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CANDU pressure tubes

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FOREWORD

At present, there are over four hundred operational nuclear power plants (NPPs) in IAEA Member States. Operating experience has shown that ineffective control of the ageing degradation of the major NPP components (e.g. caused by unanticipated phenomena and by operating, maintenance, design or manufacturing errors), can jeopardize plant safety and also plant life. Ageing in NPPs must be, therefore, effectively managed to ensure the availability of design functions throughout the plant service life. From the safety perspective, this means controlling within acceptable limits the ageing degradation and wearout of plant components important to safety so that adequate safety margins remain, i.e. integrity and functional capability in excess of normal operating requirements.

This TECDOC is one in a series of reports on the assessment and management of ageing of the major NPP components important to safety. The reports are based on experience and practices of NPP operators, regulators, designers, manufacturers, and technical support organizations and on a widely accepted Methodology for the Management of Ageing of Nuclear Power Plant Components Important to Safety which was issued by the IAEA in 1992. They have been compiled using contributions from technical experts in various countries, feedback from a September 1994 Technical Committee meeting which was attended by 53 technical experts from 21 Member States (who reviewed first drafts in specialized working groups), and review comments from invited specialists.

The current practices for the assessment of safety margins (fitness-for-service) and the inspection, monitoring and mitigation of ageing degradation of selected components of Canada deuterium–uranium (CANDU) reactor, boiling water reactor (BWR), pressurized water reactor (PWR), and water moderated, water cooled energy reactor (WWER) plants are documented in the reports. These practices are intended to help all involved directly and indirectly in ensuring the safe operation of NPPs; and also to provide a common technical basis for dialogue between plant operators and regulators when dealing with age related licensing issues. Since the reports are written from a safety perspective, they do not address life or life-cycle management of the plant components, which involves the integration of ageing management and economic planning. The target audience of the reports consists of technical experts from NPPs and from regulatory, plant design, manufacturing and technical support organizations dealing with specific plant components addressed in the reports.

The component addressed in the present report is CANDU pressure tubes. The work of all contributors to the drafting and review of this report is greatly appreciated. In particular, the IAEA would like to acknowledge the contributions of W.R. Clendening of Atomic Energy of Canada Ltd (AECL) who drafted this TECDOC and P. Kumar (Nuclear Power Corporation, India) who provided information on Indian pressurized heavy water tube reactors (PHWTRs). B.A. Shalaby (AECL) and J. Pachner of the IAEA's Division of Nuclear Installation Safety directed the preparation of this publication.

EDITORIAL NOTE

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1. INTRODUCTION

1.1. BACKGROUND

Managing the safety aspects of nuclear power plant (NPP) ageing requires implementation of effective programmes for the timely detection and mitigation of ageing degradation of plant systems, structures and components (SSCs) important to safety, to ensure their integrity and functional capability throughout a plant's service life. General guidance on NPP activities relevant to the management of ageing (maintenance, testing, examination and inspection of SSCs) is given in the IAEA Nuclear Safety Standards (NUSS) Code on the Safety of Nuclear Power Plants: Operation (Safety Series No. 50-C-O, Rev. 1) and associated Safety Guides on in-service inspection (50-SG-O2), maintenance (50-SG-O7, Rev. 1) and surveillance (50-SG-O8, Rev. 1).

The Operation Code requires that NPP operating organizations prepare and carry out a programme of periodic maintenance, testing, examination and inspection of plant systems, structures and components important to safety to ensure that their level of reliability and effectiveness remains in accord with the design assumptions and intent, and that the safety status of the plant has not been adversely affected since the commencement of operation. This programme is to take into account the operational limits and conditions, any other applicable regulatory requirements, and be re-evaluated in the light of operating experience. The associated Safety Guides provide further guidance on NPP programmes and activities that contribute to timely detection and mitigation of ageing degradation of SSCs important to safety.

The Safety Guide on In-Service Inspection (ISI) provides recommendations on methods, frequency and administrative measures for the ISI programme for critical systems and components of the primary reactor coolant system aimed at detecting possible deterioration due to the influences of stress, temperature, irradiation, etc., and at determining whether they are acceptable for continued safe operation of the plant, or whether remedial measures are needed. Organizational and procedural aspects of establishing and implementing a NPP programme of preventive and remedial maintenance to achieve design performance throughout the operational life of the plant are covered in the Maintenance Safety Guide. For SSCs important to safety, guidance and recommendations on surveillance activities (i.e., monitoring plant parameters and systems status, checking and calibrating instrumentation, testing and inspecting SSCs and evaluating the results of these activities) are provided in the Surveillance Safety Guide. The aim of the surveillance activities is to verify that the plant is operated within the prescribed operational limits and conditions, to detect in time any deterioration of SSCs, as well as any adverse trend that could lead to an unsafe condition, and to supply data to be used for assessing the residual life of SSCs. The above Safety Guides provide general programmatic guidance, but do not give detailed technical advice for particular SSCs.

Ageing management specific guidance is given in IAEA Technical Reports Series No. 338 "Methodology for the Management of Ageing of Nuclear Power Plant Components Important to Safety" and in Safety Practice No. 50-P-3 "Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing". Guidance provided in these reports served as a basis for the development of component specific technical documents (TECDOCs) on the Assessment and Management of Ageing of Major NPP Components Important to Safety. This publication on CANDU pressure tubes is one of such TECDOCs.

TABLE I. CANDU REACTORS

CANDU unit	No of fuel channels	Power output MW(e)	In-service date
Pickering 1*	390	515	1971
Pickering 2*	390	515	1971
Pickering 3*	390	515	1972
Pickering 4*	390	515	1973
Bruce 1	480	850	1977
Bruce 2	480	850	1977
Bruce 3	480	850	1978
Bruce 4	480	850	1979
Pickering 6	380	515	1982
Point Lepreau, New Brunswick	380	640	1983
Gentilly, Quebec	380	638	1983
Wolsong 1, Rep of Korea	380	629	1983
Embalse, Argentina	380	600	1984
Pickering 5	380	515	1984
Bruce 6	480	850	1984
Pickering 7	380	515	1985
Bruce 5	480	850	1985
Pickering 8	380	515	1986
Bruce 7	480	850	1986
Bruce 8	480	850	1987
Darlington 2	480	850	1990
Darlington 1	480	850	1992
Darlington 3	480	850	1992
Darlington 4	480	850	1993
Cernavoda 1, Romania	380	625	1996
Wolsong 2, Rep of Korea	380	650	1997

*The initial pressure tubes in these units have been replaced

Thin wall zirconium pressure tubes are one of the major distinguishing features of pressurized heavy water reactors operating in Argentina, Canada, India, Pakistan, the Republic of Korea and Romania. Each reactor has a few hundred zirconium alloy pressure tubes that support and locate the nuclear fuel in the reactor. The reactor primary coolant passes through these tubes and extracts heat from the fuel. Therefore, pressure tubes are required to operate in a severe environment (high temperature, pressure and neutron flux) with a high level of reliability throughout a 30 year design life.

Since the initial Canada deuterium–uranium (CANDU) fuel channel was designed and installed in the Nuclear Power Demonstration (NPD) prototype reactor that started to operate in Canada in 1962, over 10 000 pressure tubes have operated in various CANDU reactors, as indicated in Table I. Many additional pressure tubes have also operated in Indian pressurized heavy water tube reactors (PHWTRs), as indicated in Table II. This specialized application of pressure vessel design allows CANDU units to achieve very high capacity factors, because they can be refuelled while operating at full power. On-power refuelling is performed by two fuelling machines that can attach to end fittings at each end of a fuel channel. The end fittings are the out-of-core extensions of pressure tubes, as illustrated in Figure 1.

Because extensive effort has gone into the design and development of CANDU pressure tubes, and their fabrication/installation is subjected to extensive quality control, their overall performance has been good. However, some problems were encountered. The feedback obtained from their operation and a comprehensive CANDU pressure tube research and development program, have helped define the degradation concerns and the management of their ageing [1–4]. Evolutionary design improvements have also been defined for recent and future fuel channels [5].

Most instances of cracking in pressure tubes leaked long before the critical crack size was reached and the reactor was safely shut down. Hence the tubes exhibited a leak before break behaviour. The CANDU annulus gas system detects small pressure tube leaks, and procedures are in place to ensure the reactor is shut down before a crack grows to the critical length. Nevertheless, the safety analysis assumes a spontaneous rupture of a pressure tube, and the reactor has been designed to cope with the failure of a pressure tube that might occur while the reactor is at power. This capability was demonstrated when a tube ruptured in 1983 while Pickering unit 2 was operating, and again in 1986 when a tube ruptured during a cold pressurization test at Bruce unit 2.

Although the CANDU reactor design has the capability to withstand the consequences of a pressure tube rupture, designers and operators of these plants must strive to reduce the possibility of pressure tube failure. The resulting unplanned shutdown to repair damage associated with the rupture, to determine and rectify the cause of the rupture and to assess its implications for other units is a large economic penalty. In addition, if such events became frequent, this would increase the risk of them escalating to a more serious accident. Therefore, it is essential that a large safety factor be associated with the possibility of a pressure tube rupture, to ensure reactor safety and reliable operation. Any degradation mechanism which impairs pressure tubes is a significant concern. There is always some risk associated with operation of a pressurized component. This TECDOC describes the various approaches, practices and requirements used for the pressure tubes of CANDU reactors to obtain a risk of acceptable level.

1.2. OBJECTIVE

The objective of this report is to document the current practices for the assessment and management of the ageing of the pressure tubes in CANDU reactors and Indian PHWTRs.

The underlying objective of this report series is to ensure that information on the current assessment methods and ageing management techniques is available to all involved, directly and indirectly, in the operation of nuclear power plants in the IAEA Member States. The target audience includes nuclear power plant operators, regulators, technical support organizations, designers, and manufacturers.

TABLE II FUEL CHANNEL DATA FOR INDIAN REACTORS

Indian units	RAPS-1	RAPS 2	MAPS-1	MAPS 2	NAPS-1	NAPS-2	KAPS-1	KAPS 2
No of channels	306	306	306	306	306	306	306	306
Annulus gas	Air	Air	Air	Air	CO_2	CO_2	CO_2	CO_2
Annulus spacer material	ZrNbCu	ZrNbCu	ZrNbCu	ZrNbCu	ZrNbCu	ZrNbCu	ZrNbCu	ZrNbCu
No of spacers per channel	2	2	2	2	4	4	4	4
Loose/tight fitting spacers	Loose	Loose	Loose	Loose	Loose	Loose	Loose	Tight
Spacer repositioning done after hot commissioning	No	No	No	No	Yes	Yes	Yes	-
Pressure tube material	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2	Zr 2	Zr-2 5%Nb
Wall thickness (mm)	4 03	4 03	4 03	4 03	4 03	4 03	4 03	3 20
Internal dia (mm)	82 5	82 5	82 5	82 5	82 5	82 5	82 5	82 5
Power output (MW)	100	200	220	220	220	220	220	220
In-service date	Dec 73	Apr 81	Jan 84	Mar 86	Jan 91	Jul 92	May 93	-

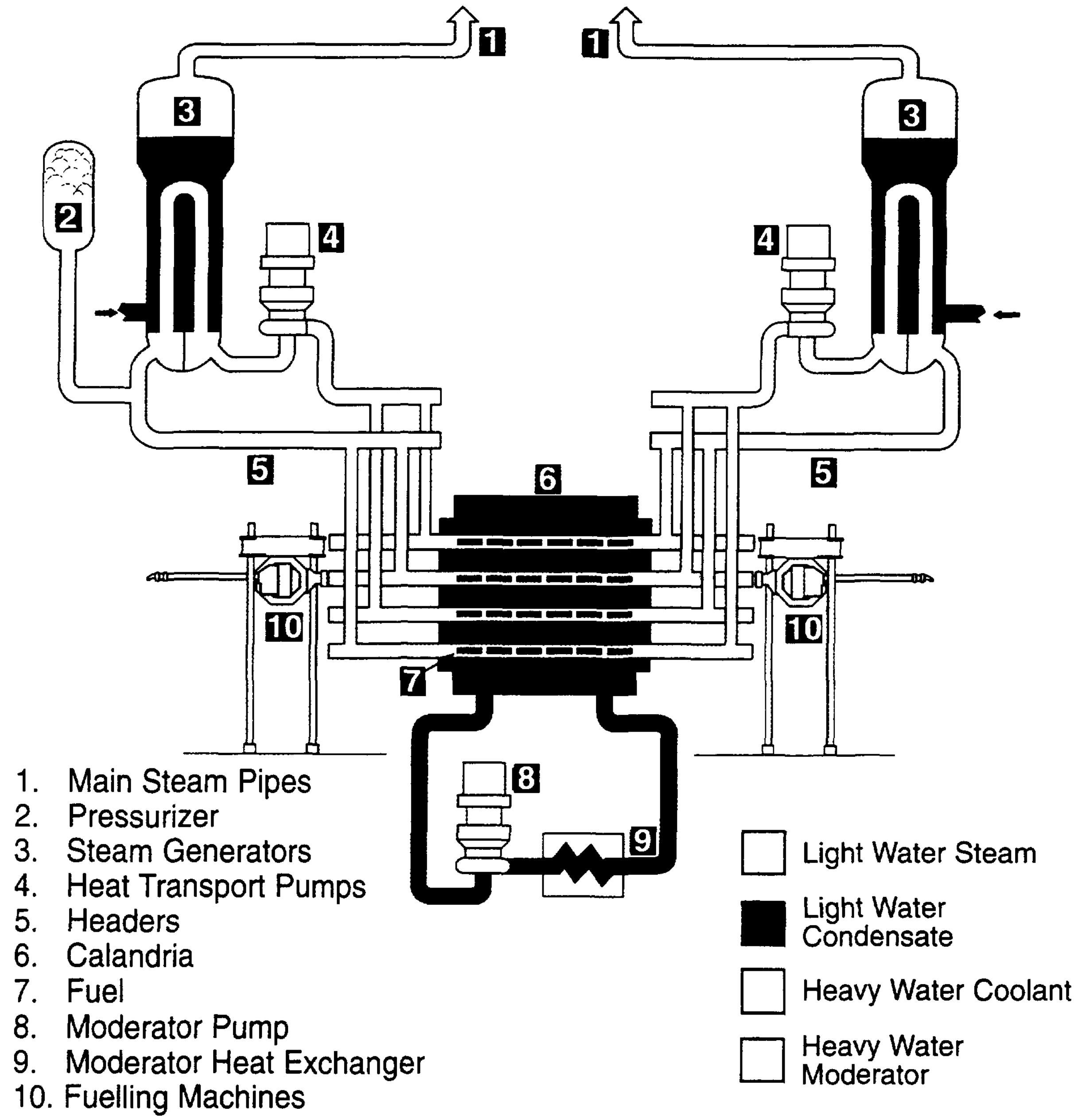


FIG. 1. Schematic illustration of a CANDU reactor core, its primary heat transport system and its fuelling machines.

1.3. SCOPE

This report deals with the pressure tubes of CANDU reactors and Indian PHWTRs. It is focused on Zr-2.5%Nb pressure tubes, however, much of the information on ageing mechanisms, and on inspection, assessment and maintenance methods is also directly applicable to Zircaloy-2 pressure tubes.

For completeness, information is also provided on the design and design basis of a fuel channel, of which the pressure tube is a key component.

The report does not address life or life-cycle management of pressure tubes because it is written from the safety perspective and life management includes economic planning.

