SOME ENGINEERING ASPECTS IN THE DEVELOPMENT OF PRESSURE TUBES FOR CANDU

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ABSTRACT

One of the main engineering characteristics of the CANDU nuclear power reactor is the use of pressure tubes rather than one large pressure vessel to contain the fuel and coolant. The power reactor basically consists of a calandria, which is a vessel containing the low pressure heavy water moderator, end shields, and an array of identical pressure tube assemblies which project through the end shields and calandria. The pressure tube is thus the pressure vessel in a CANDU reactor. This paper reviews some of the engineering aspects of the development of the pressure tube. Significant advances have been achieved related to zirconium alloys, tube fabrication, flux enhanced creep, irradiation growth, delayed hydride cracking, inspection and retubing.

INTRODUCTION

When development of nuclear reactors for electric power generation was in its early stages, many possible combinations of fuel, coolant, and moderator were proposed. Only a few reactor types have reached the commercial stage. The Pressurized Water Reactor and the Boiling Water Reactor of the pressure vessel type are the most widely accepted and are the mainstay of nuclear power programs throughout the United States, Europe, and the Far East. The United Kingdom, one of the pioneers of nuclear power, concentrated on the Gas Cooled reactor. Canada stood alone in pursuing the pressure-tube type of reactor. The decision by Canada to proceed on its own certainly left doubts in the minds of many within Canada and abroad as to whether Canada with its limited resources could achieve its goal of economic electric power generation. But when you are second, you try harder; the Canadian program was well focused and well executed. Fifteen years after the decision to follow the heavy water moderated, natural uranium fuelled, pressure-tube type reactor, the Pickering Nuclear Generating Station went into operation and soon demonstrated to the world that CANDU was number one. The Bruce and "CANDU 600" units to follow, further established that CANDU was indeed a great scientific and engineering achievement.

It is gratifying to all those who contributed to the CANDU program that the Engineering Centennial Board recognized the CANDU reactor for production of electricity as one of the ten most outstanding Canadian engineering achievements over the last 100 years.

Within the CANDU program, there were many engineering achievements; the development of the pressure tube is but one. This paper is an attempt to identify some of the key contributions and key events related to the pressure tube.

HISTORY

By the early 1950's, Canada had a good scientific and development base for heavy water moderated and natural uranium fuelled reactors at the Chalk River Nuclear Laboratories, and interest was growing in the development of nuclear electric power. At the same time, Ontario Hydro, Canada's largest electric utility, had seen the need for a diversified power generation program. The basis for a cooperative program between the federal research and development establishment at CRNL and the provincial electric utility which could meet all the challenges from basic science through to economic and reliable operation were set in motion. Contributions from Canadian industry were of equal importance to this emerging program and in 1955 NPD-1 (Nuclear Power Demonstration) was committed as a joint venture of Atomic Energy of Canada Limited, Ontario Hydro and Canadian General Electric Company Limited.

NPD-1 was to be a natural uranium fuelled, heavy water moderated, pressure vessel type of reactor. In the late 1950's, after the NPD-1 project had begun, preliminary work was started on the design concept for a 200 MWe nuclear plant for base load application. A pressure tube reactor was selected. This concept had the advantage of circumventing the size limitation that steel fabrication technology appeared to place on the heavy water moderated pressure vessel type of reactor. NPD was reassessed and was converted to a pressure tube reactor; it became NPD-2, the first CANDU-PHW (Pressurized Heavy Water coolant) to go into service (in 1961).

In 1959, the 208 MWe Douglas Point Generating Station was committed; this prototype power reactor began operation in 1967. The first full-scale commercial CANDU-PHW station of 2056 MWe was committed in 1964 at Pickering. The four units of 514 MWe each went into service between 1971 and 1973. The next major nuclear electric commitment in the Ontario Hydro system was the Bruce Station. Construction started in 1968 and the four 750 MWe units went into service between 1977 and 1979. Between 1982 and 1985 four more units went into service at the Pickering Station. Between 1984 and 1987, four more were added to the Bruce Station, and between 1989 and 1992, four units of 880 service at the Darlington Station. Although the main thrust of the Canadian program has been through a cooperative effort with Ontario Hydro, CANDU's have been and are being built in other countries and in other provinces. New Brunswick and Québec have 'CANDU 600' units, and CANDUs have been built or are being built in India, Pakistan, Argentina, Korea and Romania.

During the 1960's, design and development on two alternatives to the CANDU-PHW were proceeding. First was a CANDU-BLW (Boiling Light Water coolant) concept which had the potential for significant capital cost savings. The program progressed as far as the prototype power reactor of 250 MWe, Gentilly-1 in Québec. It went into operation in 1971. The second alternative was an organic-cooled reactor concept. Coolant temperatures over 400°C had the potential for economic gain through higher steam cycle efficiency. The program progressed as far as testing of fuel and fuel channels in the research reactors at Chalk River and Whiteshell laboratories (CRNL and WNRE).

The pressure tubes in the Pickering, Bruce, Darlington and '600' reactors are for the most part similar. The following briefly describes the CANDU 'pressure tube' reactor.

THE PRESSURE TUBE REACTOR

A CANDU-PHW power reactor basically consists of a calandria, which is a large tank containing the heavy water moderator, end shields, and an array of identical fuel channels which project through the end shields and calandria. Figure 1 is a simplified schematic of a CANDU-PH reactor. Figure 2 shows the general arrangement of a fuel channel. The principal components of a fuel channel are the pressure tube, the calandria tube, the central spacers and the end fittings .

The pressure tubes of about 6 m (240 in.) length, 4.1 mm (0.162 in.) wall thickness, and 103 mm (4.07 in.) inside diameter, contain the fuel and heavy water coolant at a pressure of 10 MPa (1450 psi) and a temperature of 300°C. The pressure tubes are horizontal and are rigidly joined to end-fittings which are firmly supported by the end shields. The calandria tubes are concentric to the pressure tubes and separate the cool moderator (about 70°C) in the calandria from the hot pressure tubes. The central spacer are springs wrapped around the pressure tube to maintain clearance between the pressure tube and calandria tube. The pressure tube reactor is of modular form. Reactors with large power output are obtained by increasing the number of fuel channel modules.

On-power refuelling is facilitated by the pressure tube concept. CANDU reactors are refuelled on power by routinely breaking into the primary heat transport system and at full temperature and pressure via the closure plug in the outboard end of the end-fitting. The reactor bridges and carriages transfer the fuelling machines between the new fuel ports and reactor faces and the spent fuel port. Figure 3 is a view of the end face of a Pickering reactor, its fuelling machine and the bridge assembly.

The pressure tube concept has many advantages. The components of the fuel channel are relatively small. Fabrication processes are repetitive and subject to close control of process parameters. The pressure tubes are of simple shape, thin section and homogeneous. Dimensional tolerances and freedom from defects are obtained by 100% nondestructive testing of the finished tube. Chemical composition, corrosion resistance and strength are guaranteed by destructive tests on pieces cut from the ends of the actual tubes used in a reactor. But there are two disadvantages; first, the components of the fuel channel are exposed to the fast neutron flux produced by the fission of the uranium within the fuel bundles in the pressure tube, and second, only neutron economic materials can be used within the reactor core.

