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L'ÉNERGIE ATOMIQUE
DU CANADA LIMITÉE

EXAMINATION OF CORE COMPONENTS REMOVED FROM CANDU REACTORS

EXAMEN DES ÉLÉMENTS DE COEUR RETIRÉS DES RÉACTEURS CANDU

**B.A. CHEADLE, C.E. COLEMAN, D.K. RODGERS, P.H. DAVIES,
C.K. CHOW and M. GRIFFITHS**

Presented at the International Conference on CANDU Maintenance
Sponsored by the Canadian Nuclear Society, Toronto, Ontario, 1987 November 22-24

Chalk River Nuclear Laboratories
Laboratoires nucléaires de Chalk River

Chalk River, Ontario

November 1988 novembre

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[1] Materials & Mechanics Branch
[2] Advanced Materials Research Branch

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RÉSUMÉ

Les éléments du coeur d'un réacteur nucléaire se dégradent du fait que le milieu est rigoureux. Par exemple, dans les réacteurs CANDU, les tubes de force doivent faire face aux effets de l'eau chaude pressurisée et des dégâts par un flux de neutrons rapides. Pour évaluer toute détérioration d'éléments et déterminer la cause de la rupture occasionnelle, nous avons mis au point une grande variété de techniques de manutention à distance pour examiner les matériaux radioactifs. En plus des tubes de force, nous avons examiné les tubes de cuve, les ressorts bracelets, les raccords d'extrémité, les barres à liquide de règlage zonal ainsi que les détecteurs de flux. Les résultats de ces examens ont donné la solution des problèmes et donnent continuellement des renseignements qui aident à comprendre les processus pouvant limiter la durée d'utilisation d'un élément.

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[1] Matériaux et Mécanique
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ABSTRACT

Components in the core of a nuclear reactor degrade because the environment is severe. For example, in CANDU reactors the pressure tubes must contend with the effects of hot pressurised water and damage by a flux of fast neutrons. To evaluate any deterioration of components and determine the cause of the occasional failure, we have developed a wide range of remote-handling techniques to examine radioactive materials. As well as pressure tubes, we have examined calandria tubes, garter springs, end fittings, liquid-zone control units and flux detectors. The results from these examinations have produced solutions to problems and continually provide information to help understand the processes that may limit the lifetime of a component.

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INTRODUCTION

To operate a reactor at high availability, the components subject to deterioration must be replaced before they fail. The components in the core must withstand combinations of temperature, stress, corrosion and neutron irradiation which can affect their surface condition and change their mechanical properties and dimensions. Research programs study the mechanisms of these processes so that the changes can be predicted. Valuable information is also obtained from the examination of components that have been removed after periods of service either because they were defective or to evaluate their condition. Although most of the components examined and tested are from fuel channels, flux detectors and liquid-zone control units have also had to be removed. Techniques have been developed to examine these components in protected bays and cells either to determine why they were defective or to assess their condition and predict the service life of similar components.

CAPABILITIES

Examination of components removed from power reactors requires the use of facilities with sufficient shielding to provide protection from the radiation fields. Two types of facilities are used: bays or pools where water reduces the radiation fields, and cells where concrete, lead and oil-filled lead-glass windows provide adequate protection. The facility selected for a particular task will depend on the size of the component, the radiation fields, and the availability of specialized equipment. The facilities required for almost any type of material examination or test are provided for irradiated material.

The metallurgical examination of any component, whether irradiated or unirradiated, follows a logical progression from non-destructive evaluations of the entire component to microscopic examinations of small sections of material. The preliminary visual examination guides all subsequent examinations. Using a video camera, a telescope or stereo-microscope, a detailed photographic record of the appearance of a component is generally made as a first step in the examination. Important features, such as welds, design reference positions and marks produced during service, are located. Much attention has been paid to establishing the positions along pressure tubes at which the garter springs operated, and the appearance of the pressure tube in contact with the calandria tube when the garter springs have been out of their design positions.

Before any cutting operations non-destructive tests may be required. Dimensional measurements can provide much useful information; for example, for pressure tubes the calculation of the as-rolled residual stresses in the rolled joint area, the location of garter springs during operation and the sag and diametral creep. Many years of research and development have produced ultrasonic and eddy-current equipment for examination of irradiated components.