1.4. STRUCTURE

The fuel channel design, of which the pressure tube is a key component, is discussed in Section 2. The design basis for the fuel channel is presented in Section 3. The pressure tube degradation mechanisms and ageing concerns are presented in Section 4. These degradation mechanisms include irradiation enhanced deformation, delayed hydride cracking and changes in material properties during irradiation. Inspection requirements and technologies are discussed in Section 5. Fitness-for-service guidelines for estimating the residual life of pressure tubes are presented in Section 6. Mitigation methods consisting of operating guidelines and pressure tube maintenance methods are discussed in Section 7. The report concludes, in Section 8, with guidelines for a systematic pressure tube ageing management programme.

2. FUEL CHANNEL AND PRESSURE TUBE DESCRIPTION

The CANDU fuel channel design results from decisions to [1]:

- Use natural uranium fuel,

- Use heavy water at high temperature and pressure to remove heat from the fuel,

- Contain the fuel and circulating heavy water primary coolant by pressure tubes, as illustrated in Figures 1 and 2, and

- Use a separate low temperature, low pressure heavy water system as the moderator that slows fast neutrons making a chain reaction possible. Figure 3 is a schematic flow diagram showing the independent hot pressurized heavy water primary coolant system, and the low temperature/pressure heavy water moderator system.

The consequences of the decisions associated with basing CANDU reactors on the pressurized fuel channel concept were:

- Fuel channel components that are part of the reactor hot primary coolant circuit would need to be thermally isolated from the low temperature moderator,

- Most reactor internals would need to be made from zirconium alloys as they have a low neutron capture cross-section,

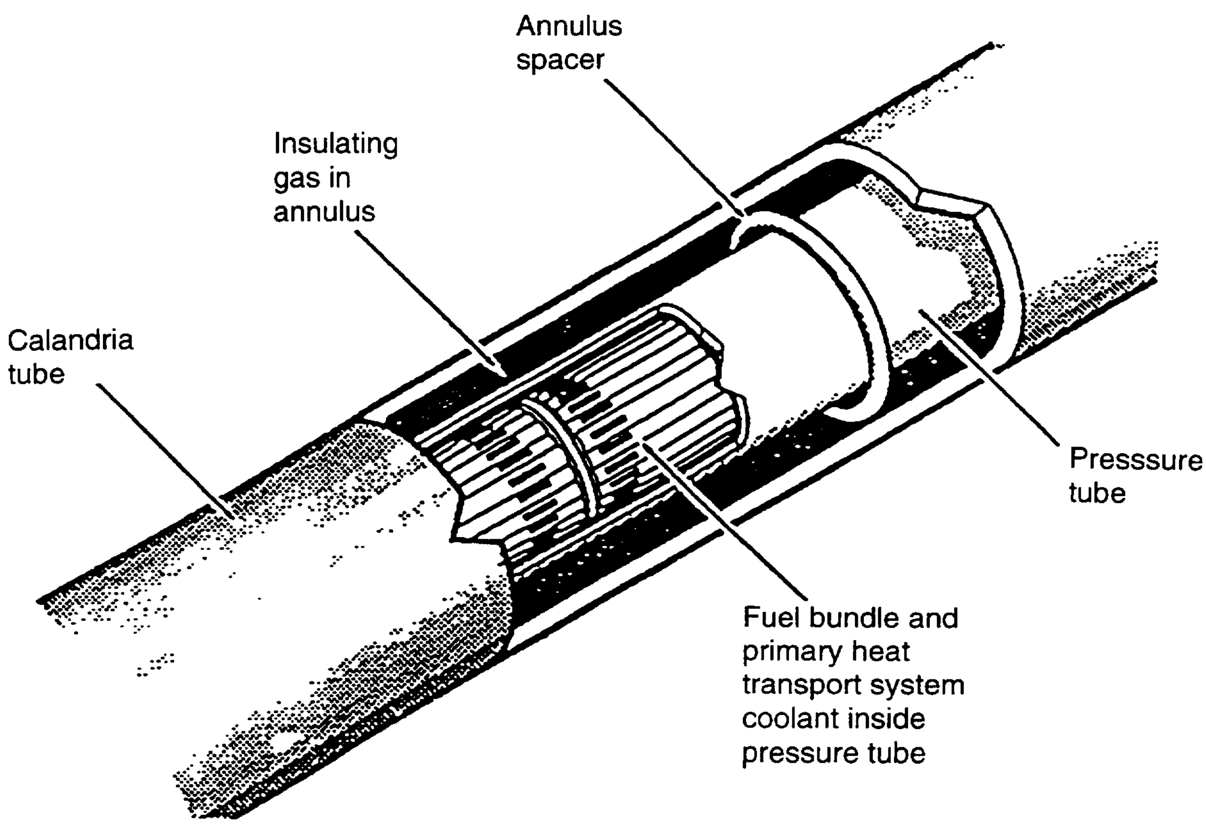


FIG. 2. Schematic illustration of a fuel channel in the reactor core.

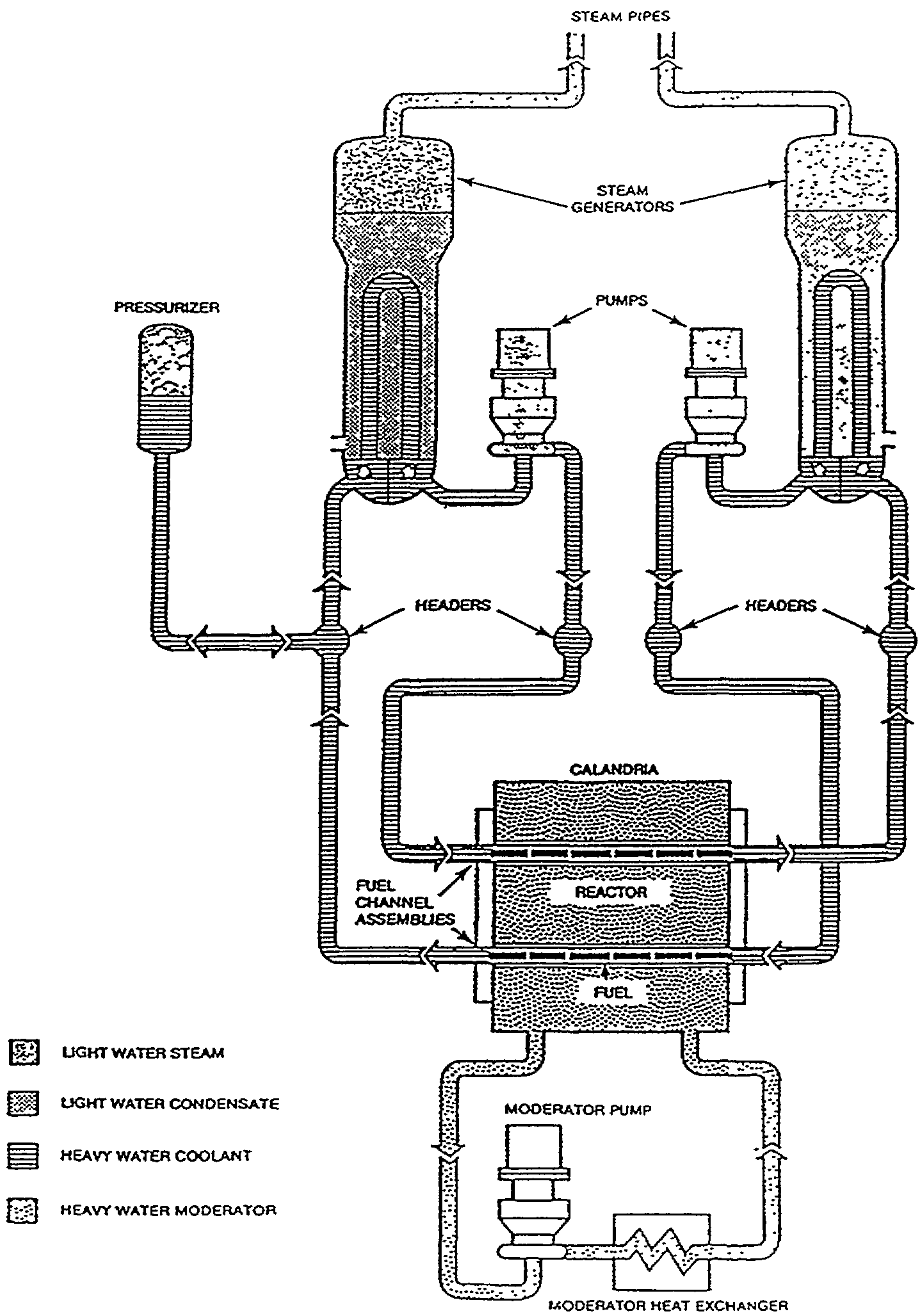


FIG. 3. Schematic illustration of the primary heat transport and moderator systems for a CANDU reactor.

- Relatively easy pressure tube replacement would need to be possible for surveillance and maintenance because knowledge of the long-term behaviour of zirconium alloys was limited, and

- Fuel would need to be moved and replaced frequently to achieve optimum burnup and maintain constant reactivity. On-power refuelling was developed so this could be done efficiently.

2.1. FUEL CHANNEL DESIGN

The design of CANDU fuel channels has evolved to accommodate higher power outputs, reaching a maximum of about 6.5 MW in the current generation of fuel channels. Temperatures and pressures have also increased during this evolution, as indicated in Table III. This has necessitated increases in the length, diameter, and strength of the pressure tubes. However, all channels have had the same basic design, which is illustrated in Figure 4. This figure shows the four key components for each of the few hundred horizontal fuel channels in a typical CANDU reactor core. These components are a pressure tube, a calandria tube, end fittings (2 per channel) and annulus spacers (4 per channel).

The core of a CANDU reactor consists of a large, thin wall, horizontal cylindrical tank (the calandria) that contains low pressure/temperature heavy water moderator. A typical calandria vessel is illustrated in Figure 5. Its ends (the end shields) are joined by a few hundred horizontal Zircaloy-2 calandria tubes having a lattice pitch of 11.25 inches in all commercial CANDU units. Some of the initial CANDU prototypes had a smaller pitch and the reactors in India have an 8 inch lattice pitch.

As illustrated in Figure 2, a pressure tube is located inside each calandria tube. The principle function of these pressure tubes, which are the major component of each fuel channel assembly, is to support and locate the fuel in the reactor core and allow the pressurized heavy water primary coolant to be pumped through the fuel and remove its heat. As the high pressure/temperature primary coolant is slightly alkaline, the pressure tubes must be adequately resistant to corrosion, as well as to creep and growth. Since CANDU and Indian reactors use natural uranium fuel, it is important to use low neutron absorption materials like

TABLE III. CANDU FUEL CHANNEL DESIGN CONDITIONS

CANDU units	In-service year	Total no. of channels	Power output per unit (MW)	Core length/PT diameter (m/cm)	Max. flux ($n/cm^2/s$)	Peak pressure (MPa)	Peak temp (°C)
NPD*	1962	132	20	4/8	-	7.4	285
Douglas Point*	1967	306	200	5/8	2.50×10^{13}	10.5	297
Pickering A (4 units)	1971/73	1560	515	6/10	2.76×10^{13}	9.4	297
Pickering B (4 units)	1982/86	1520	515	6/10	2.76×10^{13}	9.5	297
Bruce A (4 units)	1977/79	1920	850	6/10	3.69×10^{13}	10.2	308
Bruce B (4 units)	1984/87	1920	850	6/10	3.69×10^{13}	10.3	308
CANDU-6 (4 units)	1983/84	1520	630	6/10	3.71×10^{13}	11.0	312
Darlington (4 units)	1990/93	1920	850	6/10	3.69×10^{13}	11.1	313

*These two prototype units are no longer operating.

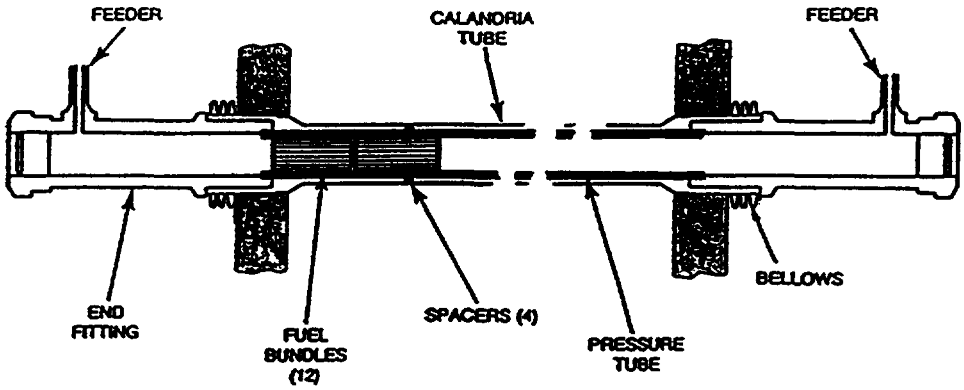


FIG. 4. Schematic illustration of one of the fuel channels located in a CANDU reactor core.

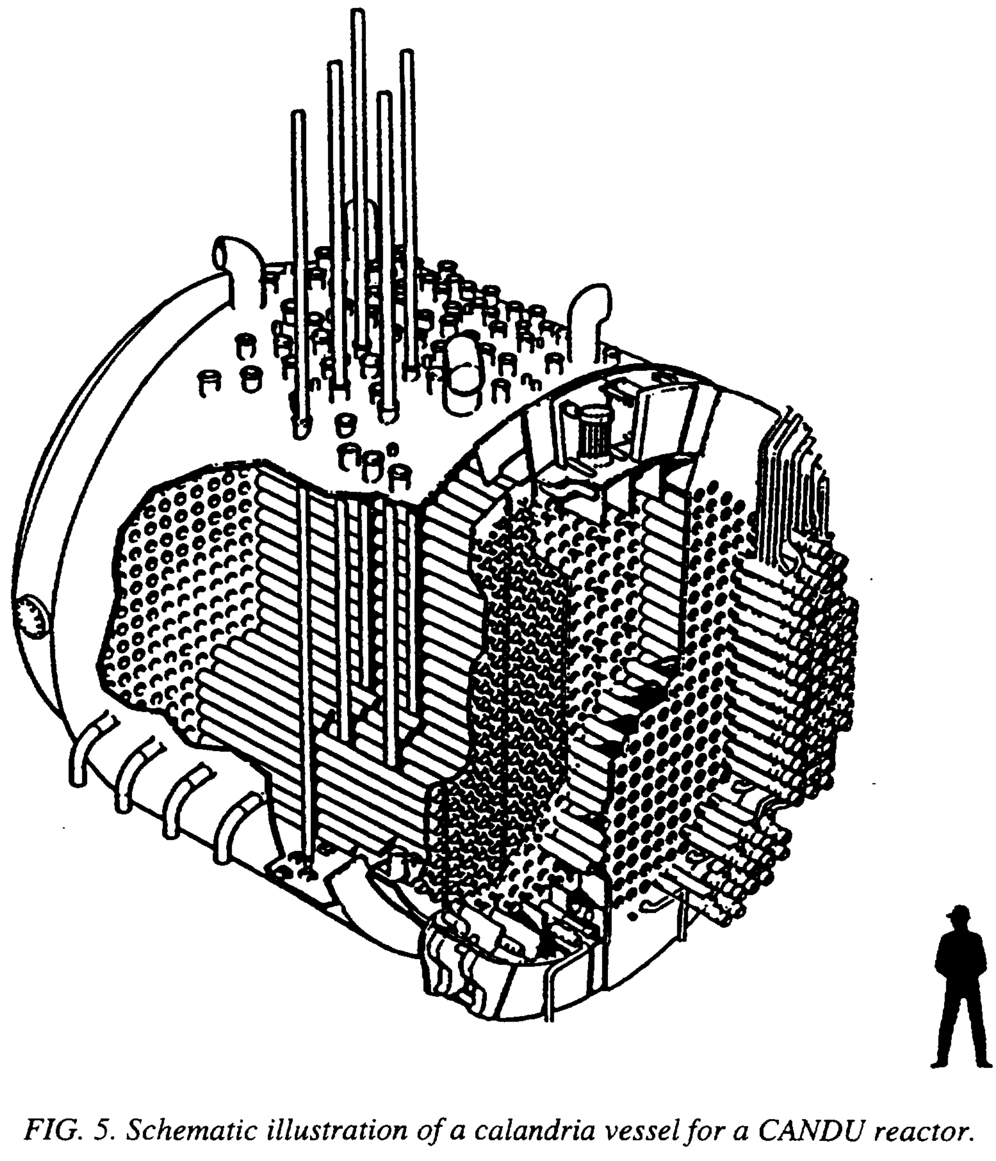


FIG. 5. Schematic illustration of a calandria vessel for a CANDU reactor.

zirconium alloys in the reactor core. In addition, it is necessary to pay special attention to the design and manufacture of in-core structures to ensure that no material is used beyond that necessary to satisfy the appropriate design code.