When the decision was made in 1956 to proceed with the 'pressure tube reactor concept', very little was known about the long term effects of fast neutron flux on structural materials. In the ensuing years much of the R&D on pressure tubes was related to the effects of fast neutron flux, but other factors such as end joints and safety considerations also received attention. The advancements in pressure tube technology can be broadly divided into five categories:

1) Tube materials and fabrication
2) Rolled joints
3) Flux enhanced creep and irradiation growth
4) Delayed hydride cracking
5) Inspection and retubing

TUBE MATERIALS AND FABRICATION

Neutron economy is important if the natural uranium fuel cycle is to be successful. The pressure tubes passing through the core of the reactor had to be of neutron economic material and the amount of material had to be kept to a minimum. Of the materials with low neutron capture cross-section which could be used (Al, Mg, Be, Zr, Pb and graphite) only zirconium alloys had adequate strength and corrosion resistance for use at 300°C. Zircaloy-2 was selected as the pressure tube material for NPD. Zircaloy-2, an alloy of Zr-1.5 wt% Sn with small additions of iron, nickel and chromium, had been developed by the US Navy and was extensively used for fuel sheathing and for other core components.

Canada was not alone in appreciating the potential of zirconium alloys as a pressure tube material. The USA, Russia, France, and later the UK, Japan and Italy had pressure tube reactors. There was worldwide interest in the nuclear community in the properties of zirconium alloys. Since cold-working of zirconium alloys increases strength, Roy Thomas and Eric Perryman focussed the Canadian effort on fabrication of cold-worked tubes. The process for fabricating tubes consisted of hot forging ingots to billets, extruding at 750 to 850°C to tubes, cold-drawing about 18% and stress-relieving at 400°C.

Pressure tubes have anisotropic physical and mechanical properties. Zirconium alloys have a close-packed hexagonal crystal structure. During fabrication tubes develop strong crystallographic textures. Considerable effort was expended on determining and understanding the effects of fabrication on tube strength and deformation characteristics in the longitudinal and transverse directions. Since tubes are subject to the internal pressure of the coolant, burst tests where the tube is pressurized until rupture occurs, provided the best practical indication of tube strength. Zircaloy-2 tubes proved to be tough. Irradiation increases strength and decreases ductility, but the irradiated tube remains tough.

Damage to pressure tubes by gouging, fretting or cracking is always a concern, burst tests on tubes with artificial defects machined partially through the tube wall demonstrated the ability of tubes to

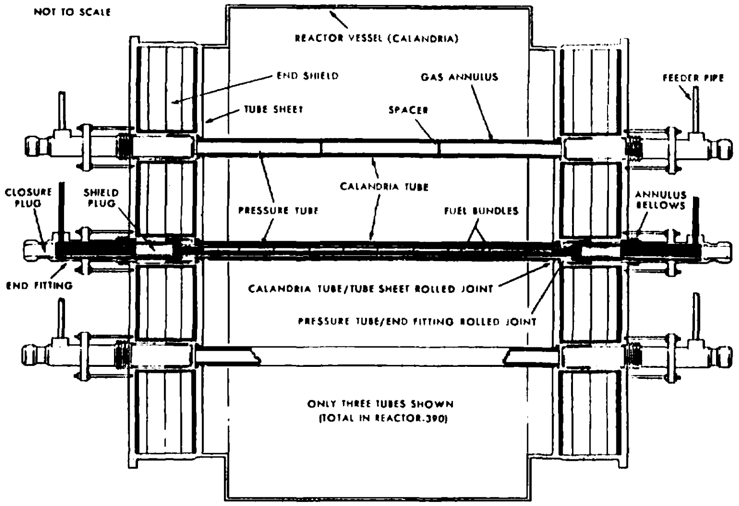

FIGURE 1: SCHEMATIC ARRANGEMENT OF A CANDU-PHW

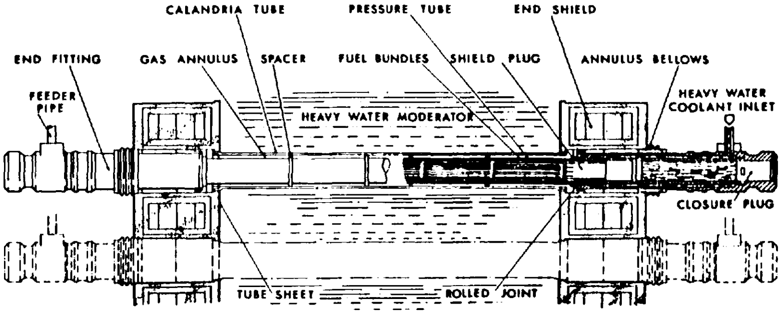

FIGURE 2: SCHEMATIC ARRANGEMENT OF A FUEL CHANNEL

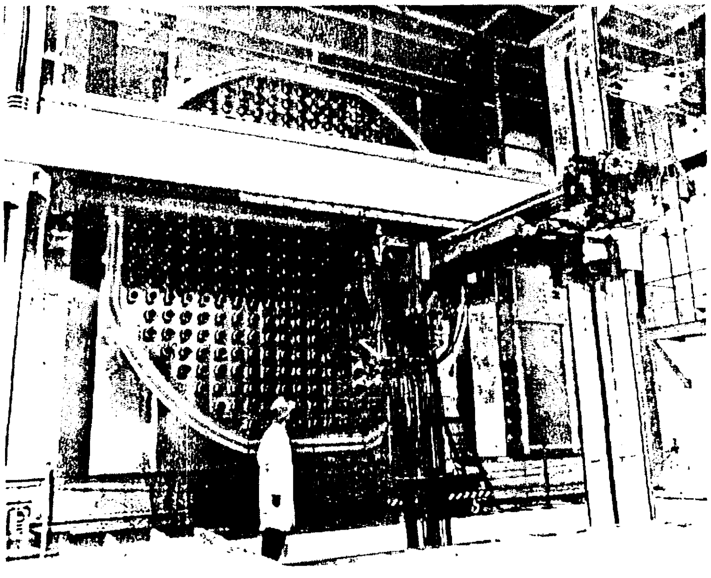

FIGURE 3: PICKERING REACTOR, END FACE AND FUELLING MACHINE

tolerate severe damage. Failure is possible only if tube is damaged or cracked sufficiently to reduce its burst strength to the operating stress of the tube. The tube irradiation program has supplied many highly irradiated tubes which could be artificially defected and burst tested. Burst tests on full-size radioactive components are only possible when the pressure vessel is of reasonable size such as a pressure tube.