The shielded cells at Chalk River Nuclear Laboratories (CRNL) and Whiteshell Nuclear Research Establishment (WNRE) contain metallurgical and chemical laboratories and much machine-shop equipment. Samples for metallography, hydrogen and deuterium analysis, and fracture toughness and tensile testing, are prepared using in-cell saws, lathes, milling machines, punches and drill presses before being examined on optical or electron microscopes, subjected to chemical analysis or tested on computer-controlled servo-hydraulic mechanical testing machines.

In addition to the facilities for highly active material, there are a number of "active" laboratories. The delicate procedures required for fractographic examinations and chemical analysis necessitate the handling of small amounts of material which, as a result of their size, are limited in total activity and can be handled carefully. These active laboratories also provide for such specialized examinations as surface-roughness measurements on plastic replicas of highly active pressure-tube surfaces; the replicas are only slightly active and contamination of the measurement instruments is restricted to the stylus in contact with the replica.

The diverse nature of AECL Research Company's ability to deal with irradiated material has enabled us to examine many CANDU core components. Although most of the effort has been concentrated on the fuel channel, particularly the pressure tube, components such as flux detectors have been examined, Table 1.

EXAMINATION OF COMPONENTS

Cracking at Rolled Joints

The pressure vessels in CANDU reactors are seamless tubes made from cold-worked zirconium alloys. They are mechanically joined to 403 stainless-steel end fittings by a rolled joint. In 1974-5 in Pickering Units 3 and 4 leaking cracks developed in a few Zr-2.5 Nb pressure tubes just inboard of the rolled joints. The cracks started on the inside surface and grew by a mechanism called delayed hydride cracking (DHC). An important feature of these failures was the presence, over a length of about 20 mm, of large residual tensile stresses in the hoop direction, up to 600 MPa, caused by an improper rolling procedure, Figure 1. The measurements of stress were made using a strain gauge and slitting technique in protective cells. The pressure tubes are designed to withstand a hoop stress from the pressure of the coolant of about 120 MPa, so the residual stresses were clearly unacceptable.

When the cracks and large residual stresses were discovered, Bruce Units 1, 2 and 3 were already built and their rolled joints had the same fault. To save retubing the reactor, the regions of high residual stress in each tube were heated to about 500°C for 0.5 h to relieve the stresses. This treatment was too late for two tubes that cracked in Unit 2 in 1982. However, measurements of residual stress showed that the heat treatment had been successful - the residual stresses had been reduced to between 50 and 100 MPa - so the prognosis for the remaining tubes is good. In subsequent reactors the rolled joints have been made with much smaller residual stresses, and in some reactors the stresses are minimised by keeping the clearance between the pressure tube and its end fittings as small as possible. In a "zero clearance" rolled joint the residual stresses are very low or compressive.

The stresses in Pickering Units 3 and 4 are decreasing because of stress relaxation. For example, after 10 years of operation, at the cooler inlet end a typical initial value of 550 MPa has reduced to 300 MPa while at the hotter outlet end this stress is expected to be close to 200 MPa. Thus the probability of cracking is continually being reduced.

Rupture of a Zircaloy-2 Pressure Tube

A Zircaloy-2 pressure tube ruptured in Pickering Unit 2 in 1983. The rupture occurred because a series of cracks had initiated on the outside surface of the pressure tube. These cracks grew and linked together forming a partial through-wall crack and when this crack reached about 100 mm long, the tube ruptured. The problem was caused by hydride blisters, Figure 2. These had formed because one of the garter-spring spacers was not in the correct position, which had allowed the pressure tube to sag down and contact the calandria tube. This introduced thermal gradients in the pressure tube at the points of contact, causing the hydrogen and the deuterium to diffuse down the thermal gradients and form a series of solid hydride blisters, Figure 3. Examination of other pressure tubes in Pickering Units 1 and 2 showed that there were many fuel channels that did not have garter springs in their correct positions, the pressure tubes were in contact with the calandria tubes and hydride blisters had formed. For the other reactors that had the same design of garter springs, equipment and techniques have been developed to determine the position of the garter springs and move them back into their correct location. A new garter-spring spacer was developed at the Sheridan Park laboratories that is a tight fit on the pressure tube and should not move during reactor construction and operation. A similar garter spring was used originally in NPD and observations there confirm that such a spacer has not moved during 25 years of operation.