The NPD and Douglas Point prototype CANDU units, as well as the early Indian units, used cold-worked Zircaloy-2 pressure tubes. The first two commercial CANDU reactors (Pickering units 1 and 2) also initially used this type of pressure tube, but the construction of all other CANDU reactors (and the recent retubings of Pickering units 1 to 4) used cold-worked Zr-2.5%Nb tubes. This alloy, which was discovered by the Russians, was developed and qualified for power reactor use in parallel by AECL. As it is about 20% stronger than Zircaloy-2, tubes having a thinner wall thickness could be used for improved fuel burnup.

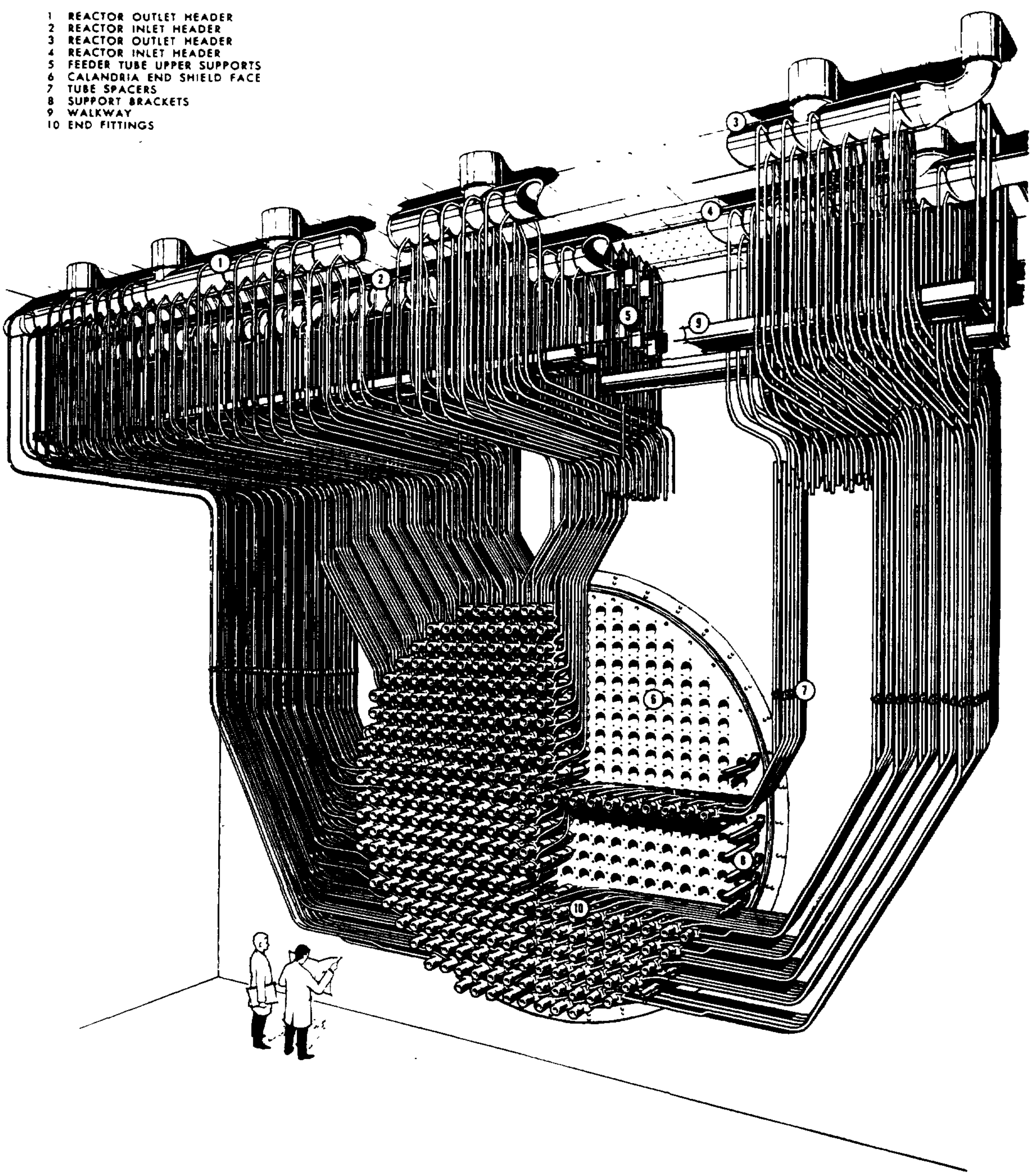


FIG.6. Schematic illustration of the feeder pipes attached to one end of a CANDU reactor core.

TABLE IV. SPACER DESIGN FOR CANDU FUEL CHANNELS

CANDU Unit	In-service date	Initial spacer design / Number of spacers per fuel channel
Pickering 1*	1971	Loose/2
Pickering 2*	1971	Loose/2
Pickering 3*	1972	Loose/2
Pickering 4*	1973	Loose/2
Bruce 1	1977	Loose/2
Bruce 2	1977	Loose/2
Bruce 3	1978	Loose/4
Bruce 4	1979	Loose/4
Pickering 6	1982	Loose/4
Point Lepreau, New Brunswick	1983	Loose/4
Gentilly, Quebec	1983	Loose/4
Wolsong 1, Rep. of Korea	1983	Loose/4
Embalse, Argentina	1984	Loose/4
Pickering 5	1984	Loose/4
Bruce 6	1984	Loose/4, but repositioned before startup
Pickering 7	1985	Loose/4, but repositioned before startup
Bruce 5	1985	Loose/4, but repositioned before startup
Pickering 8	1986	Loose/4, but repositioned before startup
Bruce 7	1986	Loose/4, but repositioned before startup
Bruce 8	1987	Tight/4
Darlington 2	1990	Tight/4
Darlington 1	1992	Tight/4
Darlington 3	1992	Tight/4
Darlington 4	1993	Tight/4
Cernavoda 1, Romania	1996	Tight/4
Wolsong 2, Rep. of Korea	1997	Tight/4

*Rehabilitation of these units has removed the initial spacers and replaced them with the tight spacer design (4 per fuel channel)

As indicated in Table II, from KAPS-2 onward Indian reactors will also have Zr-2.5%Nb pressure tubes. For the earlier Indian units, which use Zircaloy-2 pressure tubes, extensive in-service inspection and post irradiation examination programs are monitoring the material properties of these tubes to ensure safety margins continue to be adequate. These tubes may require replacement after 12 to 15 full power years of operation.

Each end of a pressure tube is roll expanded into the hub of a stainless steel end fitting to form a pressure tight, high strength joint. These end fittings provide a flow path for the primary coolant between the pressure tube and the rest of the CANDU primary heat transport system (PHTS) by having a bolted connection to a carbon steel "feeder pipe" whose diameter is about 7 cm. Figure 6 is a schematic illustration of the feeder pipes attached to one end of a typical CANDU core.

A gas filled annulus between each pressure tube and the calandria tube surrounding it insulates the high temperature/pressure primary coolant inside the pressure tubes from the low temperature/pressure heavy water moderator located outside the calandria tubes, as illustrated in Figures 2 and 4. This annulus was open to the reactor vault in early CANDU designs, but all CANDU units that are currently operating have a sealed annulus connected to an annulus gas system (AGS) that circulates dry CO_2 gas through it. The four early reactors (RAPS and MAPS) in India have open annuli, while NAPS and subsequent Indian reactors have sealed annuli connected to an AGS. As the AGS includes very sensitive moisture detecting instrumentation, it provides an excellent leak detection system for pressure tubes. For the early Indian reactors, highly sensitive leak and moisture detection instrumentation are provided to sense moisture in the calandria vault.

Annulus spacers (spaced about a meter apart) keep each pressure tube separated from the surrounding calandria tube and allow the cool calandria tube to provide sag support for the hot pressure tube containing the fuel and the circulating heavy water primary coolant. These spacers are made by forming wire into a close coiled helical spring. The spacers for early CANDU reactors, and the Indian reactors, use a Zr-Nb-Cu wire, but the current generation of CANDU fuel channels use Inconel wire.

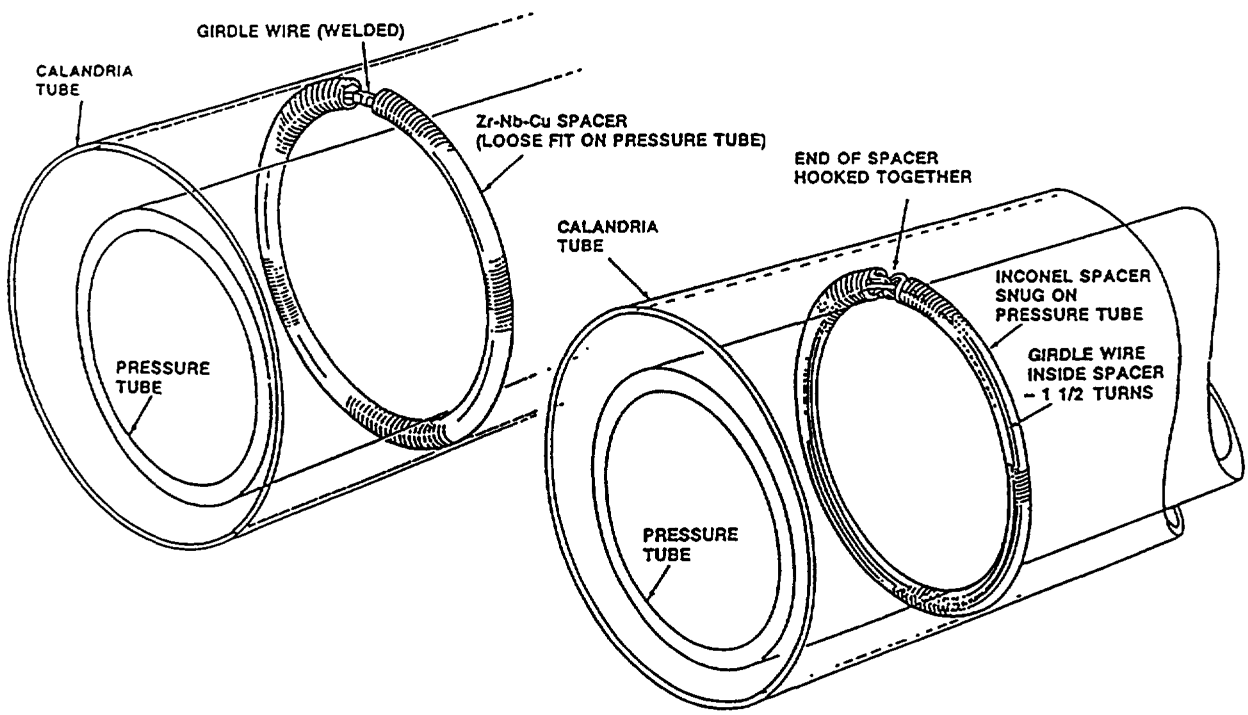


FIG. 7. Comparison of the two types of pressure tube to calandria tube spacers.

The current spacer design has hooks on the two ends of each spring so that it can be stretched around a pressure tube and then hooked together, as illustrated in Figure 7. Therefore, when installed, these annulus spacers conform to the outside diameter of a pressure tube with their tight-fit on the pressure tube preventing them from moving away from their installed positions. Stretching of these springs, as well as the existence of a diametral gap between each spacer and the calandria tube surrounding it, accommodates increase in diameter of the pressure tubes. Axial movement of a pressure tube relative to the calandria tube surrounding it is allowed by a rolling motion of the annulus spacers, which results in almost no wear on the pressure and calandria tubes. The required spacer performance has been demonstrated in an extensive test program, as well as being verified by both in-reactor measurements and examination of fuel channel materials removed for surveillance.

Early units used a spacer design that was a tight-fit inside the calandria tube, and hence a loose-fit on the pressure tube, rather than the current spacer design that is a tight-fit on the pressure tube, as indicated in Table IV and Figure 7.

2.2. PRESSURE TUBE DESIGN

The pressure tube acts as a horizontal beam supported at each end by its attachment to end fittings, and at intermediate points by the surrounding calandria tube, via the "garter spring" spacers. The design of the pressure tube is quite simple, consisting primarily of the determination of its length, inside diameter and wall thickness. The length of the pressure tube is primarily determined from the core length obtained from reactor physics considerations. The inside diameter of the pressure tube is derived from fuel passage and thermohydraulic considerations. A minimum allowable pressure tube wall thickness is determined by a stress analysis, then a 0.2 mm allowance for corrosion and wear is added to determine the minimum allowable wall thickness for pressure tube fabrication.

The current generation of CANDU pressure tubes have the following approximate dimensions:

- 6 m long,
- 11 cm diameter, and
- 4 mm wall thickness.

The dimensions of the tubes in Indian reactors are given in Table II.

In addition to the temperature and pressure of the coolant, which vary along the length of an operating pressure tube, as illustrated in Figure 8, the following aspects are considered during the calculation of stresses and assessment of the fatigue life for the pressure tube:

(a) weight of fuel channel components, fuel and coolant,
(b) feeder pipe loads and torques,
(c) fuelling machine loads,
(d) axial loads due to the fuel channel bellows, fuel movement and end fitting bearing friction,
(e) annulus spacer loads,
(f) loads imposed during a seismic event, and
(g) effects of tube initial bow, misalignment and end of design life sag and elongation.