Langford extended the program to evaluate through-wall cracks. Artificial defects consisted of thin slits cut longitudinally in the tube wall and then elongated by fatigue cycling to grow cracks at the ends of the slit. The defect represents a crack perpendicular to hoop stress, the maximum tensile stress in a tube. Failure stress decreases with increasing slit length; the slit length which causes unstable crack propogation at the pressure tube design stress is the critical crack length for the alloy condition and test temperature. For current zirconium alloys at 300°C, the critical crack length is about 60 mm at design stress. Hydrogen up to 300 ppm has little effect on critical crack length at 300°C,. The big difference between tube wall thickness (4 mm) and critical crack length (60 mm) gives a high degree of protection from the risk of unexpected tube failure. A growing crack penetrates the wall of a pressure vessel when its length is approximately five times the vessel wall thickness. The pressurized coolant will then leak through a crack, and be detected in the gas annulus between pressure and calandria tubes. Thus, if the critical crack length is more than five times the wall thickness, the pressure vessel will leak but the crack will not propogate unstably. This is the "leak-before-break" criterion developed from fracture mechanics theory. "Leak-before-break" cannot guarantee that tube rupture will not occur but it does add another element of confidence.

Although zirconium alloys have not been included in the ASME Boiler and Pressure Vessel Codes, the pressure tubes for CANDU reactors have been designed to the intent of the Codes. One-third of the ultimate tensile strength for the unirradiated tube is the design criterion. The NPD design stress, based on data available at that time was chosen as 93 MPa (13500 psi) at 280°C, a very conservative value. Based on data produced at CRNL, the design stress for tubes in Douglas Point and Pickering was established as 110 MPa (16000 psi) at 300°C.

The hot coolant inside the pressure tube and moist gas on the outside cause oxidation. Hydrogen release occurs as a result of the oxidation reaction, which results in some hydrogen pick-up in the zirconium alloy. Hydrogen in the metal has little or no effect on the mechanical properties except when concentrations exceed the solubility limit and zirconium hydride platelets form. The platelets form in preferred direction because of tube anisotropy; platelets tend to reduce ductility and impact strength when aligned perpendicular to the direction of principal stresses. The studies concluded that hydrogen was unlikely to be a problem, and a small increase in tube wall thickness was needed to compensate for material loss due to corrosion.

At that time (1960), it was thought that creep strength in-reactor was at least as great as out-reactor. Since out-of-reactor at a stress of one-third of the UTS, creep tests indicated that after 100,000 h creep strain would be considerably less than 1%. (ASME Boiler Code Creep Design Criterion for Ferrous Materials is 1% in 100,000 h), creep was not considered to be either a factor in the design, or life of pressure tubes.

In adopting the pressure tube reactor, test programs were directed at understanding the consequences of a pressure tube rupture. Of particular interest was whether the rupture of one pressure tube could lead to rupture of an adjacent tube and thus propogate through the reactor. Should a pressure tube rupture, the hot pressurized water explosively decompresses discharging coolant and ejecting fuel bundles. Experiments on simulated NPD and Douglas Point reactor arrangements gave confidence that propogation of failures was unlikely; the quantity of hot coolant in a channel is small and the calandria tube around the pressure tube dampens the initial explosive discharge[1].

As good as cold-worked Zircaloy-2 was, there was still an economic incentive to reduce the amount of structural material in the core of the reactor. Two possibilities existed - first was the 'cold-tube' design, and second was a zirconium alloy of higher strength.

In the 'hot-tube' design, the neutron absorbing material consists of the pressure tube, the calandria tube and central spacers. The cold tube design consists of a thin inner liner, an insulant, and a pressure tube in contact with the cool moderator. Cold-tube designs were examined for both pressurized heavy water and organic coolants. Many of the metallurgical problems encountered in the hot-tube design could be by-passed to some extent by using the cold-tube design. The internal insulation protects the pressure tube from the hot coolant, reducing corrosion, hydrogen pick-up and creep of the pressure tube. Neutron economy is improved by virtue of the higher allowable pressure tube design stress at low temperatures. However, new problems were created, the major ones being the choice of a suitable insulant, the proof of the mechanical and metallurgical stability of the liner. Liner stability under thermal cycling was of particular concern.

K.L. Smith played a key role in developing the cold-tube. Full size, prototype cold-tube assemblies were extensively tested and consideration was given to installing assemblies in the research reactor NRU at the Chalk River Nuclear Laboratories (CRNL). Similar progress was made on a cold-tube for use with organic coolant. However, concerns about damage to the liner by fuel movement, liner stability, complexity of the rolled joint and overall assembly, and the consequences of failure, were slowly eroding interest in the 'cold-tube' concept. Furthermore development on new higher strength alloys was progressing well, and the potential for thinner tubes using the hot-tube design looked more promising than pursuing the cold-tube concept. In the mid-1960's, the cold-tube concept was abandoned.

In 1958, the USSR presented information on their research into the properties of zirconium alloys containing niobium, at the 2nd Geneva Conference on the Peaceful Uses of Atomic Energy. The capture cross-section to thermal neutrons was low, the strength was high and the corrosion behaviour at CANDU coolant temperatures was good. Dr. Lewis recognized the potential of this alloy as a pressure tube material, and a considerable amount of the metallurgical R&D at Chalk River was directed at securing Zr-2.5 wt% Nb tubes in a heat-treated form. The incentive was large, heat-treated Zr-2.5 wt% Nb tubes could reduce the tube wall thickness by 30 to 40% relative to cold-worked Zircaloy tubes.

A long intensive development program was thus set into motion. First appropriate material had to be procurred, and small specimens were fabricated to simulate possible tube fabrication routes. By the early 1960's, results indicated that the preferred route for producing pressure tubes was heat-treatment by quenching in water from 880°C, cold-drawing and aging for 24 hours at 550°C in vacuum. W. Evans was given the task of procurring the tubes. It was quickly realized that fabricating hard thin tubes was no easy task.

From a metallurgical viewpoint, a strong heat-treated tube was very desirable. Evans also realized from an engineering viewpoint, a Zr-Nb tube hot-extruded and cold-drawn in a similar manner to Zircaloy-2 tubes, had a lot in its favour, even though the wall-thickness reduction would only be about 20 to 25% relative to Zircaloy-2. Programs to secure both heat-treated (i.e., water-quenched, cold-drawn and aged) and cold-worked (i.e. hot-extruded and cold-drawn) were initiated, and tubes were produced[2].

The tube fabrication program was further complicated by the need for tubes of different wall thicknesses. In the early 1960's, Canada was evaluating three coolants; pressurized heavy water, boiling light water, and organic. The BLW design pressure was about 30% lower than for PHW. For an organic cooled reactor with coolant at 400°C and 2.8 MPa (400 psi), the proposed materials were Ozhennite 0.5 (Zr-0.2 wt%, Sn-0.1 wt%, Fe-0.1 wt%, Ni-0.1 wt% Nb) and at a later date cold-worked Zr Nb. Tubes of different materials and different wall thickness were needed for metallurgical and mechanical testing, for rolled joints and for installation in the research reactors.

Another decision was soon to be made to add to the tube fabrication and evaluation programs. Pressure tubes for NPD and Douglas Point were of 83 mm inside diameter (3.25 in I.D.). Diameter is chosen to meet fuel bundle requirements. For the proposed full-scale reactors of 514 MWe at Pickering, power output could be better achieved by increasing the number of elements in a fuel bundle, i.e., adding another ring of elements. There were arguments for and against the change in size of the fuel bundle. The end result was in favour of a larger fuel bundle and hence a pressure tube with an I.D. of 103 mm (4.07 in.) was specified. The decision on what material and fabrication route was to be specified for the pressure tubes in the Pickering Station was still to be made.