Flaws in Pressure Tubes

Cracks can initiate in zirconium alloys by DHC if there is a combination of notch and stress that is greater than a critical value. The stresses in pressure tubes from the pressurised heavy-water coolant are too low to initiate cracks unless there is a sufficiently large notch present. Two pressure tubes in Bruce Unit 2 have developed through-wall cracks by DHC because they contained flaws. These flaws were formed during the fabrication of the tubes, but they were at a shallow angle to the inside surface and the sides of the flaws were partially bonded together, so that they were very difficult to detect by non-destructive techniques. During service the flaws were opened up by crevice corrosion until they were large enough to initiate DHC, Figure 4. Detailed examination of the cracks and evaluation of the manufacturing processes identified the source of the flaws and appropriate changes have been made to the manufacturing process and the inspection procedures.

Decrease in Fracture Toughness of Pressure Tubes

The pressure tubes in a CANDU reactor are operated on the basis of Leak-before-break; i.e., if a growing defect is present in a pressure tube it will grow across the wall and leak sufficiently for it to be detected before reaching the "critical crack length" for unstable crack propagation. Once leak-before-break can no longer be assured, then a pressure tube is considered to have reached its end of life. Many factors have to be considered in such an assessment, including the capability of the leak detection system, rate of leakage, mechanism and rate of crack growth, crack shape and how it may change with crack growth, rate leakage, as well as the critical crack length for axial crack propagation.

Both increasing fluence and deuterium concentration have a detrimental effect on critical crack length and at sufficiently high levels can induce low toughness fracture, especially at low temperatures. Burst tests and small fracture-toughness specimens are used to determine the fracture toughness and critical crack length from room temperature up to operating temperatures. A significant shift in the "fracture transition temperature" to higher temperatures is then one indication of approach to "end-of-life".

In February 1984, the Zircaloy-2 pressure tube was removed from fuel channel G07 in NPD after operating for 134 000 EFPH. Fracture toughness tests on small specimens machined from the region of the tube that had a high fluence and deuterium concentration (about 0.8 at% deuterium) exhibited a brittle-ductile transition temperature of about 220°C, Figure 5. In May 1987, the pressure tube was removed from fuel channel F08 after operating for 156 900 EFPH. Both the fluence and the deuterium concentration (about 2.2 at% deuterium) were higher than the previous tube and the brittle-ductile transition temperature was 280°C, Figure 5. This temperature is above the operating temperature range of 252 to 277°C for NPD pressure tubes. Since leak-before-break could not be assured for pressure tubes exhibiting brittle fracture behaviour in the operating temperature regime (unacceptably short critical crack lengths), a recommendation was made to discontinue operation of the pressure tubes. A shift in transition temperature of about 60°C during the last 3 years of operation indicated that the rate of deterioration had been rapid, mainly due to an accelerating deuterium pick-up rate.

Ingress of Hydrogen Isotopes into Pressure Tubes

Hydrogen has been associated with all the fractures of pressure tubes. The effect requires the concentration of hydrogen isotopes (mostly protium and deuterium) to exceed a critical value at the temperature of interest, so that the embrittling agent, zirconium hydride, can form. The critical value is called the Terminal Solid Solubility (TSS) and the temperature at which hydrides are just formed is called the Solvus Temperature, T_S. The initial hydrogen concentration is specified to be less than 0.18 at%; with this amount of hydrogen T_S is about 220°C and hydride cracking can only happen below this temperature. During operation deuterium is picked up from the corrosion reaction with the coolant and sometimes from the annulus gas; at the rolled joint, crevice corrosion between the end fitting and the pressure tube also makes a contribution. Measurements of deuterium concentration in pressure tubes removed from power reactors suggest that hydrogen pickup accelerates after a certain time but the pickup in Zircaloy-2 is three to ten times greater than in Zr-2.5 Nb. This information was used as part of the decision to retube Pickering Units 1 and 2 and in the recommendation for NPD. These examinations have increased the awareness for improving the information on hydrogen pickup and has led to the development of a tool for sampling tubes without removing them. Also, several suggestions for reducing hydrogen ingress (e.g., controlling material composition) and redistributing it away from highly stressed regions of the tube (e.g., using getters or cooler fins) are being explored for replacement tubes or for new reactors.