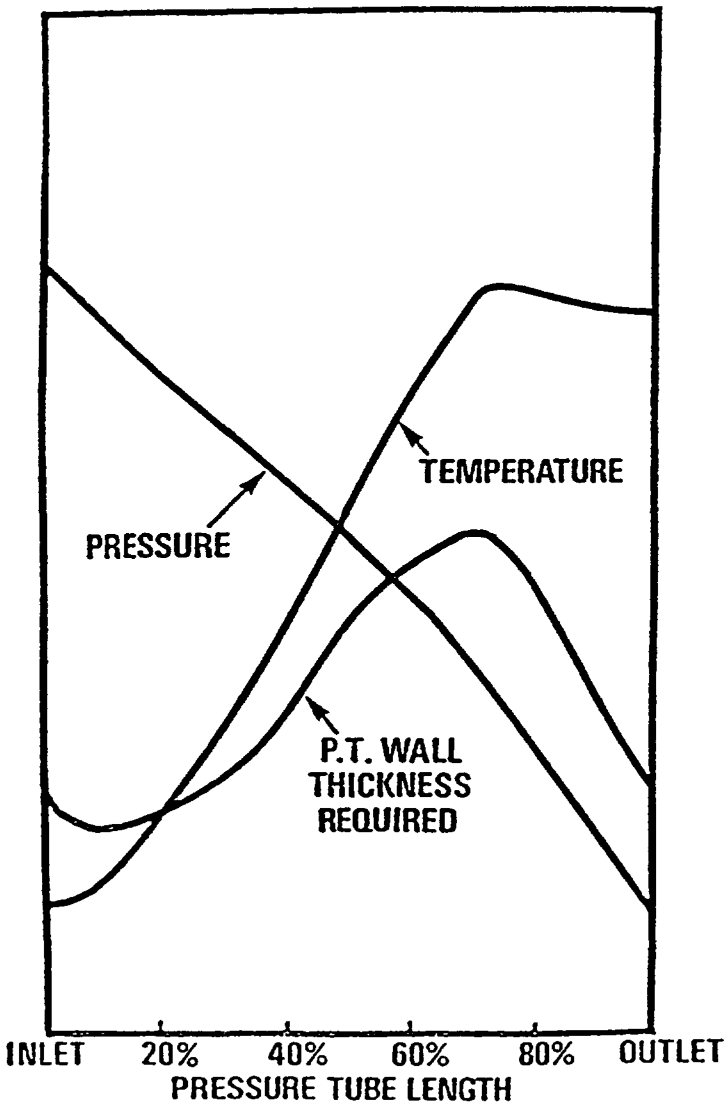


FIG. 8. Typical pressure and temperature variation along the length of a pressure tube and their influence on required wall thickness.

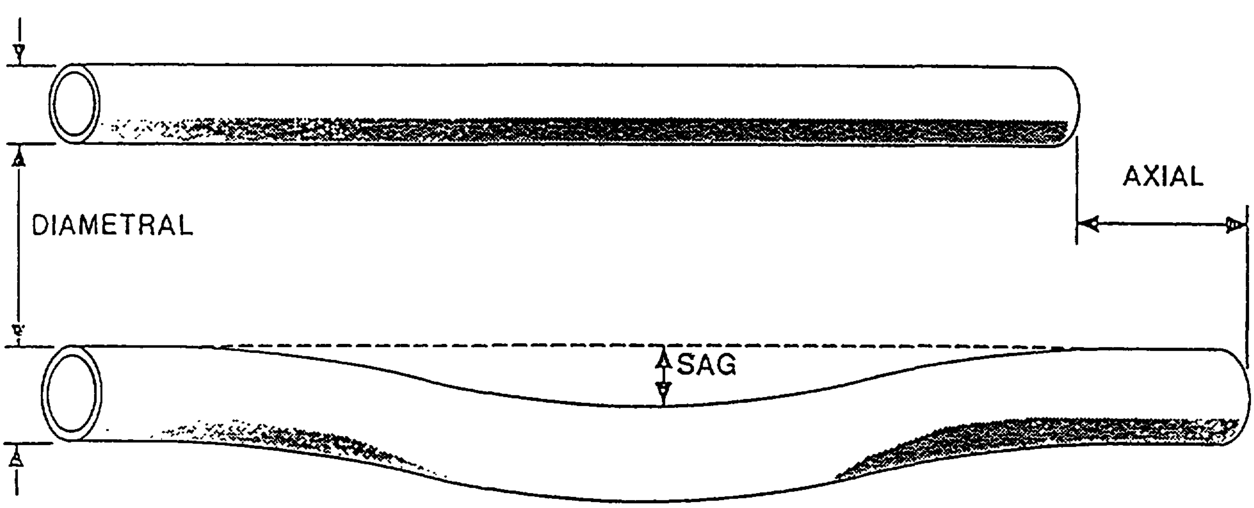


FIG. 9. Pressure tube creep and growth.

Although zirconium alloys are not included in the ASME Boiler and Pressure Vessel Code, the pressure tube allowable design stress was established by an extensive test program on the same basis as that for ASME Class 1 materials. The allowable design stress values for pressure tubes are specified in the Canadian Standards Association (CSA) Standard N285.6.

The pressure tube design analyses take account of the effects of creep and growth, which causes pressure tubes to permanently increase in length and diameter, and to sag from the weight of the fuel and coolant contained in them. These dimensional changes are illustrated in Figure 9. Pressure tube stresses are evaluated for both beginning of life and end-of-life conditions to account for the pressure tube dimensional changes that occur during their 30 year design life. At the beginning of life, the initial wall thickness and unirradiated material properties apply. At the end-of-life, pressure tube diameter and length have increased, while wall thickness has decreased and material strength has increased. For the end-of-life condition, credit is taken for only a small fraction of the strength increase obtained from irradiation, which occurs mostly in the first few years of operation.

Pressure tube deformations due to creep and growth are calculated using equations empirically correlated to measurements of tube deformations in operating reactors and extensive research reactor testing. Calculation of pressure tube axial elongation is used to establish:

- the required length for fuel channel bearings,
- the extension of fuel channel bellows, and
- the stresses associated with the feeder pipe connection.

Calculation of pressure tube diametral increase is used to establish:

- the maximum allowable garter spring spacer diameter that does not become squeezed between the pressure and calandria tubes,
- the PHTS coolant flow conditions at end-of-life, to ensure that flow by-pass does not result in unacceptable heat removal from the fuel, and
- that the pressure tube end-of-life diameter is not more than 1.05 times its initial diameter, to ensure that an adequate creep ductility is maintained, as established in test reactors.

Calculation of channel sag is used to establish:

- the number and spacing of garter spring spacers to prevent pressure tubes from contacting the cooler calandria tubes surrounding them,
- the spacer loads (for use in spacer design),
- that adequate clearance is maintained for fuel passage through the pressure tube as it sags, and
- that clearance is maintained between a sagging fuel channel and the reactivity mechanisms located below it.

Figure 10 is a schematic illustration of a sagged CANDU fuel channel.

Each end of a pressure tube is roll expanded into an end fitting. As can be seen in Figure 11, the pressure tube rolled joint design is very simple, involving only three grooves in the end fitting hub. Each end of a pressure tube is inserted into an end fitting hub with a shrink

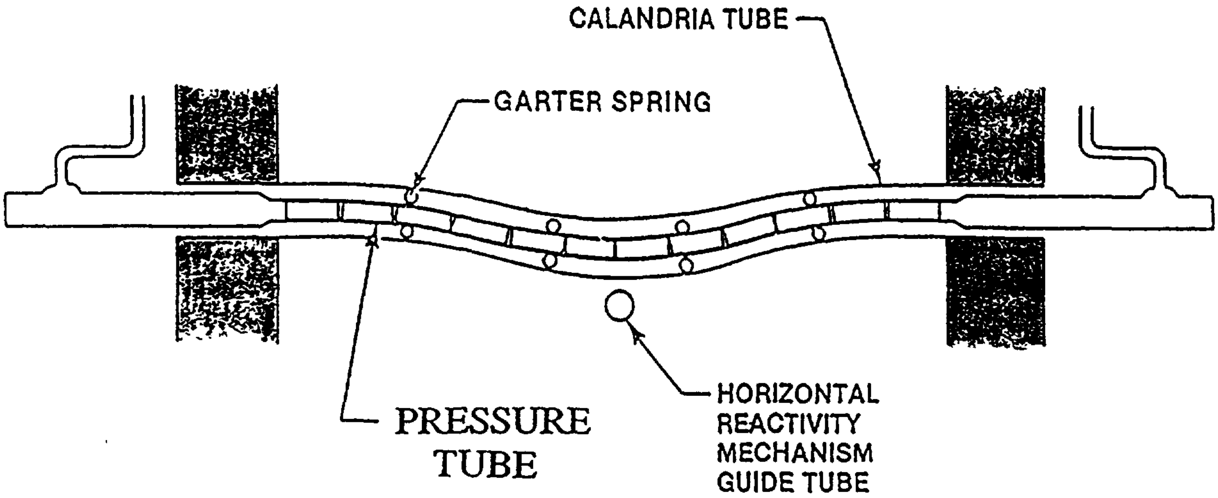


FIG. 10. Schematic illustration of fuel channel sag.

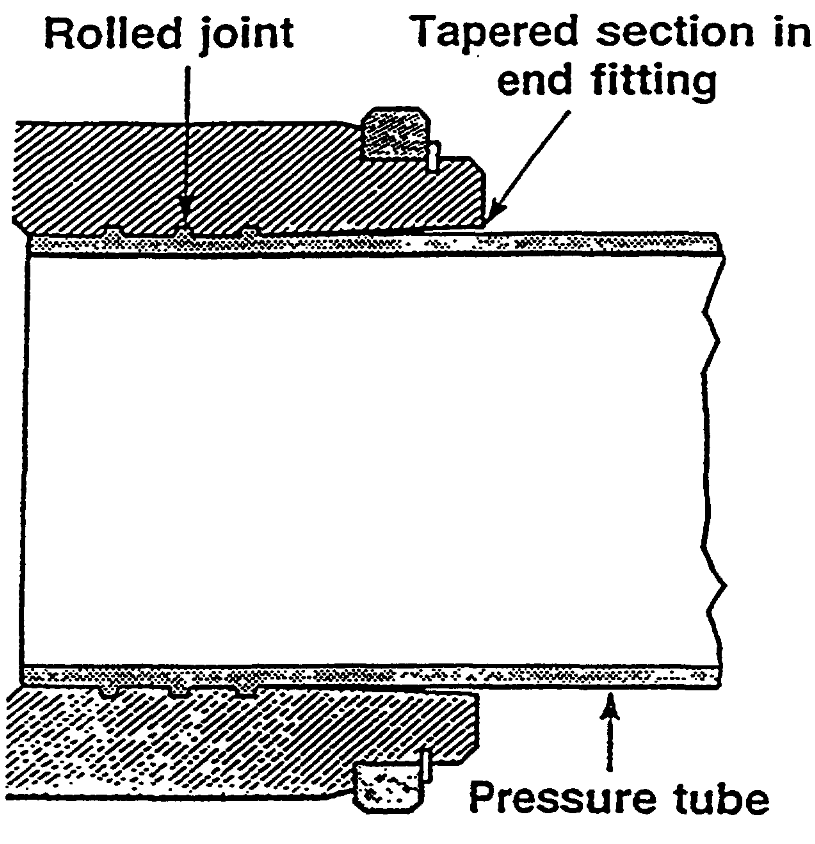


FIG. 11. Pressure tube rolled joint.

fit prior to rolling. The pressure tube is then roll expanded into its hub, creating about a 13% reduction in wall thickness. Although no rules are given in the ASME code for the use of roll expanded joints in Class I vessels, their use has been justified and approved for this application by extensive development testing and analysis, as required by the Canadian Standards Association (CSA) Standard N285.2. Finite element stress analysis of the rolled joint design has been conducted to confirm that the stresses in the rolled joint region meet the allowable limits specified by Section III of the ASME Code. In addition, sample pressure tube rolled joints have been subjected to an extensive test program to demonstrate their acceptability for use in fuel channels [6]. The fuel channel installation procedures ensure that all parameters for production rolled joints are within the ranges established in qualification testing and strictly enforced process controls ensure that there are no deviations from these procedures. After installation of fuel channels in the reactor core, each rolled joint receives a careful visual and dimensional inspection to verify that rolling has been done correctly, and the leak tightness of each joint is checked using highly sensitive helium leak detection techniques.

Because pressure tubes are located in the reactor core, they experience significant changes in mechanical properties and dimensions. As the knowledge available to predict these changes was limited for the early designs, the expectation was always that the life of pressure tubes was finite. Thus, a key feature of the fuel channel has always been that they are designed to allow relatively easy replacement. Because Zr-2.5%Nb pressure tubes are expected to reach their end-of-life after 30 years of operation, while it is believed that most of the other reactor components can operate much longer, replacing all of a unit's pressure tubes permits life extension of a CANDU reactor core. Furthermore, the replaceability of individual fuel channels allows:

- the removal of any individual pressure tube that may have been damaged during operation (for example, severe fretting damage), and

- the removal of surveillance pressure tubes that can be tested in hot cells to provide valuable information on degradation mechanisms for development and predictive purposes.

2.3. MANUFACTURE OF PRESSURE TUBES

The cold-worked Zr-2.5%Nb pressure tubes used in CANDU reactors are made by hot extruding hollow billets into tubes, which are then cold-worked to produce the final dimensions. These fabrication techniques have been developed so pressure tubes can be consistently produced within close dimensional tolerances and with uniform physical and mechanical properties. All manufacturing steps are closely controlled to ensure that the tubes meet very stringent specifications. For example, to ensure that pressure tubes are free of unacceptable flaws, the ingots, billets, and finished tubes are each ultrasonically inspected.

The fabrication process for CANDU cold-worked Zr-2.5%Nb pressure tubes [7], which determines the tube properties that control their performance, is illustrated in Figure 12 and summarized below:

(a) Ingot

Zirconium sponge, which is produced from zirconium silicate ore, is compacted into briquettes along with a master alloy of zirconium, niobium and recycled material. These briquettes are then electron beam welded together to form a long rod that is melted in a consumable electrode arc furnace to form a Zr-2.5%Nb ingot about 0.6 m in diameter. The ingots, which can each produce 30 to 40 tubes, are machined to remove surface flaws and are then ultrasonically inspected for solidification flaws using a normal beam technique on the machined cylindrical surface. The machined ingot is also checked for conformance with the specification requirements for the two major alloying elements (niobium and oxygen) and 23 residual elements.

