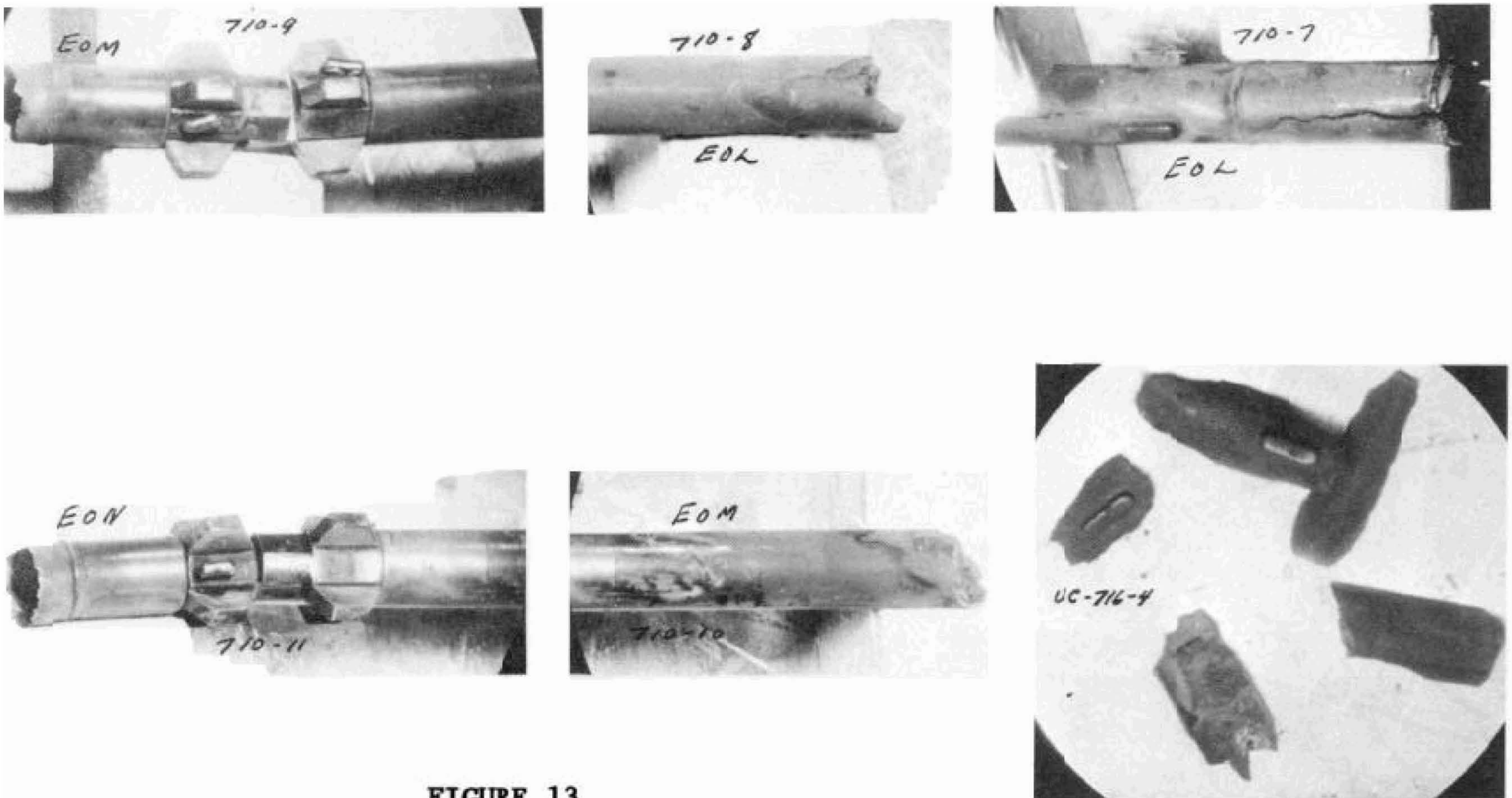


FIGURE 13

Example of Damage Done to the U-607 Fuel String by Up-Stream Dryout