Condition of the Inside Surface of Pressure Tubes

Some concern about the effects of fuel bundles wearing on pressure tubes has led to examinations of the inside surfaces. Following video and some destructive examinations, it has been established that crevice corrosion, corrosion of the pressure tube beneath the fuel-bundle bearing pads, is most severe at the primary heat-transport system coolant outlet end of the pressure tube. Half-meter lengths of tube, near the PHTS coolant outlet, are studied for the extent of crevice corrosion and depth of fuel-bundle bearing-pad scratching. To date, the depth of both crevice corrosion and fuel-bundle scratches have not exceeded the corrosion and wear allowances of the design, about 150 μm total. These examinations have primarily been visual and metallographic, but plastic replicas and surface-roughness measurements have also been made.

Examination of Garter Springs

Zr-Nb-Cu garter-spring spacers have been removed from Pickering Units 1 to 4 and Bruce Unit 2 reactors. The springs have been chemically analyzed, examined metallographically and mechanically tested. All the springs had picked up some deuterium from the annulus gas, the Bruce springs less than the Pickering springs, and some springs had solid surface hydride where they were in contact with the cooler calandria tube. In all cases the springs were in good physical and mechanical condition and were well able to perform their duty of maintaining the gap between the pressure tube and calandria tube.

Strength of Calandria Tubes

The calandria tubes are made from annealed Zircaloy-2 sheet, either 1.4 or 1.6 mm thick, that is formed into a tube and then seam welded axially. These tubes are not required to withstand large stresses during normal reactor operation although it would be desirable if the calandria tubes could survive the rupture of a pressure tube. To evaluate the latter situation the strength has been measured by uniaxial and biaxial tensile testing of both irradiated and unirradiated material. The tubes are 30 to 50 MPa stronger in the longitudinal direction than in the circumferential direction. The strength is increased by increases in irradiation (e.g., 180 MPa for a neutron fluence of 6×10^{25} $n.m^{-2}$) and strain rate (e.g., 15 MPa for each decade increase in strain rate) and decreased by increases in temperature (e.g., about 1 MPa per °C). In practice, the calandria tubes are fixed to the ends of the reactors and a biaxial test had to be devised that simulated this situation and could be done in protective cells. The results showed that biaxial strengthening was about 30 to 40% because of the crystallographic texture. This extra strengthening gives sufficient margin for calandria tubes to accommodate full system pressure (e.g., P2 G-16) although they provide no protection against water hammer (e.g., B2 N-06).

Failure of Flux Detectors

The self-powered flux detectors presently in use in most CANDU reactors are mostly Pt or V cored co-axial cables with MgO as insulator and Inconel 600 as outer sheath material. The thinner lead cables carrying the detector signal generally have core wires and outer sheath also made from Inconel 600. The core wire material and detector configuration vary depending on the application but in all cases Inconel 600 is used for the outer sheath. Inconel 600 was chosen for this application because of its excellent corrosion resistance, good cold workability and minimal β-activity following neutron activation. Sheath integrity and core wire continuity must be maintained for successful operation of these detectors. Although Inconel 600 should be resistant to corrosion from acid attack, experience has shown that this cannot always be guaranteed because of surface contamination or sensitization during final heat treatments. Accelerated corrosion in crevices has also been a problem and has been redressed to an extent by having smooth surfaces and by design of the assembly to eliminate joints. Because of acid formation from the radiolysis of moist air or reaction of surface contaminants with a damp environment, all detector assemblies are currently operated in an inert (He) atmosphere. They may be either in a partially encapsulated assembly (open to the moderator) or in a fully encapsulated assembly (in a sealed Zircaloy-2 guide tube). Even in the absence of moisture there has been cable embrittlement and failure due to nitriding in some encapsulated assemblies which have been open to air. Recent failures have occurred because the He purge has been interrupted or because of leaks in assemblies which normally have a static He atmosphere. The failures can be categorized into (a) sheath penetration due to acid attack when the assembly is open to the moderator and (b) wire embrittlement due to nitriding when the assembly is isolated from the moderator but open to air.

Examination of Liquid-Zone Control Units

Liquid-zone control units are some of the vertical reactivity mechanisms in the CANDU core. This mechanism consists of small (< 20 mm), concentric Zircaloy-2 tubes through which He is bubbled to adjust the light-water level in one of two or three zones, thereby altering the reactivity of the core. The smaller inner tubes are contained within a larger (about 100 mm outside diameter) Zircaloy-2 assembly tube.