(b) Forging

A portion of an ingot is preheated to close to 1000°C and forged into round logs about 0.2 m in diameter using press and rotary forges. A second chemical analysis is done on some of the material, then the logs are machined to remove their outer surface and to produce hollow billets. To refine the grain size and produce a more uniform microstructure, the machined billets are beta quenched from about 1000°C. The machined billets are then

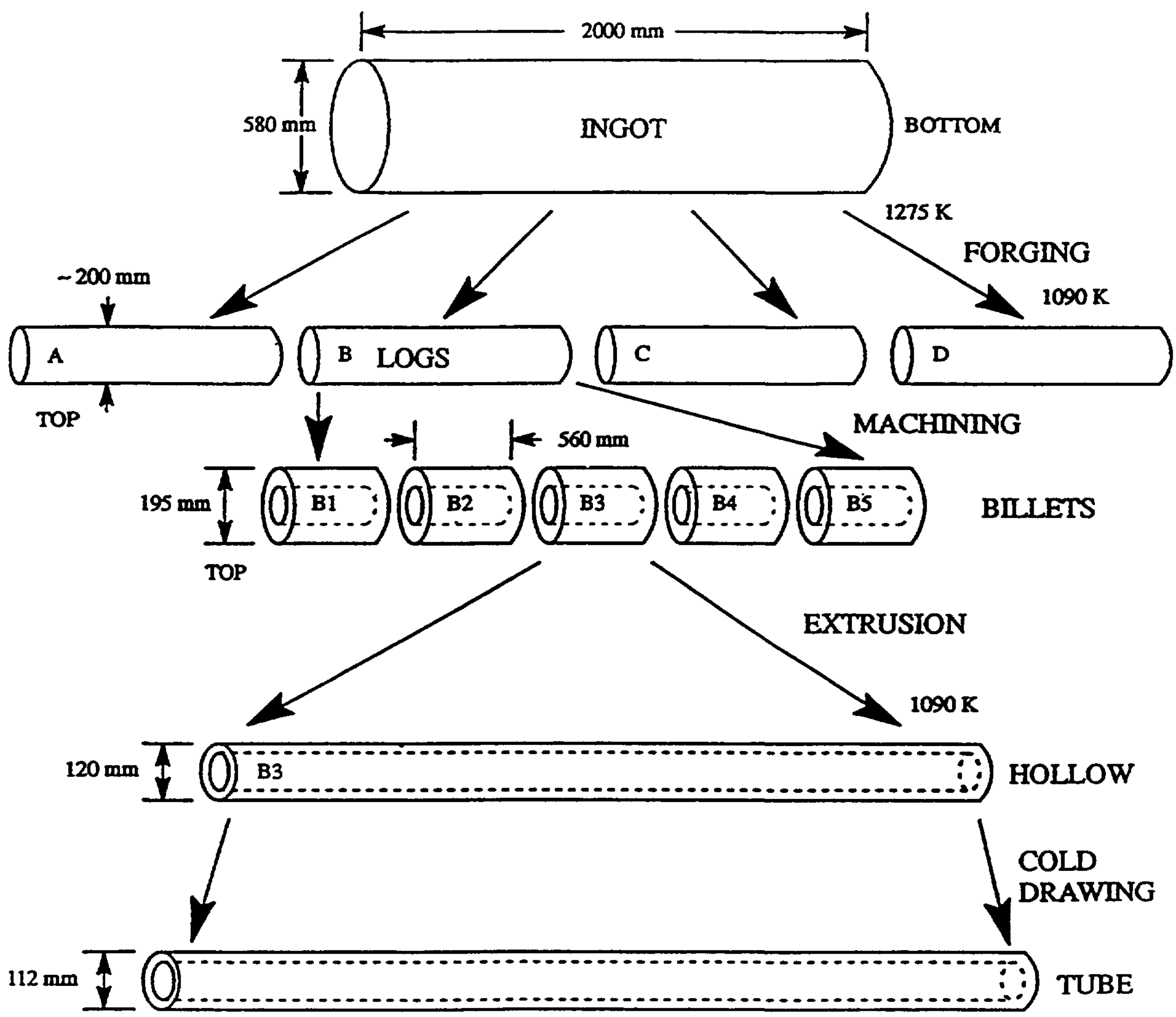


FIG. 12. Schematic illustration of production sequence for CANDU pressure tubes.

ultrasonically inspected for flaws using the pulse-echo immersion technique. Two angled shear wave beams, travelling in opposite directions around the circumference, and one compression wave beam propagating in a radial direction, are used.

(c) Extrusion

The short hollow billets are preheated to about 850°C for extrusion into tube hollows.

(d) Cold drawing

The tubes are then cold drawn in two passes, each 12 to 15% reduction in area. The cold drawn tubes are honed with at least 0.06 mm being removed from the inside surface to ensure that any small surface flaws are removed. The outsides of the tubes are then centreless ground to the required wall thickness.

After the honing and centreless grinding operation, an ultrasonic immersion inspection is conducted that includes angled shear wave examination with two beams travelling in opposite directions around the circumference and two beams travelling in opposite directions parallel to the axis of the tube.

The product specification requires that the following tests and examinations also be carried out on each pressure tube before it can be accepted:

(i) Hydrostatic pressure test to about twice the operating stress.
(ii) Chemical analysis of an off-cut to ensure conformance of the four major alloying elements (niobium, oxygen, nitrogen and hydrogen).
(iii) Tensile testing of an off-cut at 300°C to ensure conformance with the specified strength requirements.
(iv) Corrosion testing of a small sample to confirm that the metallurgical condition is as required.

(e) Finishing operations

The tubes are stress relieved in an autoclave for 24 hours at 400°C. This treatment produces a hard adherent black lustrous oxide layer on the tubes which acts as a hydrogen/deuterium barrier and provides some wear resistance during operation. This autoclaved surface also reveals if the tube surface had any contamination or mishandling prior to autoclaving. The pressure tubes finishing operations also involve:

(i) Cleaning,
(ii) Visual examination inside and out,
(iii) Additional ultrasonic examination,
(iv) Eddy current examination for surface flaws, and
(v) Dimensional inspection (corrections of ovality and bow are allowed so that the tubes meet the dimensional requirements).

All of the manufacturing inspection, testing and finishing for the pressure tubes of Indian reactors is very similar to those described above. However, from MAPS-2 onward, Indian reactors have cold-pilgered pressure tubes. The microstructure, dislocation density and texture of these pilgered tubes was selected after extensive testing. Their texture provides a favourable hydride orientation in the tube's circumferential plane.

3. DESIGN BASIS FOR THE FUEL CHANNEL AND PRESSURE TUBE

The primary purpose of the fuel channels in CANDU and Indian reactors is to support and locate the fuel within the reactor core and to facilitate the flow of pressurized primary coolant to remove the heat generated by the nuclear fission process. The pressure boundary components for the reactor primary coolant, and the internals of the fuel channel, are designed to withstand the primary coolant flow, temperature, and pressure, including the transient service conditions imposed by the primary heat transport system (PHTS). The design life for all fuel channels installed during initial reactor construction has been 30 years at 80% capacity.

The materials of all channel components must exhibit acceptable corrosion resistance and change in properties, etc., during their design life. In addition, the fuel channel must *satisfy all of the safety, performance, etc., requirements given below.*

3.1. FUNCTIONAL REQUIREMENTS

The functional requirements of the fuel channel are to:

(a) support and locate fuel in the reactor core,

(b) permit the PHTS coolant flow to efficiently remove fuel heat with low pressure drop and acceptable fuel vibration,

(c) permit passage of fuel through the reactor core during on-power refuelling,

(d) provide for low leakage fuelling machine connections onto the fuel channel ends at full primary coolant pressure and temperature for refuelling at full power,

(e) form part of the PHTS pressure boundary,

(e) provide connections to the PHTS feeders,

(f) accommodate thermal, as well as pressure tube creep and growth dimensional changes,

(g) provide for detection of leakage from the pressure or calandria tube,

(h) provide thermal insulation so the transfer of primary coolant heat to the moderator and end shield coolant is minimized in normal operation,

(i) minimize neutron absorption,

(j) provide shielding to attenuate radiation emanating from the reactor core,

(k) retain shield plugs and channel closures,

(l) permit full length in-service inspection of pressure tubes for surface and volumetric flaws,

(m) permit measurement of either the gap between a pressure tube and its calandria or the axial positions of annulus spacers,

(n) allow for fuel removal with one fuelling machine disabled, and

(o) be quickly replaceable.

3.2. SAFETY REQUIREMENTS

The pressure boundary integrity of the pressure tube is maintained during all level A, B, C and D service conditions. In addition, it is demonstrated that the spontaneous rupture of any fuel channel does not lead to rupture of any other channel, or impair the ability to shut down the reactor. It is also shown that, under certain postulated accident conditions, the fuel channel design provides for effective heat transfer from the fuel channel to the moderator.

3.3. SEISMIC REQUIREMENTS

The Canadian seismic requirements are documented in the CAN3-N289 series of standards. Fuel channel seismic qualification involves analysis to confirm the integrity of its pressure boundary to ensure that PHTS coolant will continue to flow through it after a design basis earthquake (DBE). This analysis shows that ASME, Section III, requirements are satisfied using the DBE as a level C loading condition.

An analysis is done for the fuel channels as part of the whole reactor assembly to determine the seismic loading on the fuel channels. The calculated magnitude of the seismic loads, and the number of seismic stress cycles, is then provided as input to the fuel channel stress analysis.

3.4. RELIABILITY AND MAINTAINABILITY REQUIREMENTS

The fuel channel components are designed, fabricated and installed to ensure reliable and essentially maintenance-free operation, and a high level of confidence that no channel failure is expected during their design life. It is anticipated that all of the Zr-2.5%Nb pressure tubes in a CANDU reactor core will be replaced after about 30 years operation so that the life of the reactor core can be extended. The replacement of fuel channels is made as efficient and simple as possible to minimize the cost and radiation exposure associated with such retubing of a reactor core.

The early Indian reactors have Zircaloy-2 pressure tubes, hence their deuterium pickup rates during service are higher than Zr-2.5%Nb tubes. Therefore, these Zircaloy-2 tubes may require replacement after 12 to 15 full power years of operation. The time at which the tubes in early Indian units will be replaced will be determined by leak before break (LBB) criteria that is based on an extensive program of post-irradiation examination and analytical studies. These reactors may provide an additional 30 years of operation after the installation of Zr-2.5%Nb tubes in their cores.

3.5. INSPECTION AND TESTING REQUIREMENTS

In-service inspection of fuel channels is performed in accordance with the requirements of CSA Standard N285.4 on the Periodic Inspection of CANDU Nuclear Power Plant Components. Besides mandatory periodic inspections, utilities have instituted additional in-service inspections for reasons of maintenance, operational reliability, to provide assurance to regulators, etc. As much as possible, fuel channel inspections are conducted when a reactor is shut down for other reasons, so these inspection have the minimum possible effect on reactor capacity factors.

3.6. PERFORMANCE REQUIREMENTS

The fuel channel performance requirements are:

(a) Shielding requirements

The fuel channels must incorporate radiation shielding where they pass through the calandria end shields, so that maintenance and inspection activities can be carried out in low radiation fields during reactor shutdowns.

(b) Leakage requirements

All joints in the fuel channel are designed and fabricated to minimize leakage, as dictated by the high cost of heavy water and the requirement to avoid interference with the LBB behaviour of pressure tubes. Each production rolled joint that connects a pressure tube to an end fitting is helium leak tested to show it has an equilibrium leak rate of less than 2×10^{-5} atm.cc/s of helium.

(c) Leak detection sensitivity

To provide a LBB behaviour for pressure tubes, the gas annuli between the outside of each pressure tube and the inside of the surrounding calandria tube must be capable of being monitored to detect a leak from a postulated crack in the pressure tube. The leak detection capability provided by the annulus gas system (AGS) of CANDU reactors is sufficiently sensitive so that the reactor can be shut down and depressurized long before a postulated crack, growing by the delayed hydride cracking (DHC) mechanism, reaches its unstable length. DHC is discussed in more detail in Section 4.1.

Leak detection reliability of the early Indian reactors, for which the fuel channel annuli are open to the reactor vault, is achieved by closely monitoring the sensitivity and reliability of the vault moisture detection instrumentation to ensure an adequate margin is maintained between leak detection and a postulated pressure tube crack becoming unstable. This ensures reactor shutdown and depressurization can be accomplished before a crack reaches the critical size. Pressure tube replacement is planned when the time margin between a reactor being in a safe shutdown condition after leak detection and a crack reaching the critical size becomes unacceptably small.

3.7. MATERIAL REQUIREMENTS

The material specifications for fuel channel components are listed in the Canadian Standards Association (CSA) Standard N285.6. These define the requirements for impurities, heat treatment, mechanical properties, cleanliness, and testing.

(a) Corrosion and wear allowance

Pressure tube wall thickness includes an allowance for both internal and external corrosion, for thinning due to creep and growth and for wear due to fuel movement. These allowances are taken into account when calculating the dimensions to be used for stress analysis.

(b) Allowance for the effect of environment

The combination of stress, temperature, and fast neutron flux results in irradiation enhanced creep and growth of the pressure tube, as illustrated in Figure 9. Pressure tube axial and diametral dimensional changes, and fuel channel sag are accommodated.

It has been demonstrated that the in-service effects of stress, irradiation, temperature, hydrogen/deuterium absorption, and any other significant environmental factors on material properties do not cause an unacceptable reduction in the stress margins governing new construction. Furthermore, protection against non-ductile failure is provided in accordance with ASME Section III, paragraph NB-3211(d).

3.8. CODES AND STANDARDS

The fuel channel components are manufactured to the CSA-Z299 Quality Assurance Standards. The design, analysis, and testing of fuel channel primary coolant pressure boundary components is performed to the requirements of CSA Standard N285.2 for Class 1 components. This standard applies the intent of Section III of the ASME Code to the unique features of pressure tube reactors with on-power refuelling, by providing rules that are supplementary to ASME in areas like the design, materials and joining techniques for fuel channels.

(a) Design

Fuel channel components use the design-by-analysis process of ASME, Section III, Subsection NB3200. The ASME Code rules are complemented by additional requirements in CSA-N285.2, in particular to address the possibility of DHC in zirconium alloys.

To prevent failure due to DHC, CSA-N285.2 imposes limits on the maximum tensile stress under level A and level B service conditions, plus the initial residual tensile stress. It also requires that pressure tubes be supported such that they do not contact the calandria tube, as this could produce large thermal gradients through the tube wall which may lead to hydride accumulation and the potential for unstable cracking.

(b) Materials

The cold-worked Zr-2.5%Nb alloy is not presently listed in the ASME Code, Section III, Appendix 1 as a material for Class 1 application. Requirements have therefore been specified in CSA-N285.6.1, "Seamless Zirconium Alloy Tubing for Fuel Channels". Some early CANDUs used cold-worked Zircaloy-2 for pressure tubes, which is also covered by this standard. The design data for these alloys are provided in CSA-N285.6.7, "Zirconium Alloy Design Data".

For inspection and nondestructive examination of zirconium alloy components, the requirements of the ASME Code, Section V, are supplemented by the additional requirements of CSA-N285.6.6, "Inspection Criteria for Zirconium Alloys".

(c) Joining

Zirconium alloys cannot be welded satisfactorily to steels due to the formation of brittle intermetallic compounds. Mechanical (rolled) joint designs have therefore been developed for pressure tubes and calandria tubes to join them to their fittings, tubesheets or housings. These joints involve residual compressive stress for sealing. In addition, the tube material is extruded into grooves in their hub to provide pull out strength, so that the joints will not separate under their service loadings.

CSA-N285.2 provides rules which these rolled joints must meet. Prototype pressure tube rolled joints are required to be qualified by both test and analysis. Production joints are made using the same procedures and tooling design as used in the qualification of the prototypes. Each production rolled joint is tested for leakage and receives a thorough dimensional inspection to verify that rolling has been done correctly.

3.9. EVOLUTION OF FUEL CHANNEL DESIGN AND MANUFACTURING

Since the early reactors were designed before the irradiation enhancement of pressure tube deformation was known, they did not have adequate allowances for the deformation that occurs during 30 years of operation. The channels associated with these reactors have required an ongoing program of axial repositioning (TUBESHIFT) and it is expected that many of these channels will be replaced before they have operated for 30 years. The channel designs for later reactors include modifications (longer fuel channel bearings, smaller diameter coils for the annulus spacers and additional spacers associated with each pressure tube) to provide deformation allowances that accommodate a 30 year pressure tube design life.

Because all crack propagation, and most crack initiation, for CANDU pressure tubes has been associated with delayed hydride cracking (DHC), most design modifications associated with these tubes have been to eliminate the possibility of DHC occurrence (by reducing hydrogen/deuterium levels or large stress concentrations in the tubes, etc.) and to increase the probability that leak before break (LBB) will be the consequence if DHC does ever occur at operating temperature.

(a) Change in pressure tube material

An accelerated corrosion and rate of deuterium (D) pickup from the PHTS coolant has been observed for Zircaloy-2 pressure tubes. As this causes their hydrogen (H) plus D concentration to significantly exceed the terminal solid solubility (TSS) before they have operated for 30 years, all CANDU reactors built after Pickering units 1 and 2 have used Zr-2.5%Nb pressure tubes instead of Zircaloy-2 tubes. Examination of several Zr-2.5%Nb pressure tubes has shown that they do not exhibit the accelerated D pickup rate of Zircaloy-2.