ROLLED JOINTS

Zirconium alloys are expensive and there was little need or desire to extend the pressure tube much beyond the edge of the core of the reactor. Stainless steels were suitable as end fitting material and a Zircaloy-to-steel transition was needed. Welding, brazing or metallurgical bonding of dissimilar metals did not look attractive. Of the various mechanical joints possible, roll-forming became the standard for CANDU pressure tubes. Tubes are rolled into quenched and tempered 12% Cr stainless steel (AISI-403) end fittings.

Figure 4 shows the cross-section of a rolled joint. Three grooves are machined in the end fitting bore in the rolled joint area. The pressure tube is inserted into the end fitting covering the grooves. A tube expander is introduced into the pressure tube through the end fitting. The tube is roll expanded into the end fitting. The tube wall thickness is reduced by 12% to 15% and the grooves in the end fitting are partially filled with tube material. The material in the grooves tends to lock the tube to the end fitting producing satisfactory leak tightness and axial strength. The end fitting material must be hard enough to not deform significantly as the tube is squeezed into the grooves. Simple rolled joints are preferred when the wall thickness to diameter ratio for the tube is from 0.04 to 0.05. When a tube is thin (thickness to diameter ratio less than 0.03), a sandwich rolled joint is preferred. The calandria tube joint as shown in Figure 4 is an example of a sandwich rolled joint. The calandria tube is flared, and a stainless steel ring is inserted. The calandria tube is thus squeezed between the ring and the steel tube sheet by roll forming.

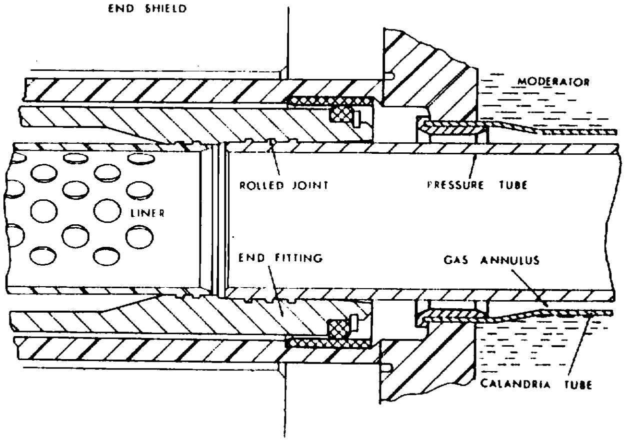

FIGURE 4: CANDU-PHW ROLLED JOINTS

Canadian General Electric and Westinghouse Canada Ltd. conducted most of the development work needed to produce joints for the many variations of pressure tubes under consideration. Experimental and analytic work established a general understanding of how to make a joint and why it works. Zircaloy-2, the softest of the candidate materials had a wall to diameter ratio of 0.05 and as such joints were relatively easy to make. c.w. Zr Nb with a ratio of 0.04 was developed without too much difficulty. Heat -treated Zr-Nb joints were more difficult to make; the material was strong and hard, almost as hard as the end fitting material. Consideration was given to the use of harder stainless steels and Inconel. For a PHW cooled reactor, the wall thickness ratio for a h.t.Zr Nb tube would be about 0.034; too thin for a simple rolled joint and too thick for a sandwich type joint. For BLW and Organic coolant applications, the lower pressure (than for PHW) allowed thinner walls and sandwich-type rolled joints were developed.

FLUX ENHANCED CREEP AND IRRADIATION GROWTH

NRX and NRU are superb reactors for conducting research and development. Its hard to believe that these reactors, which were designed well before Canada made the commitment to develop the pressure tube reactor, would be so valuable to the development of CANDU fuel and pressure tubes. Two types of irradiation facilities are available. First is the in-pile loop which can accommodate a full-length experimental or prototype power reactor pressure tube. Within the tube are fuel bundles and coolant operating at power reactor conditions. Second, there are holes through the reactor into which experimental assemblies can be inserted. The first conclusive evidence on flux enhanced creep were a direct result of two advances in test equipment for use in these irradiation facilities, creep machines and an inspection system.

Interest was growing internationally in the development of creep machines to operate in a reactor environment. In 1958 Orenda Engines Ltd. (later Hawker Siddely Canada) was commissioned to design and build a machine for insertion in an experimental hole in NRX. The first machines were unreliable, and results from NRX and elsewhere were conflicting. It wasn't until 1963 that the first conclusive evidence of flux enhanced creep was secured. A creep test on a specimen cut from an NPD pressure tube was conducted by V. Fidleris. For a period of many months, NRX essentially operated at conditions dictated by the creep experiment. When the reactor was operating, the creep rate was high; when the reactor shutdown, the creep rate was low.

At the same time as the creep machines were being developed, Orenda was asked to build a system for inspecting pressure tubes. Battelle Northwest Laboratories had developed a system for the pressure tubes in the Plutonium Recycle Test Reactor at Hanford. With cooperation from BNWL and building on their experience, Ross-Ross worked with Orenda to produce gouging equipment to measure tube diameter, surface defects and wall thickness[3]. The focus was on establishing the extent of any surface damage which might arise as a result of wear, fretting or gauging of the tube wall by vibration or movement of fuel bundles. The ID measurments were simply to inspect for any unusual deformation resulting from reactor operation. Inspections were done with the reactor shutdown and the pressure tube drained and empty. First in-reactor gaugings were done in an NRU loop tube and two NPD tubes to establish base measurements in 1963.

The second measurement in the NRU loop tube was done in July 1964. The results of the measurements showed a clear relationship between strain and fast neutron flux. The fast flux was linearly related to the fuel power of the bundle in the tube. The strain profile was almost a direct match of the cosine-shaped power (flux) profile from end to end of the section of tube extending through the core of NRU. The maximum strain was only 0.1% (i.e., 0.08 mm or 0.003 in.). As if to add icing to the cake, the strain profile also showed dips of about 0.01% which coincided exactly with the fast neutron flux depressions at the ends of each fuel bundle. The gauging head designed to detect any unusual deformation suddenly became a precision instrument for investigating creep.

Creep was now a potential design problem and the reactor designers were fully informed of the importance of in-reactor creep. The creep program at CRNL was rapidly expanded with the objective of studying mechanisms as well as for securing engineering data for predicting tube behaviour.

The first loop tube was not fabricated to power reactor tube specifications and had been removed to allow installation of one from the production run of c.w. Zircaloy-2 pressure tubes for Douglas Point. Gauging results were of direct interest and confirmed that the circumferential (or transverse) creep rate was almost linearly proportional to the fast neutron flux. Results also showed a small temperature sensitivity. Whereas out-of-reactor tests on c.w. Zircaloy-2 showed a decreasing creep rate with time, the in-reactor results showed no decrease in creep rate over the 11000 hours of operation in NRU. This was very disturbing as very high creep strains were indicated in Douglas Point which was designed for a 30 year lifetime. Although there was great interest in extending test time, the need to get on with development of larger fuel bundles for Pickering, was much greater and the 83 mm Douglas Point tubes were removed and a 103 mm c.w. Zircaloy-2 (Pickering prototype) was installed.