Examination of one liquid-zone control unit (LZCU) from Bruce NGS Unit 1 was undertaken in 1982, because of a defect at the point where the LZCU passes through the calandria shell, which permitted light water to enter the moderator. After removal from the reactor, it was found that the outer shell of the LZCU had fretted against the bearing insert in the nozzle entering the calandria vessel. Some wear of internal components was also observed, but did not contribute to failure of the unit.

In mid-1986, the LZCU bubbler tubes were removed from Bruce NGS Unit 6 because of problems associated with maintaining light-water levels. The bubbler tubes examined were found to have worn as a result of fretting within the light-water outlet tubes, providing a leak path for the helium. In-situ video examination of the water tubes showed no wear and replacement bubbler tubes were installed. The designs of the Bruce 'A' and Bruce 'B' LZCU's are slightly different. The removal of a LZCU in late 1986 from Bruce NGS Unit 3 provided an opportunity to compare the two designs. No significant wear of the internal tubing of the Bruce 'A' design was found; the differences between the two designs, the configuration of tubes and their supports, are believed to have resulted in different loading and vibration conditions. Based on these examinations, the Bruce 'A' design is preferred, although material substitution has been suggested as remedial action for other designs.

SUMMARY

Many components have been removed from the core of CANDU reactors for examination and testing in the shielded facilities developed at the CRNL and WNRE laboratories. Examples of the components and the solutions that have been developed for various problems are:

DHC at overextended rolled joints between the pressure tube and fitting.	develop low and zero clearance rolled joints that have low end stresses.
Cracking at hydride blisters due to garter springs being out of position.	new garter spring developed that does not move during service.
DHC at manufacturing flaws in pressure tubes.	new specifications for fabrication and inspection.
Wear of bubbler tubes	new design developed.

In addition, from the examination and testing of the components, we are continually obtaining a better understanding of the processes that control the life of the components in the core - corrosion and deuterium ingress, in-reactor deformation and reduction in fracture toughness - which enables us to recommend when components should be replaced before they cause problems or could fail with unacceptable consequences.

ACKNOWLEDGEMENTS

We are grateful to Ontario Hydro for funding and for continued interest by their personnel in all aspects of this work. We would also like to thank the many people at CRNL, WNRE and CO who contributed time and effort to the examinations and helped bring them to successful conclusions.

Table 1 - Core Components Examined After Removal from CANDU Reactors

COMPONENT	COMMENT		NUMBER EXAMINED
PRESSURE TUBES	Zircaloy-2	Material Evaluation and Failure Analysis	19
	Zr-2.5wt%Nb	Material Evaluation and Failure Analysis	82
CALANDRIA TUBES	Material Evaluation and Failure Analysis		3
GARTER SPRINGS	Zr-2.5wt%Nb-0.5wt%Cu	Material Evaluation	~30
	Inconel X750	Material Evaluation	2
END FITTINGS	Examination of liner tubes, bearing rings and rolled joint region		many
LIQUID-ZONE CONTROL UNIT	Examination of helium bubbler tubes, water outlet tubes and centering springs		3
FLUX DETECTORS	Failure Analysis		10