(b) Reducing hydrogen concentration of a new pressure tube

Recent improvements in the manufacturing process for Zr-2.5%Nb pressure tubes have significantly reduced the H concentration of tubes after manufacturing, so tubes are now fabricated with an H content of less than 5 ppm [5], as illustrated in Figure 14.

(c) Eliminating tube lamination flaws

A few top-of-ingot tubes in early CANDU units have had a significant tensile stress concentration which had escaped detection during manufacture. They contained sharp lamination flaws caused by deep shrinkage cavities [25]. The tube manufacturing process has been changed to minimize the size of the shrinkage cavities in ingots and the tops of ingots are now cropped and removed. For tubes now entering service, final inspections employing enhanced techniques ensure that the probability of such laminations being detected is very high.

(d) Reducing rolled joint tensile residual stresses

If there is not a large temperature gradient through the pressure tube wall, then the rolled joint is the area of the pressure tube most at risk to DHC because it has the highest D ingress and the highest tensile stress. This was illustrated in 1974/5 when through-wall cracks were created by DHC near the rolled joints of some early Zr-2.5%Nb pressure tubes (primarily in Pickering units 3 and 4) and caused leakage of coolant into the annulus between pressure and calandria tubes [21]. As this leakage was detected and the units were safely shut down for tube replacement, LBB was demonstrated.

It was found that a very high tensile residual stress had been caused by improper positioning of the rollers used to make the rolled joints in these early units, as illustrated in Figure 19. The rolling procedure permitted the roll expander to be inserted too far into the pressure tube, and expanded the tubes in the tapered portion of the end of the end fitting bore where they were not supported properly.

When this problem of excessive rolled joint residual stresses was discovered, fuel channels for the two earliest Bruce units had been installed, but not operated. To prevent early life DHC, all joints were given a pre-service stress relief.

The rolled joint high residual tensile stresses associated with improper rolling procedures for early reactors were rectified for subsequent units. Improvements were made in both the rolling procedure and the rolled joint design.

(e) Leak detection improvements

The pressure tube cracking experience described above emphasized the need to have the annulus gas system (AGS) provide a sensitive detection of leakage, so LBB limits the consequences of pressure tube cracking, if it occurs. The initial AGS designs for early units were open to the reactor vault. When the annuli were first sealed from the vault, they did not have gas continuously flowing through the fuel channel annuli, instead the gas was periodically (every few days) removed and replaced. This provided an adequate detection of pressure tube leakage when DHC could only occur slowly during cold shutdowns, and could not occur at operating temperatures when tube cracking would have a much higher velocity.

For the latter portion of the 30 year pressure tube design life, the H/D concentration increases and the possibility of DHC occurrence at operating temperatures also increases. To ensure that there continues to be a high probability of LBB, in case of DHC, the AGS design was modified to have gas continuously recirculating through the fuel channel annuli. This allows leakage from a pressure tube crack to be detected very quickly.

(f) Increasing pressure tube fracture toughness

In a DHC scenario where a leaking pressure tube crack is not detected, the crack may grow until it reaches the critical crack length (CCL), and lead to unstable fracture. Therefore, to ensure LBB behaviour for pressure tubes, it is desirable that the CCL value be as large as possible. It was recently observed that a modification of the ingot melting process (using a multiple melting process rather than the double melting process used previously) results in the pressure tube fracture toughness being 2 to 4 times higher than previously [36]. Future CANDU pressure tubes will be made using this improved ingot melting process.

(g) Annulus spacer design improvement

Some pressure tubes in early reactors have sagged into contact with the cooler calandria tubes surrounding them, because the fuel channel spacer design used in these reactors permitted some of the spacers to be displaced from their design positions during reactor construction and commissioning. If spacers were sufficiently displaced, contact between a pressure tube and the cooler calandria tube could result from pressure tube sag. To eliminate the local accumulation of hydrides that occurs at a "cold spot" where a hot pressure tube has sagged into contact with the cooler calandria tube, all CANDU fuel channels fabricated in the past decade have used an improved spacer design, as indicated in Table IV. The design of spacers was improved by changing the material used to fabricate them, from a zirconium alloy to Inconel X750. This allowed the spacer design to be a tight fit around the pressure tube, as illustrated in Figure 7. If fuel channels have four spacers of this newer design, then it is predicted that contact between pressure and calandria tubes does not occur for much longer than 30 years.

4. DEGRADATION MECHANISMS AND AGEING CONCERNS FOR PRESSURE TUBES

Pressure tubes are exposed to temperatures up to 313°C, internal pressure up to 11 MPa, neutron fluxes up to 3.7×10^{13} n/cm^2/s and fluences up to 3×10^{22} n/cm^2 (in 30 years of operation at 80% capacity). These conditions cause changes in dimensions and material properties through irradiation damage and microstructural evolution. The tubes are also subjected to corrosion by the slightly alkaline heavy water coolant that flows inside them, with some of the deuterium resulting from this corrosion process being absorbed by the zirconium alloy pressure tubes. Although the overall performance of the more than 10 000 pressure tubes installed in CANDU reactors has been good, some of the pressure tubes in early reactors have leaked and two pressure tubes have ruptured, one while at operating conditions, due to delayed hydride cracking (DHC).

Generic concern about the material condition of the Zircaloy-2 pressure tubes used during the initial construction of the first two commercial CANDU reactors led to the replacement of all these tubes in Pickering units 1 and 2 after slightly more than 10 years of operation.

After 15 to 20 years of operation, all of the pressure tubes in Pickering units 3 and 4 have also been replaced, because they did not have sufficient allowances for the deformation of their initial pressure tubes. Although the pressure tubes installed in CANDU and Indian reactors during their initial construction have always had a design life of 30 years, the in-service deformation for the tubes in the earliest units was under-estimated.

The primary ageing mechanisms for zirconium alloy pressure tubes are:

- Delayed hydride cracking,
- Irradiation enhanced deformation and
- Changes of pressure tube material properties.

These primary pressure tube ageing mechanisms are discussed below.

4.1. DELAYED HYDRIDE CRACKING (DHC)

The solubility limit of hydrogen (H) and deuterium (D) in zirconium is low [8], as illustrated in Figure 13. Hydrides are always present in pressure tubes at room temperature, while in the range of reactor operating temperatures, hydrides can form only when the H equivalent (Heq) concentration (the sum of the H ppm concentration in a new tube plus half the ppm concentration of the D picked up during service) is greater than the terminal solid solubility (TSS) at operating temperature. The TSS in operating pressure tubes is about 40 to 70 ppm, depending on the tube temperature.

For DHC to occur, there must be both hydrides present and sufficient tensile stress. If the H/D content of a pressure tube is low, then DHC can only occur when the reactor is cold, because hydrides could not be present at operating temperatures. If the H/D content is large enough to allow hydrides to exist at operating temperatures, then DHC could occur during reactor operation, if a sufficiently large tensile stress exists.

The consequence of DHC, when it occurs, usually is pressure tube cracking, leakage, and reactor shutdown for tube replacement. It is expected that leak before break (LBB) will

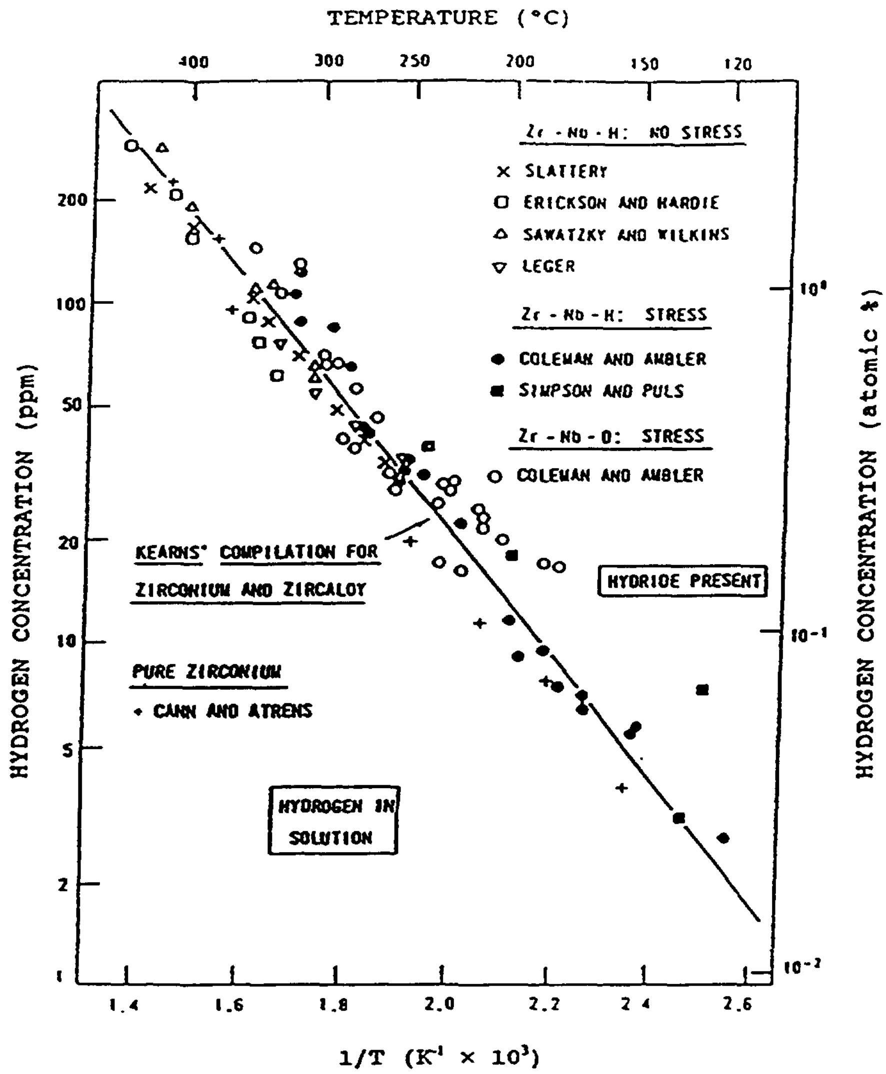


FIG. 13. Comparison of measurements of solubility limits of hydrogen in zirconium alloys.

usually be associated with a DHC failure of a pressure tube [9, 10]. However, tube rupture could occur if leakage is not detected before a leaking crack grows to the unstable length.

The presence of hydrides has been associated with all crack propagation in zirconium alloy pressure tubes, and most of their crack initiation processes [11, 12], therefore the H content in new tubes and the D picked up during service need to be minimized.

4.1.1. Hydrogen concentration of a new pressure tube

Analysis of CANDU pressure tubes shortly after being fabricated indicates that they contain a small amount of H. For tubes fabricated before 1990, 5–15 ppm of H is the typical range for the H concentration of the as-fabricated tubes (up to 25 ppm of H was allowed). However, recent improvements in the manufacturing process for Zr-2.5%Nb pressure tubes have significantly reduced the H concentration of tubes after manufacturing, so tubes are now fabricated with an H content of less than 5 ppm [5, 13, 14], as illustrated in Figure 14.

With a typical D ingress rate of 1 ppm (Heq) per year, the consequence of this can be to increase the time it takes to form hydrides in operating pressure tubes by up to 20 years.

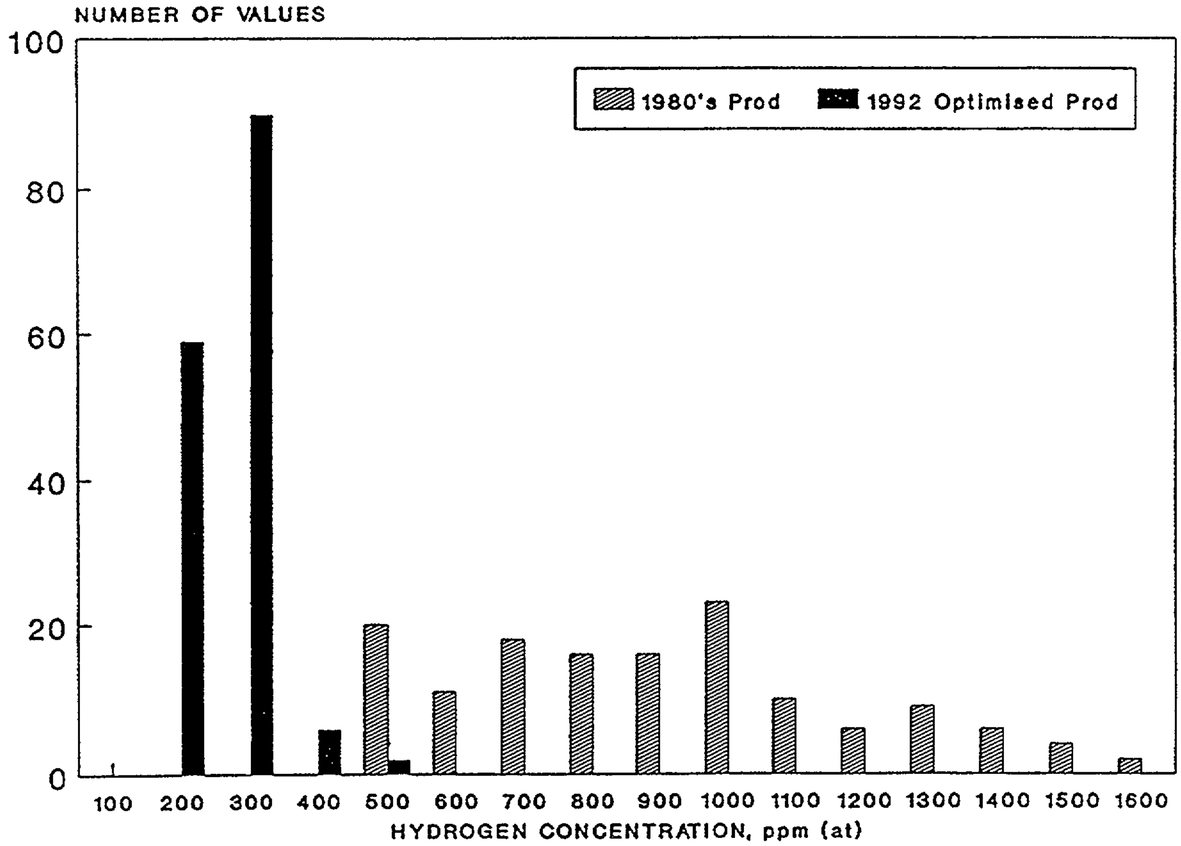


FIG. 14. Hydrogen concentration in fabricated pressure tubes [5] (1 ppm is 100 ppm (at)).