Cold-worked Zr-2.5 wt% Nb tubes became available in 1963 and the first experimental tube (of 83 mm ID) was installed in 1965. Results indicated the creep rate to be about one half that of cold-worked Zircaloy-2 (at comparable conditions) but test time was short (7000 h) and the tube was removed to make way for 103 mm ID tubes of c.w. Zr-Nb. The feasibility of fabricating heat-trated tubes was established in 1965 and 1966. Out-of-reactor corrosion tests showed that increasing cold-work between quenching and aging improved corrosion resistance of Zr-2.5% Nb while out-of-reactor and in-reactor creep tests indicated that creep properties deteriorated. As a compromise between desired corrosion and creep properties, the cold-work between quenching and aging was specified at 10 to 15% for pressure tubes.

Based on uniaxial creep test results in-reactor (no tube results were yet available) heat-treated Zr-2.5% Nb was specified in 1966 for Gentilly-1 (BLW) and KANUPP (PHW in Pakistan). Later h.t.Zr-Nb was adopted by Japan for FUGEN. H.t.Zr-Nb has performed well in these reactors.

The specification of pressure tubes for Pickering was more conservative than for Gentilly and KANUPP and it was decided that tubes for Units 1 and 2 should be cold-drawn Zircaloy 2 with a 1% strain limit, which was expected to be reached in 8 to 15 years. The decision for Pickering Units 3 and 4 was to be made later.

Results were being accumulated on c.w. Zircaloy-2 from NPD and loop tubes, and analysis of data indicated that the sensitivity of creep rate to stress was a linear relation. This was unexpected and in conflict with results from uniaxial stressed specimens from in-reactor creep machines which indicated a much higher stress sensitivity. In 1967, Ross-Ross produced the first design equation for transverse creep of c.w. Zircaloy-2 pressure tubes[4]. It was very simple, -

$$\text{creep rate} = 4 \times 10^{-27} \kappa \sigma \times \phi \times (T - 160) \ h^{-1}$$

where σ was the hoop stress in psi, ϕ was the fast neutron flux in neutrons/cm^{2}s and T was temperature °C (but appliable only from 250 to 290°C).

Starting in 1965, experimental tubes for the NRU loops were machined to reduce the wall thickness over short lengths to create zones of different stress. Results confirmed that in the expression: creep rate varies as σ^n, the stress exponent n was 1 to stress levels well above the range of interest to designers. That n = 1 was of great significance. If 'n' is low, a material is said to behave in a highly plastic manner where the tendency to necking is reduced and exceedingly high ductibility is obtained. The localized irregularities in the strain profile at bundle junctions, and sharp well-defined steps in the strain profile coinciding with stress steps in the machined tubes were indicative of highly plastic deformation.

The strains in pressure tubes as a result of operation in loops are very small relative to the design life of a reactor of 30 years. In 1967 E.F. Ibrahim began a series of tests on small tubular specimens in-reactor at conditions designed to accelerate creep deformation. Specimens have been strained uniformly to over 10%. The design limit was progressively increased as more and more results from Canada and abroad confirmed the highly plastic deformation behaviour. In the early 1970's, the recommended limit for increase in tube diameter was set at 5%. Avoiding high strains by selecting materials more resistant to creep than Zircaloy-2 was now of equal importance as reducing tube wall thickness for neutron economy.

Interest in the metallurgical community in heat-treated Zr-2.5 wt% Nb was high and when reactor grade heat-treated tubes became available, they were immediately fabricated into loop tube assemblies and installed in NRU. In mid-1967, the two cold-worked Zr-Nb tubes in NRU were removed and replaced by heat-treated tubes.

In early 1968, the available information on the three candidate materials (c.w. Zircaloy-2, c.w. Zr-Nb and h.t. Zr-Nb) were analyzed. Of all factors considered, design strength, tube fabrication, rolled joints, and creep were of most importance. Cold-worked Zr-Nb tubes were available well before heat-treated tubes, there was little concern about tube fabrication, and rolled joints had been developed. Concern remained about the fabrication route for heat-treated tubes, and there had not been adequate tube or time to fully prove the joints. Test time for both cold-worked and heat-treated Zr-Nb tubes was very short but the data available indicated that cold-worked Zr-Nb was more creep resistant than heat-treated Zr-Nb and had a transverse creep rate less than half that of c.w. Zircaloy-2. Mostly on the basis of creep behaviour, cold-worked Zr-2.5 wt% Nb was selected for Pickering Units 3 and 4 (and all reactors after Pickering). Design stress was 145 MPa.

Two further aspects of pressure tube performance were being evaluated in the mid to late 1960's, - creep sag and length change.

The ends of the pressure tube are rigidly joined to the end-fittings which are firmly held by the bearings between the end-fitting and end shield. The bearings accomodate thermal expansion. A pressure tube is then a fixed beam subject to a uniformly distributed load from the fuel and coolant and to point loads provided by the central spacers which in turn are supported by the calandria tube. The tube is subject to internal pressure and bending. The problem was one of multi-axial stressing. The problem was complicated by the anisotropic behaviour of zirconium alloys.

Creep results from uniaxially stressed specimens could not be correlated to tube results. Creep rates from material worked in the longitudinal direction were typically greater than rates from specimens cut from sheet or tubes and stressed in the transverse direction. From the meager information available, the indications were that creep behaviour was highly anisotropic.

Stress relaxation is the time-dependent change in stress at constant strain. Creep is the time-dependent change in strain under constant stress. Creep information can be derived from stress-relaxation results. In a beam-type test where the specimen is bent, the stress analysis remains simple as long as the stress distribution remains linear. With the stress exponent 'n' equal to 1, this condition is maintained. A stress-relaxation program was initiated in 1968 to compare and correlate rates from specimens cut from the longitudinal and circumferential directions to tube creep rates, and to serve as a relatively simple test to compare different materials, and metallurgical structures.

From stress-relaxation, uniaxial creep machine and tube results, anisotropic factors were derived using the Von Mises yield criterion generalized for anisotropic materials to describe the multiaxial stress state[5]. Once again, a stress exponent 'n' equal to 1, greatly simplified the analysis of test data. Pettigrew and Ross-Ross applied creep information to a computer analysis to determine tube sag characteristics[6].

The sag analysis showed that the pressure tube quickly sagged to transfer the fuel load through the spacers to the calandria, and the pressure tube would slowly continue to sag down between the spacers and in some cases contact the calandria tube. Stress-relaxation results from Bettis showed that creep also occurred at low temperatures. A creep rate was estimated for the calandria tube and indicated that the calandria tube would sag at a rate of about 1.4 mm per year. For Pickering Units 1 and 2 with cold-worked Zircaloy-2, contact between the pressure tube and calandria tube could occur as soon as 17 years. For the more creep-resistant materials (Zr-Nb) contact would be much later. Sag of the pressure tube between spacers could be easily reduced by increasing the number of spacers.