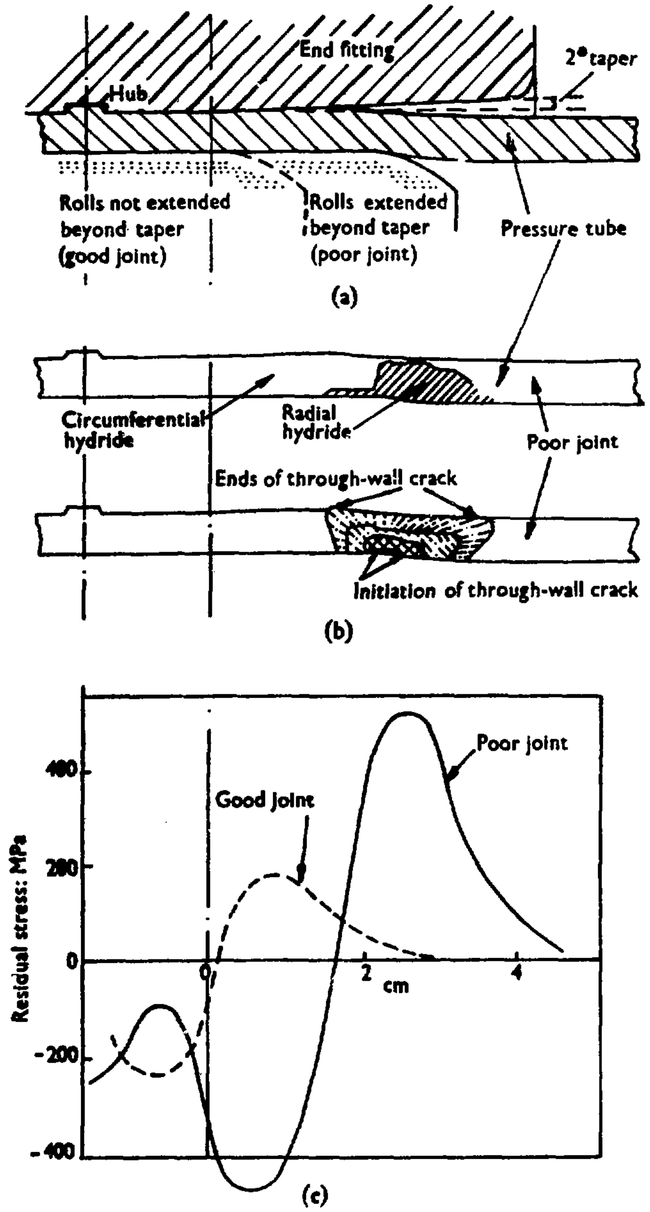

Figure 1: Rolled joints from Pickering 3 and 4: (a) relative position of rolling tool during installation; (b) position of radial hydrides and cracks in a poor joint; (c) residual hoop stress distribution on the inner wall of a good and poor joint.

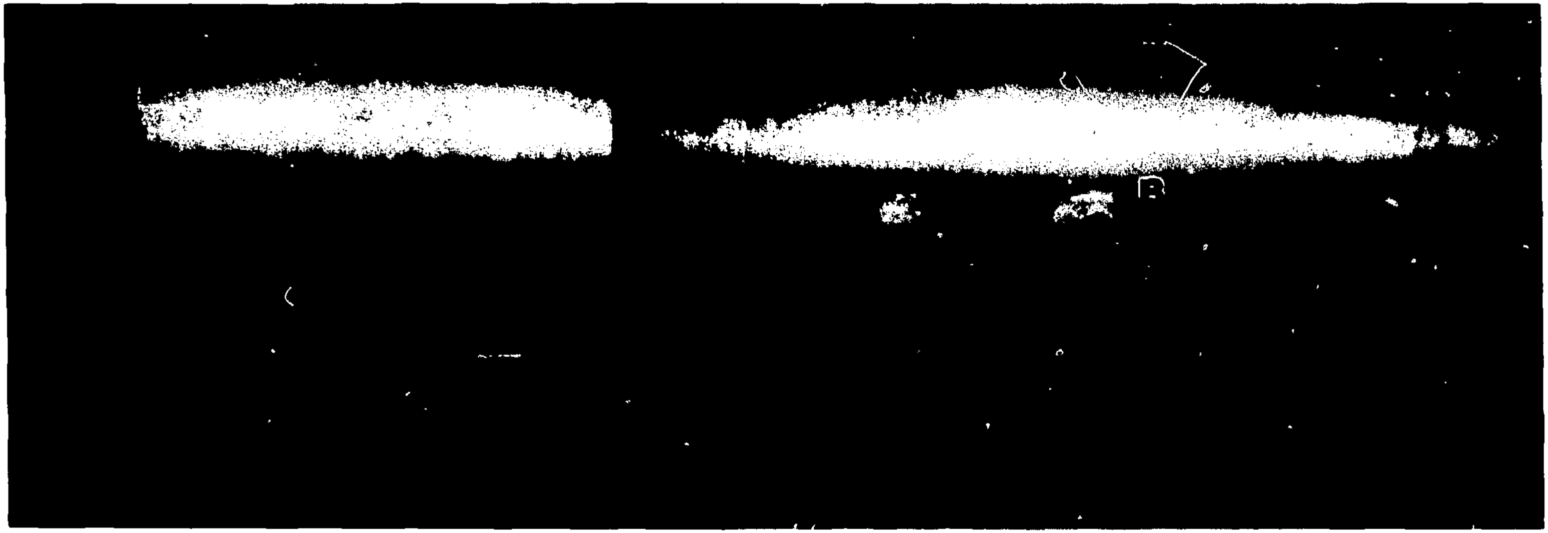

Figure 2: Hydride blisters (B) on the outside surface of the Zircaloy-2 pressure tube that ruptured in Pickering Unit 2 in 1983.