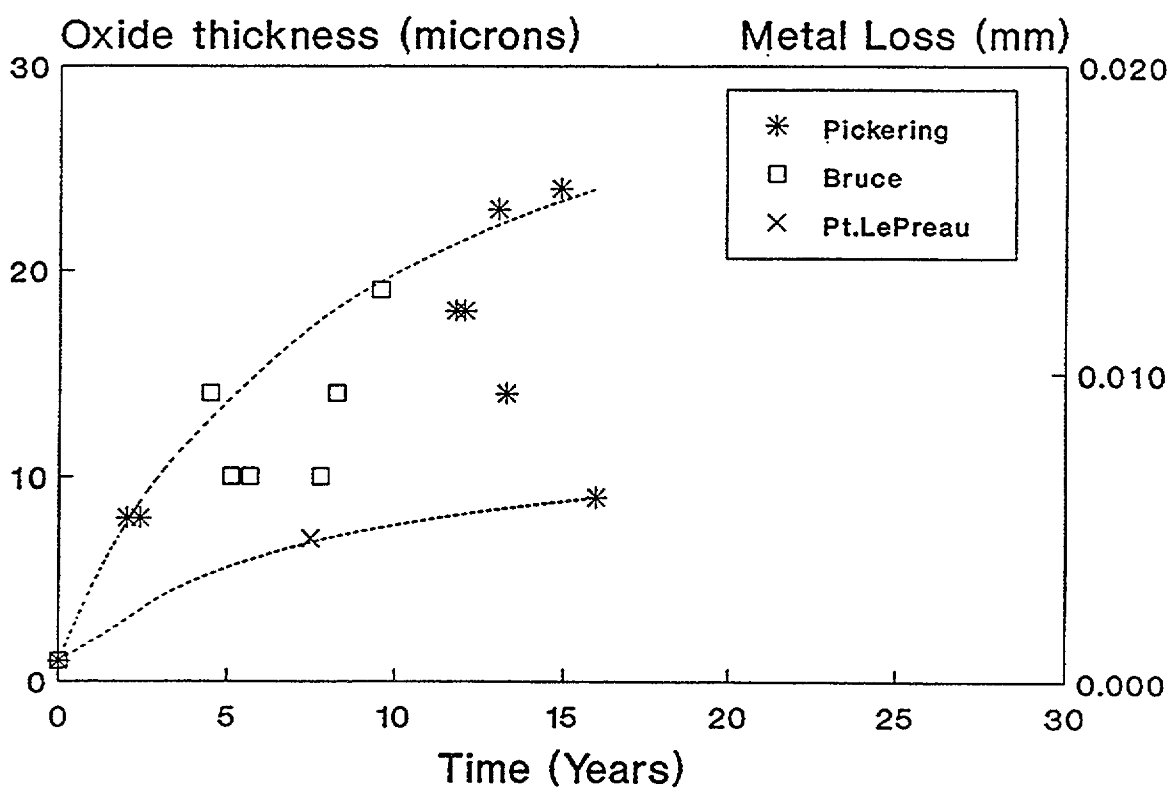


FIG. 15. Increase in oxide thickness on the inside of Pickering units 3 and 4, Bruce and Point Lepreau pressure tubes during service [3].

4.1.2. Deuterium ingress

During reactor operation, the heavy water flowing through pressure tubes slowly corrodes their inside surface and increases their oxide thickness [3, 15], as illustrated in Figure 15. The products of this chemical reaction are ZrO_2 and D. The loss of metal from this reaction is very small and does not limit pressure tube life, however Zircaloy-2 pressure tubes absorb such a large percent of the D produced by this corrosion, with the pickup rate increasing as the tube's oxide thickness increases, so that it is considered they cannot be safely operated for 30 years [2]. In contrast, Zr-2.5%Nb pressure tubes absorb only about 5% of the D produced from corrosion by the heavy water coolant flowing through them. In addition, the microstructural changes that occur during irradiation make the in-reactor oxidation resistance of Zr-2.5%Nb much superior to that of Zircaloy-2 [16].

D ingress into the pressure tubes in CANDU reactors is currently being monitored by removing a sliver of metal from operating tubes [17, 18], as illustrated in Figure 16, and by the periodic removal of pressure tubes in lead units for surveillance purposes. Figure 17 shows

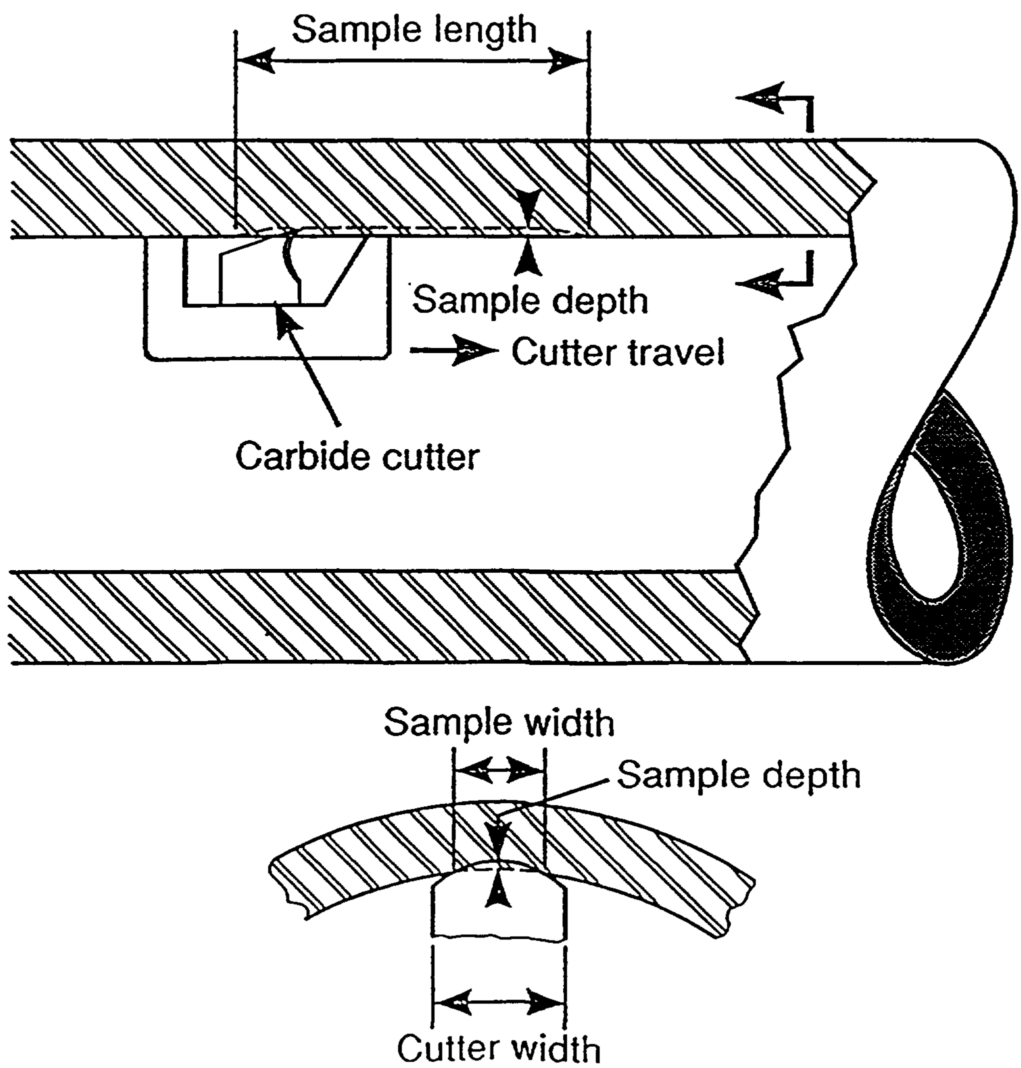


FIG. 16. Schematic illustration of a new inspection tool that obtains a sliver of metal from an operating pressure tube in order to measure its deuterium content.

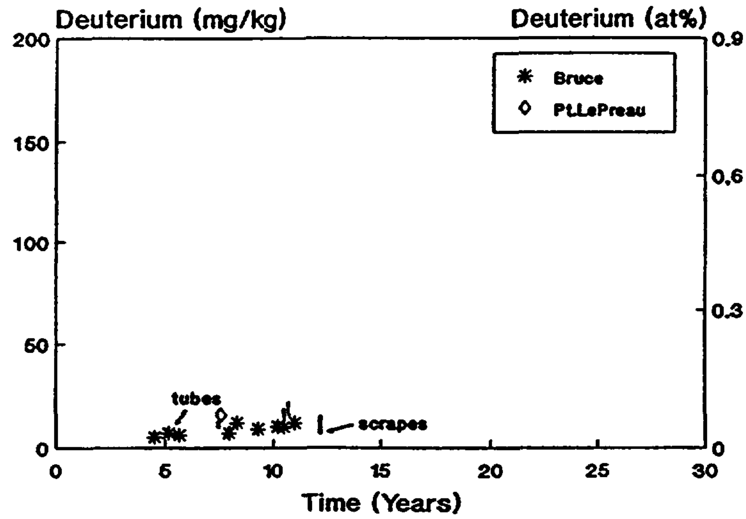


FIG. 17. Increase in deuterium concentration at the hottest end of Bruce and Point Lepreau pressure tubes [3].

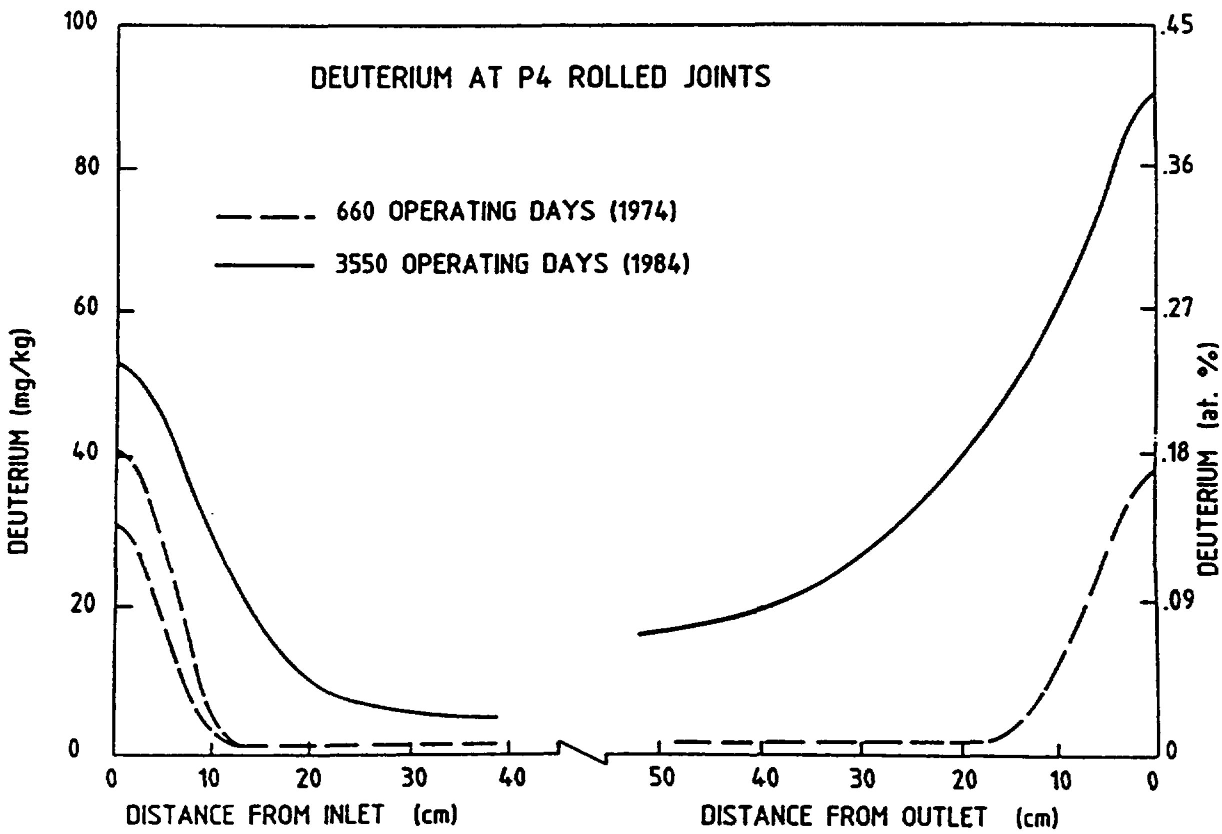


FIG. 18. Deuterium concentration along Zr-2.5%Nb pressure tubes removed from Pickering [3].

that during reactor operation the peak D pickup rate for Zr-2.5%Nb pressure tubes, which occurs at their hottest end, has been approximately constant with a maximum value of about 2 ppm per year (an Heq of 1 ppm per year) [3]. This rate of D ingress is much lower than the corresponding value for Zircaloy-2. However, since no tubes have yet operated for their 30 year design life, it is desirable to monitor the D ingress rate of the tubes in lead units to confirm that the D ingress for Zr-2.5%Nb continues to be small.

The data from ongoing measurements of D concentration is needed not only to give an early warning if tubes have significantly higher ingress than is anticipated, but also to evaluate the fitness-for-service of any pressure tube flaws that may be found.

In addition to absorbing D directly from the heavy water coolant, some pressure tubes in early CANDU units have also absorbed H/D from the gas in the annulus between each tube and the surrounding calandria tube [19]. To maintain the oxide condition on the pressure tube outer surface, which provides an effective barrier to the entry of H/D from the annulus gas, it is important to use an annulus gas that is sufficiently oxidising (dry $C0_2$ with added 0_2), and to have a continuous flow of this gas through all channel annuli.

As illustrated in Figure 18, the D ingress rate is higher at the rolled joint region of pressure tubes compared to the body of the tube, because of additional H/D ingress from galvanic reactions in the crevice between the pressure tube and the end fitting [3]. This results in additional ingress of H/D into the tube ends [20]. The ingress rate is higher at the outlet ends than at the inlet ends because the higher temperature permits faster D migration.

An empirical model has been developed, based on data from the early reactors, to predict the Heq profiles near rolled joints [20]. It predicts that near both inlet and outlet rolled joints, pressure tubes will reach TSS before completing their 30 year design life. The inlet rolled joints will reach TSS before the outlet end due to the lower inlet temperature. As this model is based on a limited database, it is important to obtain additional rolled joint ingress data from various reactors.

4.1.3. Hydride buildup

The main concern associated with the increasing Heq concentration in pressure tubes is that when Heq exceeds TSS at operating temperature, hydrides can exist during reactor operation, and can accumulate to initiate DHC.

In addition to keeping the overall H/D concentration of pressure tubes as low as possible, it is also important to avoid conditions that will cause local concentrations of the H/D in a tube. All DHC problems for pressure tubes have been associated with the local accumulation of H/D, causing hydrides to locally form and crack by DHC [2, 3]. Such local accumulation of hydrides has occurred either at large tensile stress concentrations or at a "cold spot", where a hot pressure tube has sagged into contact with the cooler calandria tube.

(a) Hydride blisters

In general, DHC in a pressure tube will result in leakage, not rupture. This LBB behaviour has been demonstrated at early Pickering [21] and Bruce units. However, if a pressure tube has sagged into contact with the cooler calandria tube surrounding it, because its garter spring spacers are not at their design positions, a significant temperature gradient will occur through its wall thickness. Brittle hydride "blisters" can then develop in the pressure tube at the contact location, and create a sufficiently long crack extending only part of the way through the tube wall thickness so LBB may not occur.

Some pressure tubes in the early reactors have sagged into contact with the cooler calandria tube because their loose-fit fuel channel spacers were displaced from their design positions during reactor construction and commissioning. In 1983, the Zircaloy-2 pressure tube in fuel channel G16 at Pickering unit 2 ruptured without prior detectable leakage [22, 23]. It was found that the failure of this pressure tube was associated with a combination of factors:

(i) A larger than anticipated pickup rate of D, both from the primary coolant and annulus gas system, to levels significantly exceeding TSS.

(ii) Movement of the outlet end garter spring had permitted contact between the pressure and calandria tube. This resulted in a significant thermal gradient in the pressure tube and H/D migration towards its cooler outer surface at the contact location.

(iii) Hydride started to form when the local Heq concentration in the contact region of the pressure tube exceeded TSS for the temperature of this region. The size of this hydride increased because H/D continued to migrate to the cold spot of the pressure tube outer surface.

(iv) The combination of the hoop stress from the tube internal pressure and the expansion stress associated with growth of the hydride blister at its contact location caused an axial crack to initiate from the blister. Due to the large temperature gradient in the wall of this pressure tube, any H/D that remained near its inner surface was in solution. Hence this crack could not propagate to the tube inside surface by the DHC mechanism.