The anisotropic factors derived to determine deformation under multiaxial stressing indicated that the pressurized tube should also experience length change. A closed ended tube with isotropic creep behaviour should exhibit no creep strain in the longitudinal direction when pressurized. With resistance to creep lowest in the longitudinal direction, the pressure tube should lengthen. Another mechansim which could result in length change was becoming evident, - irradiation growth.

Creep is slow deformation as a result of applied stress. Irradiation growth is deformation due to fast neutron flux with no applied stress. During the late 1960's, interest in irradiation growth was increasing and a number of experimenters in the UK and USA were studying the subject. Early results indicated length increases were usually in the direction of working in cold-worked materials, but results were scattered and the mechanisms were certainly not understood. Tubes in the NRU loops were carefully measured before installation but measurements on 30 m assemblies of active material were for the most part inconclusive. One loop tube and measurements in NPD indicated small length changes.

The Chalk River Laboratories were producing a lot of information on in-reactor creep, and V. Fidleris was in contact with nearly all the contributors to the pool of information accumulating internationally on creep and growth in a reactor environment[7]. His efforts were well rewarded when personnel working on the Hanford N-reactor (with cold-worked Zircaloy-2 tubes) contacted him to report elongations of 25 mm (1 inch) at the centre of a dish-shaped distribution over the face of the reactor. It was conclusive evidence that anisotropic creep and/or growth in the longitudinal direction was significant and was now a design consideration.

The N-reactor has 1004 Zircaloy-2 pressure tubes of 83 mm ID. The tubes had elongated during 12 years of operation in proportion to accumulated channel power output. Tubes of different fabrication methods had elongated different amounts. A cooperative program emerged where Canadian experience and equipment combined with Hanford experience and measurements from a reactor full of highly irradiated specimens resulted in the first good understanding on the relation of creep and growth to fabrication variables [8].

Earlier work at CRNL had already shown in-reactor creep increased with increased cold work. Stress relaxation tests on bent beams, which showed a linear relation between loading stress and deformation, and uniaxial creep test results, which showed an almost linear relation through the tension-compression regime had provided the evidence that for engineering purposes, in-reactor creep and irradiation growth are additive.

The 1970's was a period of consolidation and application. Of most importance, new design equations were formulated for the various materials used in the fuel channel. The equations were becoming more complex. Information from low temperature tests (mostly on calandria tube materials) and from pressure tubes with organic coolant in WR1 at Whiteshell, helped in establishing temperature dependency from room tempeeature to 400°C. Once the extent of deformation was estimated, allowances for deformation could be made. As the reactor construction program progressed, design changes were incorporated in new units such that a creep lifetime of 30 to 40 years would be achieved.

In the early 1970's, there were interests in new alloys with higher strength. Various alloys were examined. Efforts eventually focused on an alloy identified at CRNL as EXCEL (Zr-3.5% Sn, 0.8% Mo, 0.8% Nb). EXCEL offers higher design stress hence thinner tube wall, and better creep resistance than c.w. Zr-Nb. With good creep resistance design complication associated with large allowances for deformation are avoided. Experimental tubes have been produced and loop tube assemblies are currently operating in NRU.

The early reactors, however, were designed and built without knowing there was such a thing as flux enhanced creep and irradiation growth. Douglas Point, Pickering A and the first Bruce A units could not accommodate all of the deformation predicted by the high creep rates. Increase in diameter would mean that the pressure tube would expand and squeeze the central spacer spring between the pressure and calandria tubes. The creep sag analysis indicated that the channels with only two central spacers would sag between spacers and touch the calandria tubes. The operators were forewarned, inspection equipment which could measure changes existed, and reactor behaviour could be assessed.

With significant elongations likely to occur, Causey and Ross-Ross developed simple computer analyses to establish the probable loads on the end stops if and when clearances decreased to zero. Causey and MacEwen later developed a model to simulate the whole reactor[9]. The central high flux channels would take up clearances first and push on the end-shield which would in turn load the calandria tubes which tie the end shields together. This type of analysis was valuable in predicting the consequences of high creep strain and then identifying how best to handle it. There was good coordination and cooperation and such information was freely exchanged between experimenter, designer and operator.

The experimental programs at CRNL and WNRE had produced a great deal of valuable information, but there were aspects that could not be easily obtained from the research reactors. Only the power reactors could provide the long-term information needed to determine the validity of the design equations and analyses of reactor behaviour.

DELAYED HYDRIDE CRACKING

In August 1974, routine checks in Pickering Unit 3 revealed that heavy water was leaking into the system that circulated dry nitrogen gas for corrosion resistance through the annulus between the pressure and calandria tubes. Chemical analysis identified the water as coolant from the primary heat transport system and not moderator water; thus indicating the leak was from the fuel channel. With difficulty, the leak was located and the fuel channel was replaced (i.e., pressure tube, end-fittings and central spacers). Procedures to dry out the gas annulus system were unsuccessful. A continued high collection rate led to the unpleasant conclusion that there must be more leaking tubes.

During August and early September the cause of the leakage was still not known but one of the strong points of the Canadian nuclear power program was already emerging, i.e. the ability to organize itself and react when troubles occur. Before the first channel was removed, the radiation safety, security, transport, and numerous other service groups from Atomic Energy of Canada and Ontario Hydro, who were familiar with the transportaion of highly radioactive components, were being organized. The components were moved by special transport in heavy shielded flasks to the Chalk River Nuclear Laboratories where numerous facilities exist for handling radioactive components.

By mid-September, pressure tests had confirmed the leak to be in the pressure tube near the end fitting at the coolant inlet end of the channel. At this point Canadian industry became deeply involved; a team from Canadian General Electric, who had extensive experience in the design and development of rolled joints for CANDU reactors, was called in to help with the investigation. At first, it was suspected that the rolled joint had loosened and begun to leak. However, after extensive ultrasonic testing and dimensioning of the joint had been completed, the

the stainless steel end fitting was cut away from the tube. Three cracks, about 18 mm long, in the tube were revealed. Dimensional checks on the rolled joint revealed that the rollers used during fabrication must have extended beyond the parallel part of the end fitting producing an "over-extended" joint. The rolling tools used during tube installation were located and tested. The badly worn tool produced extremely high residual tensile stresses. The stresses approached yield strength. By late September the first major clue as to the cause of failure had been found. With the number of leaking channels increasing alarmingly during September and the discovery of cracks, it was realized that the investigation into the cause of the cracks, and the actions required to correct the situation was a job of major proportions. Although cooperation between various groups had been extremely good, a coordinating committee (headed by J. Ingolfsrud of AECL and A. Jackman of Ontario Hydro) was set up early in October to direct the investigation and repair work, set priorities, insure that all resources which could be of value to the investigation were enlisted and avoid duplication of effort. Coordination was through Dunn at AECL Sheridan Park, Ross-Ross at CRNL, Mitchell at Ontario Hydro, Towgood at Pickering and Hunter at CGE. Many contributions were made. Examination of the oxidized crack surface led Dutton, Ambler and Ells to the conclusion that the material had failed as a result of 'delayed hydride cracking', a cracking mechanism that was almost unknown and barely understood.