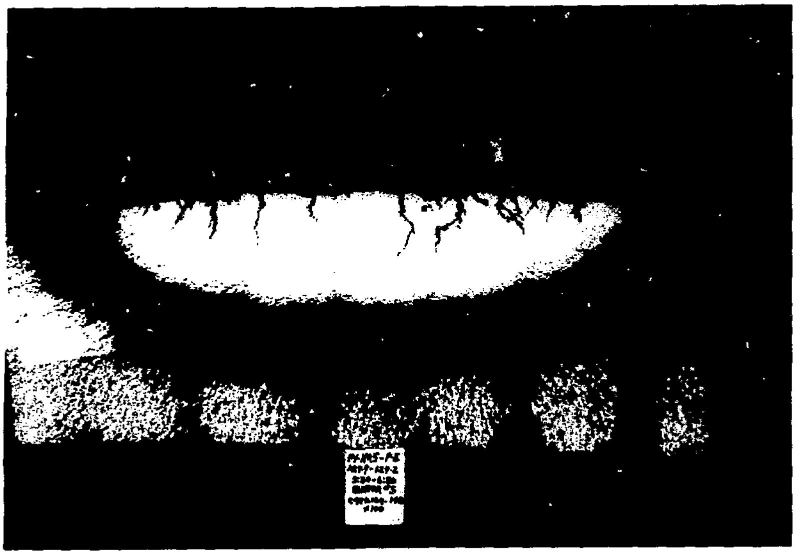

FIGURE 3: Metallographic cross-section through a hydride blister at the outside surface of a Zircaloy-2 pressure tube, removed from Pickering Unit 2 in 1984.

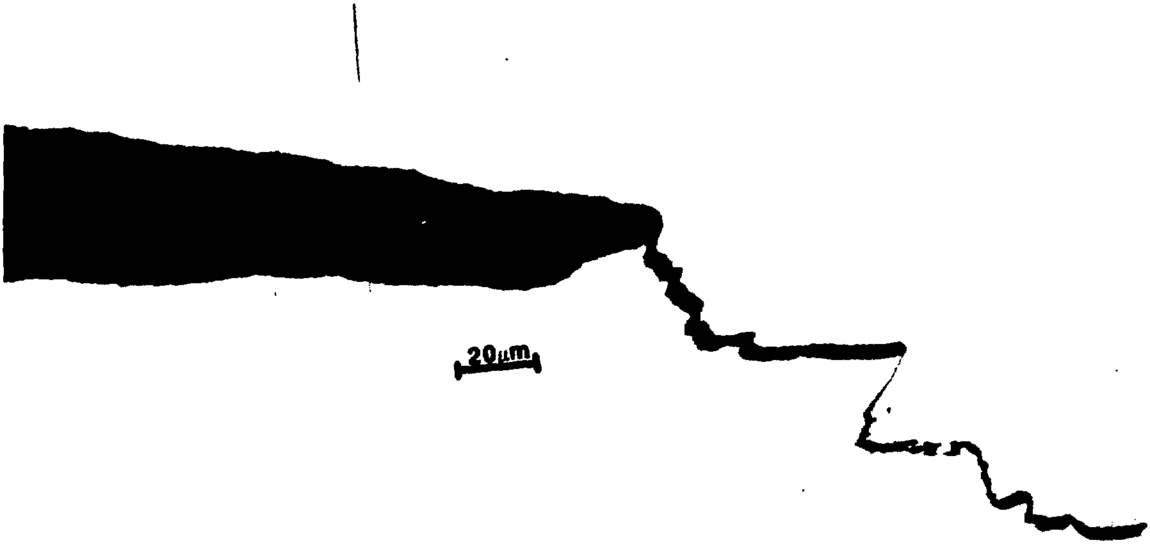

FIGURE 4: Delayed hydride cracking caused by oxide wedging in Zr-2.5 Nb pressure tube. Contrast enhanced by IBAS image analyzing computer.