Metals that form brittle hydrides can fail by 'delayed hydride cracking'. Hydrogen dissolved from hydrides migrate (or diffuse) up a stress gradient and precipitate at a stress concentration (the crack tip) as hydride. If the stress is high enough, the hydride cracks, the crack length is extended, and the sequences is then repeated.

Time is required for hydrogen to migrate to a stress concentration and for the hydride to nucleate, grow and crack. This is the incubation time for the initiation of cracking. The time between the growing steps is usually shorter than the incubation period. The incubation time and the rate of crack growth are dependant on the rate of accumulation of hydrogen which in turn is dependent on diffusion and solubility of hydrogen. The velocity of a growing crack is almost independent of stress distribution at the crack tip (usually defined as the stress intensity factor).

From the above investigations[10], it was concluded that the basic cause of the cracks was the high residual tensile stresses in combination with long periods with the coolant and tubes cold. The hydrogen normally found in pressure tubes (about 10 ppm) migrated to areas of high residual stress on the inner wall where the stress intensity factor at some discontinuity or small defect was high enough to initiate cracking. Crack propagation was by fracture of hydrides which are brittle when cold. The cracks progressed outward as far as the compressive zone under the hub of the rolled joint, and inward as far as the zone of zero residual stress in the pressure tube. Once initiated and of a size greater than the size of the incipient cracks found on the inner wall, the cracks proceeded through the tube wall by the repeated formation and fracture of the hydrides at the tip of the crack when the heat transport system was cold. When the system was hot, the hydrogen was in solution and crack growth did not proceed. The growth of the crack growth was directly related to the reactor operating history. It was was concluded that all cracks had originated early in the life of the reactor before significant stress relaxation had occurred. Cracking may have initiated during commissioning when the residual stresses were highest and the tubes were subject to additional stresses during pressure and thermal cycles.

The rolled joints in Pickering Unit 4 had been made using the same tools and procedures as Unit 3. In May 1975, water was found in the gas annulus system indicating Unit 4 also had cracks. Pressure tubes, end-fittings and central spacers were removed and new components installed in 17 fuel channels in Unit 3 and 52 in Unit 4. Only a few channels had through-wall cracks, most tubes removed had cracks partially through the wall. These cracks were located by NDT technique. Most of the components removed from Units 3 and 4 were shipped to CRNL for examination, testing and disposal.

During the rolling operations in Pickering Units 1 to 4 and Bruce 1 and 2, the danger of over-extension had never been foreseen and the rolling procedure did not specifically prevent it happening. Current procedures exercise careful controls on the rolling tool and its location. Units 1 and 2 have Zircaloy-2 tubes. Zircaloy-2 is also susceptible to delayed hydride cracking but Zircaloy-2 has a lower yield strength and hence lower residual stresses after rolling. No trouble was experienced in Units 1 and 2. In 1985, one more c.w. Zr-Nb tube had a through-wall crack (probably initiated at the same time as the crack investigated in 1974) which leaked, and the tube was replaced.

The c.w. Zr-2.5% Nb tubes in Bruce Units 1 and 2 had also been installed in 1972 and 73 with over-extended joints, but the joints had not yet been taken up to temperature. Inspection for cracks using an eddy-current type of probe in 1974 indicated no cracks had initiated. The residual stresses were reduced by stress-relieving the over-extended region of the rolled joint by insertion of a special heating coil in early 1975. In 1982 and 83, three tubes in Bruce 2 were removed after leaks were detected. Tests on small specimens had demonstrated very long incubation periods at room temperature before cracking is initiated. It is believed a few incipient cracks may have existed but had gone undetected at the time of the eddy-current inspection or originated after the inspection, but before the stress-relieving[11].

The tubes in Pickering and Bruce with through-wall cracks had leaked, been detected and then removed, thus supporting the 'leak-before-break' concept.

Although the cracking as experienced in Pickering Units 3 and 4 could easily be avoided by proper rolling, the experience had much more important implications. A mechanism had emerged which could cause zirconium alloy components to fail. Delayed hydride cracking received considerable attention at Ontario Hydro, CRNL and WNRE and results of many studies were published, e.g., [12][13].

In August 1983, a cold-worked Zircaloy-2 pressure tube in Pickering Unit 2 ruptured. Coolant was being lost (subsequently estimated at a rate of 17 litres per second) from the primary heat transport system. The reactor was shutdown using normal procedures. No safety systems were activated.

Once again a coordinated effort by Ontario Hydro and AECL was initiated to investigate the cause of rupture. The conclusions of the investigation identified two significant factors - first, the hydrogen content in the tube was higher than expected and second, one central spacer was well out of position[14]. As a result of corrosion, the Zircaloy-2 pressure tube had absorbed much more hydrogen (as deuterium) than predicted. With a central spacer out of position, the pressure tube sagged (by creep) and had contacted the calandria tube over a long length. Upon contact a steep temperature gradient was created. Hydrogen in the tube migrated down the temperature gradient. Hydride 'blisters' formed at the cool points of the pressure tube. Cracks originated in the 'blisters of hard brittle hydride, and grew by the delayed hydride cracking mechanism. A series of cracks formed intermittently along the contact line but not through the wall. The series of growing cracks produced a fault greater than critical crack length and at some point the remaining web failed and the crack ran to produce a 2 m long split. The discharging coolant did not rupture the calandria tube but leaked through the bearing at the end-fittings. The defect did not 'leak (and be detected) before break'. Cheadle, Field, Dunn and Price led the investigations and an excellent series of papers dealing with all aspects of the failure and delayed hydride cracking were presented at a CIMM Conference in Québec City in 1985[15].

The pressure tube rupture created new implications. Another form of delayed hydride cracking had been found; high hydrogen content along with a high temperature gradient is a bad combination. Although the central spacer being out of position allowed premature contact, channels with only two spacers (Pickering 1, 2, 3 and 4 and Bruce 1 and 2) will eventually experience contact. Contact should not occur in the later CANDU's with four correctly located spacers.

A cold-worked Zr-2.5% Nb tube which had sagged and had been in contact with its calandria tube showed no evidence of hydrogen accumulation[14]. Hydrogen concentration was low indicating Zr-Nb may not be as susceptible to 'blister' formation as Zircaloy-2. Cold-worked Zr-2.5% Nb has performed well, and a good understanding of the delayed hydride cracking mechanism has been gained. Continuing R&D along with periodic tube inspections and removal of pressure tubes for testing and destructive examinations should insure that good performance will be maintained.

TUBE INSPECTION AND RETUBING

The first inspection system for 83 mm tubes was developed by Koehler and Ross-Ross and was used in NRU and NPD in 1963. The basic techniques for inspecting tubes had been established[3]. Surface defects, diameter and straightness were measured and viewing was done using boroscopes. The system was portable and used in pressure tubes in WR1, Douglas Point, KANUPP, RAPP, PRTR and N-reactor. With the advent of the Pickering Units, Ontario Hydro needed equipment. The first Ontario Hydro inspection system was an improved version of the CRNL system. These inspection systems were only used during periods with the reactor shutdown and the tube drained and empty. Early concern about damage to tubes by the fuel proved unfounded, and inspection programs redirected with emphasis on securing creep information.

In 1974 an ultrasonic technique was introduced to look for cracks in the rolled joints in Pickering Unit 3. A later development allowed installation of an ultrasonic device into a channel without draining or defuelling, by loading the equipment into a fuelling machine. The fuelling machine removed the closure and shield plugs in the selected channel and inserted the inspection device and a special seal plug.

With the ultrasonic specialists involved with Pickering cracks, the eddy-current specialists at CRNL developed devices for inspecting the rolled joints in Bruce. Eddy-current devices were later developed to measure the gap between pressure and calandria tubes and to locate the central spacers [16]. TV viewing replaced the cumbersome boroscopes.

Ontario Hydro developed a new system in the early 1980's for in-channel inspection; equipment was mounted on the fuelling machine bridge[17]. After installation and set-up, the system was controlled (mostly by computer) from outside the reactor vault.

The original Ontario Hydro program for gauging power reactor pressure tubes was primarily aimed at predicting when tubes had to be replaced. Thus only infrequent (about every 5 years) were proposed to satisfy operating and licensing requirements; much to the chagrin of those at CRNL and Sheridan Park who looked forward to long term creep data. During the 70's, creep and growth prediction and the cracking of Pickering tubes accelerated Ontario Hydro's interest in gathering data. Pickering and Bruce inspections are now producing some of the best long term data available on diameter, length change and sag. Results generally confirm the early predictions and analyses. Operators are now able to tailor the operations to secure maximum creep life from the fuel channels.

In most instances, advancements in the reliability and usefulness of inspection devices has gone hand in hand with post-removal examinations of pressure tubes. In the early days, inspection data was checked by post-removal dimensioning and replicas of surface damage. More recently, interpretations of crack characteristics by NDT have been judged against destructive examinations of real cracks and blisters removed from reactor tubes.

Whereas end of life of a pressure tube due to dimensional change is relatively straight forward, future developments in NDT, aimed at detecting any changes in metallurgical characteristics such as corrosion, hydrogen content, effects of neutron bombardment, will definitely benefit from the tube surveillance program which provides for periodic removal of tubes for examination.

The life of a pressure tube is uncertain but is considered to be limited by creep deformation and could range from 20 to 40 years depending on details of each reactor design. Current reactors were designed for 30 year lifetime.

Retubing can be done. Over 70 tubes have been installed, irradiated and removed from the Canadian test reactors NRU, NRX, WR1 and from NPD. In 1968, one pressure tube and one calandria tube were replaced in the Douglas Point reactor because of damage to the calandria tube caused by faulty installation of a booster rod. In 1974 and 75, 17 and 52 tubes were removed from Pickering Units 3 and 4 because of cracks. Dunn and Murdoch played a lead role in design and development of improved systems for changing fuel channel components. Shielding cabinets were placed on the fuelling machine platforms to reduce the radiation fields in the work area. The bridge system moved workers and tools to the desired location. The operations at Pickering under Towgood went extremely well. Unit 3 was returned to service in 8 months and Unit 4 in 10 months; it was an amazing engineering achievement when it is realized that the basic cause, delayed hydride cracking, was virtually unknown in the beginning. The whole program was a fine example of the effectiveness of good teamwork.

During the early days of the Pickering investigation, when the number of cracked tubes was increasing alarmingly, preliminary plans were formulated for completely retubing a reactor. Fortunately, such steps were not needed. Procedures for single tube removal were well established and pressure tubes have been removed from NPD, Pickering and Bruce either because of damage or cracks or as part of a surveillance program.

In the late 1970's, it was becoming evident that retubing of the early reactors, i.e., Pickering Units 1 to 4 and Bruce Units 1 to 3, which were designed and constructed before the consequences of creep and growth were fully understood, would eventually be required. Ontario Hydro with assistance from SPAR Aerospace, AECL, CGE and other Canadian industries developed plans for Large Scale Fuel Channel Replacement[18]. The basic system assumed reasonably high radiation fields would exist in the reactor vault. A large shielding cabinet mounted on the fuelling machine bridge assembly would house turrets of tool packages that could be operated remotely from a control centre outside the vault.

With the rupture of a Zircaloy-2 tube in Pickering Unit 2 and knowing that there was the potential for other tubes to rupture in a similar manner, and knowing that tube life was limited by creep and growth, the future of these units was carefully assessed. Units 1 and 2 had been down for some time and radiation fields were low. Extended life related to tube elongation could be achieved by jacking the fuel channel assembly to an outboard position which would allow maximum use of the bearings on the end fittings, but this operation involved a fair amount of shutdown time. In March 1984, the decision was made to proceed with retubing of Pickering Units 1 and 2 using tools and schemes that were simpler than those based on the original large scale concept using mostly remotely operated equipment.

SUMMARY

The decision in 1956 to base the CANDU power reactor program on the pressure tube type was a bold one for little was known at that time about the performance of zirconium alloys as a structural material. It was a wise one because small is beautiful, and the R&D facilities at CRNL and WNRE were well equipped to handle the relatively small pressure vessel - the pressure tube. With world class facilities, it did not take long to develop world class expertise, and Canada was soon a leader in zirconium technology. Canada contributed to and benefitted from the pool of information being developed internationally.

Unexpected problems occur in any program as extensive as CANDU nuclear power; the capability to identify and solve problems quickly was essential to the program. The coordinated program conducted by experts representing operator, fabricator, designer and developer, effectively found engineering solutions to the crack and creep problems. Such problems can be avoided by design and installation procedures. The experience from one reactor or incident is beneficial to the next and the fuel channel design and safety criteria were periodically reviewed and improved. Future reactors will be designed with careful attention to fuel channel removal and replacement.

The fuel channel module and the fuelling machine bridge concept helped to make the horizontal pressure tube type of reactor easy to inspect and maintain. Development of inspection and maintenance devices also benefitted from the small size and simple shape of the pressure tube.

Cold-worked Zr-2.5 wt% Nb has performed well; the faults experienced by tubes were not due to material deficiencies. Much has been learned about the detrimental effects of hydrogen, and although hydrogen pick-up has been low, continued efforts are being expended in an attempt to foresee any deficiency that might arise as a result of long exposure of tubes to a reactor environment. Periodic inspection and periodic removal of pressure tubes from the Pickering and Bruce reactors should be of great value in detecting any change that could indicate a metallurgical limit to the life of a tube. Although cold- worked Zr-2.5 wt% Nb has performed well, there is interest in new alloys. Tubes of zirconium with additions of tin, molybdenum and niobium which have higher strength and better creep resistance, are now operating in the research reactors.

CANDU has done much to demonstrate Canadian scientific and engineering capability. The coordinated effort of a government funded R&D establishment, Canadian industry and the ultimate user of the technology has proven to be extremely effective; CANDU is also an economic success. The proving of the pressure tube is but an example of how the various R&D teams were effective in developing their own programs and drawing on the resources of the world to achieve their goals. The CANDU program was mission oriented, well focused and very productive. One can only hope that the government agencies responsible for R&D in Canada, and Canadian industry who stand to enjoy the benefits arising from good R&D, will in the years to come, learn from the CANDU experience. Perhaps Canada with its limited resources can continue to produce winners.

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