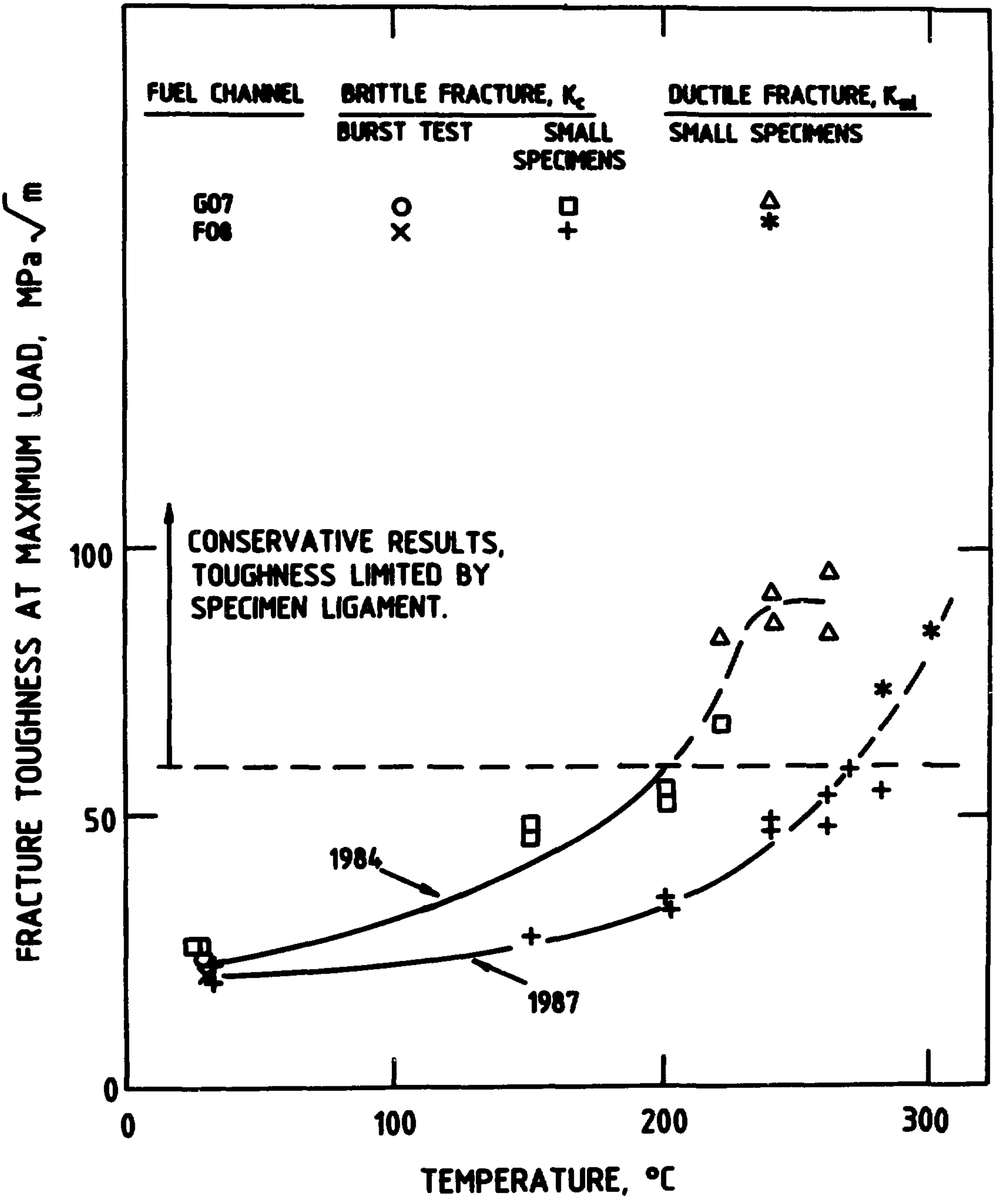

FIGURE 5: Comparison of fracture toughness of Zircaloy-2 pressure tube NPD F08 (1987) with previous results for NPD G07 (1984). Note that small specimens provide conservative results once the failure mode is ductile.

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