(v) Axial propagation by DHC of the partial through-wall crack continued until fast fracture of the pressure tube occurred. The remaining thin web of material on the inside surface then broke and there was a sudden extension of the through-wall crack to a length of about 2 metres.

There were two primary causes of this Pickering pressure tube rupture. First was a high H/D level in the Zircaloy-2 tube, a result of accelerated corrosion and D ingress for this zirconium alloy. The second was a significant displacement of one annulus spacer. Both of these causes have been corrected in the more recent reactors.

All CANDU reactors built after Pickering units 1 and 2 have used Zr-2.5%Nb pressure tubes instead of Zircaloy-2 tubes. Examination of several Zr-2.5%Nb pressure tubes has shown that they do not exhibit the accelerated D pickup rate of Zircaloy-2 [2]. No blisters have yet been observed in Zr-2.5%Nb tubes, although hydride accumulations were observed in a few of the pressure tubes that had been contacting calandria tubes prior to their removal from Pickering unit 4.

All CANDU fuel channels fabricated in the past decade have used an improved spacer design, as indicated in Table IV. Because the initial spacer design was a loose-fit around the pressure tube, as illustrated in Figure 7, it could be displaced from its design position by the vibration experienced by empty pressure tubes during construction and commissioning activities. (After the pressure tubes were filled with fuel and heavy water, the "garter spring" spacers became pinched between the pressure and calandria tubes and could no longer be displaced.) If spacers were sufficiently displaced, contact between a pressure tube and the cooler calandria tube surrounding it could result from pressure tube sag. The design of spacers was improved by changing the material used to fabricate them, from a zirconium alloy to Inconel X750, which allowed the spacer design to be a tight fit around the pressure tube, as illustrated in Figure 7. In fuel channels with four spacers of this newer design, it is predicted that contact between pressure and calandria tubes will not occur for much longer than 30 years.

When the problem of spacer movement was detected in 1983, five CANDU units whose channels were fully assembled had not yet gone critical. The spacers in these fuel channels, that had been displaced during construction and commissioning activities, were relocated in 1984 prior to reactor startup. A similar SLAR (spacer location and relocation) process has been developed for relocating displaced spacers at the early operating units. SLAR is planned to be performed for all operating CANDU reactors which have four loose-fit spacers in their fuel channels, except the five units whose spacers were relocated in 1984 before they went critical.

The maintenance strategy for early CANDU reactors is thus currently focused on preventing blister formation. This requires inspection of all fuel channels prior to a conservatively predicted point in time when there may be sufficient H/D in their pressure tubes for blister formation/growth to occur. Any spacers that are significantly displaced from their design positions are identified and then sufficiently relocated to ensure their pressure and calandria tubes remain separated throughout the remainder of their 30 year design life. This is a very conservative maintenance strategy for which there is a large safety margin associated with the possibility of a pressure tube rupture. It is based on assessments performed using upper bound values for the H/D content of tubes to show that blister formation is prevented. It takes many years after blister formation before the rupture of a Zr-2.5%Nb pressure tube due to DHC occurs. After spacer relocation, the residual/remaining life of all CANDU fuel channels that now have four potentially displaced spacers is expected to be sufficient to allow them to reliably achieve their 30 year design life.

In early reactors where fuel channels had only two spacers, pressure tube to calandria tube contact cannot be avoided during the last 10 to 15 years of their 30 year design life. Therefore, these reactors will have their initial pressure tubes replaced before they have operated for 30 years. The first four Pickering units have already had all of their initial fuel channels replaced, which included replacing the two original loose-fit spacers in each channel with four improved tight-fit spacers. This has extended the life of these reactor cores 10 to 15 years beyond their original 30 year design life.

(b) Rolled joints

The rolled joint is usually the area of the pressure tube most at risk to DHC due to its higher D ingress and higher tensile stress than any other part of the tube. The potential for DHC in the rolled joint area of pressure tubes was illustrated in 1974/5 when through-wall cracks were created by DHC near the rolled joints of some early Zr-2.5%Nb pressure tubes (primarily in Pickering units 3 and 4) and caused leakage of primary coolant into the annulus gas system [21]. It was found that a very high tensile residual stress had been caused by

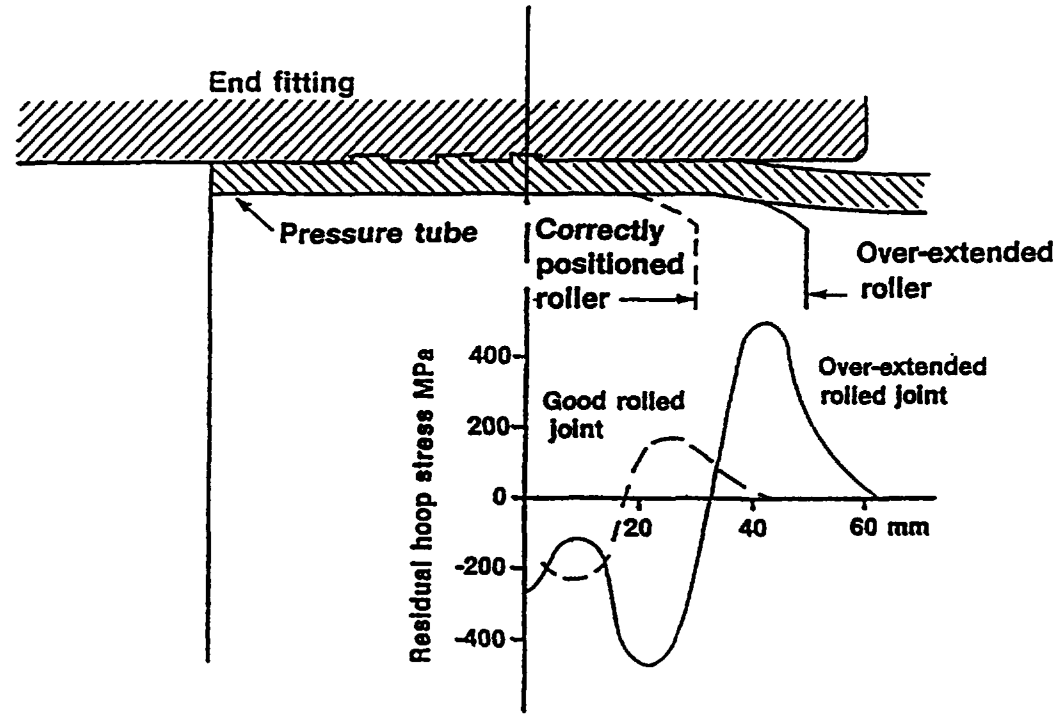


FIG. 19. Residual stress in rolled joints.

improper positioning of the rollers used to make the rolled joints in these early units, as illustrated in Figure 19. The rolling procedure permitted the roll expander to be inserted too far into the pressure tube, and expanded the tubes where they were not supported by the end fitting. DHC occurred during cold shutdowns in the first few years of operation for Pickering units 3 and 4, but did not occur when the units were operating as the tubes did not contain enough H/D for hydrides to exist when they were hot. In 1974/5, about 70 tubes were replaced in Pickering units 3 and 4 because they had cracked near their rolled joints [24]. Because stress relaxation during service prevented most further crack initiation, only one additional tube leak occurred in these units, in 1985. This was also a LBB failure which allowed the unit to be safely shut down for tube replacement.

Examination of various cracked tubes showed that short axial cracks had initiated by the DHC mechanism where a very high tensile residual stress existed in the portion of the pressure tube immediately adjacent to the rolled joint. The H/D in that portion of the tube had concentrated and formed hydrides, which were then cracked by this high stress. The axial length of these cracks when they first penetrated through the tube wall and started to leak was about four times the tube wall thickness.

When this problem of excessive rolled joint residual stresses was discovered, fuel channels for two of the early Bruce units had been installed, but not operated. To prevent early life DHC, all joints in these units were given a pre-service stress relief. However, two Bruce tubes leaked after a few years of operation because cracking had started before the stress relief was done. Again LBB was exhibited and the unit with the leaking tubes was safely shut down for tube replacement. The axial length of these cracks when they first started to leak was a bit longer than for the Pickering cracks mentioned above, probably because the Bruce tubes were operated for a longer period with partial thickness cracks and they became partially plugged with oxide.

More recent reactors do not have this high tensile residual stress problem because improvements were made both in the rolling procedure, to avoid improper positioning of the rollers, and in the rolled joint design, to make it less sensitive to roller position and by decreasing the clearance between the pressure tube and the end fitting bore. A shrink fit is now produced by thermally expanding the end of the end fitting to permit insertion of the pressure tube before roll expansion of the pressure tube into the end fitting.

The pressure tube cracking experience outlined above emphasizes the importance of having the annulus gas system provide a sensitive detection of leakage, so that LBB limits the consequences of pressure tube cracking. To prevent DHC from initiating a tube rupture, it is essential both that DHC in pressure tubes be a low probability event, and that a defence in depth be provided by assuring a high probability of LBB.

(c) Pressure tube flaws/defects

Since H/D may concentrate at large tensile stress concentrations and precipitate hydrides that initiate DHC, it is important to eliminate (or at least minimize) any potential causes of significant pressure tube tensile stress concentrations, such as flaws and defects created during tube manufacture, installation, commissioning or operation.

(i) Manufacturing flaws

A few tubes in early CANDU units have had a significant tensile stress concentration because they contained a sharp manufacturing defect. Two pressure tubes in Bruce unit 2 leaked as a result of DHC that initiated at laminations which had escaped detection during the extensive inspections performed by the tube manufacturer [25]. In 1986, channel N06 of Bruce unit 2 ruptured during a cold pressurization test after the unit had been shut down because of detection of leakage at a manufacturing defect that had propagated through-wall by DHC. A subsequent review of the pressure tube manufacturing process concluded that top-of-ingot tubes produced from ingots that contained deep shrinkage cavities may contain lamination flaws.

Even though these lamination flaws had a long axial length, they were very difficult to detect by standard non-destructive inspection techniques, and also difficult to see metallographically. Hence these flaws could escape detection during the type of manufacturing inspections that were previously performed. They were prone to opening up during service by an oxidation process and to subsequent DHC initiation.

The manufacturing process has been changed to minimize the size of the shrinkage cavities in ingots and the tops of ingots are now cropped to remove them. Final tube inspection by new techniques with a very high probability of detecting such laminations significantly lowers the risk of having such defects in tubes now entering service.

Following the B2N06 failure, comprehensive programmes were initiated to assess the potential for lamination flaws in tubes in the operating reactors. Because tubes produced from top-of-ingot material have the highest risk of obtaining lamination defects, such tubes in all of the operating CANDU units were inspected. In addition, archive offcuts from the top tube for each ingot (or the second tube if the top tube had been rejected) were subjected to a high frequency normal beam ultrasonic examination, and/or were examined metallographically. A few similar flaws were discovered by these post-installation inspections, and these tubes were replaced.

(ii) Service induced damage

In-service wear of pressure tubes in CANDU reactors can be caused by fuel bundle scratching during refuelling, bearing pad fretting caused by fuel pencil vibration and fuel bundle rocking, crevice corrosion at fuel bundle bearing pad positions, and debris fretting. Pressure tube Fitness-for-Service Guidelines are being developed to provide criteria and assessment methodologies for addressing all such wear damage [26]. Also, research and development programmes are in place that are producing the pressure tube flaw tolerance data that will be needed for assessing the in-service wear of tubes when they are near the end of their design life. The following addresses each of the pressure tube service induced damage mechanisms and identifies those that have a potential for affecting pressure tube integrity.

(1) Refuelling scratches

Examination of many fuelling scratches in the pressure tubes removed from the lead operating CANDU units has shown that these scratches are shallow (less than 30 microns) and rounded. These are not a concern for pressure tube integrity.

(2) Fuel fretting

The examination of fuel bundle bearing pad fret marks in tubes removed from operating units has shown them to be very shallow (<55 Tm). "Abnormal fuel support" conditions which cause deeper fret marks at the inlet ends of the Bruce and Darlington reactors are associated with features of the fuel handling system used only for these units, so this type of pressure tube fretting does not occur in any other CANDU fuel channels. Thus, except in the Bruce/Darlington reactors, pressure tube fretting caused by fuel bearing pads is not a concern.

(3) Crevice corrosion

Crevice corrosion has been observed at fuel bundle bearing pad positions in operating pressure tubes, towards their outlet end. This corrosion is due to LiOH concentrating under localized boiling conditions which exist between the fuel bearing pads and pressure tubes. Experimental evidence shows that the damage in terms of depth is self limiting. Metallographic examination of the crevice corrosion marks in tubes removed from the Pickering reactors has shown them to be shallow and wide with a maximum depth of about 200-300 microns. As these flaws are self limiting in depth, they are considered to not be a concern for pressure tube integrity, with the periodic inspection programme (PIP) being adequate for monitoring them.

(4) Debris fretting

Debris fretting is a known in-service flaw mechanism in CANDU reactors that can cause pressure tube flaws with depths exceeding the acceptance level (0.15mm) of CSA Standard N285.4. As indicated in Figure 20, debris marks primarily occur during early operation, because construction debris such as metal turnings left in the primary heat transport system (PHTS) can get trapped in the fuel and cause pressure tube fretting damage [3]. The amount of debris that could

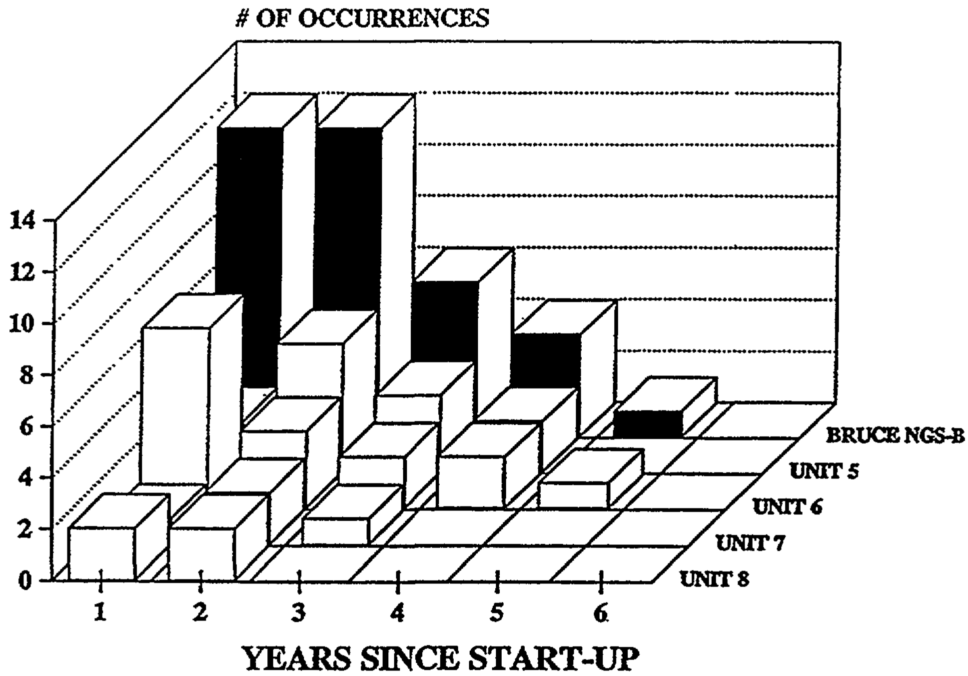


FIG. 20. Occurrences of fuel with debris damage in Bruce 'B' reactors [3].

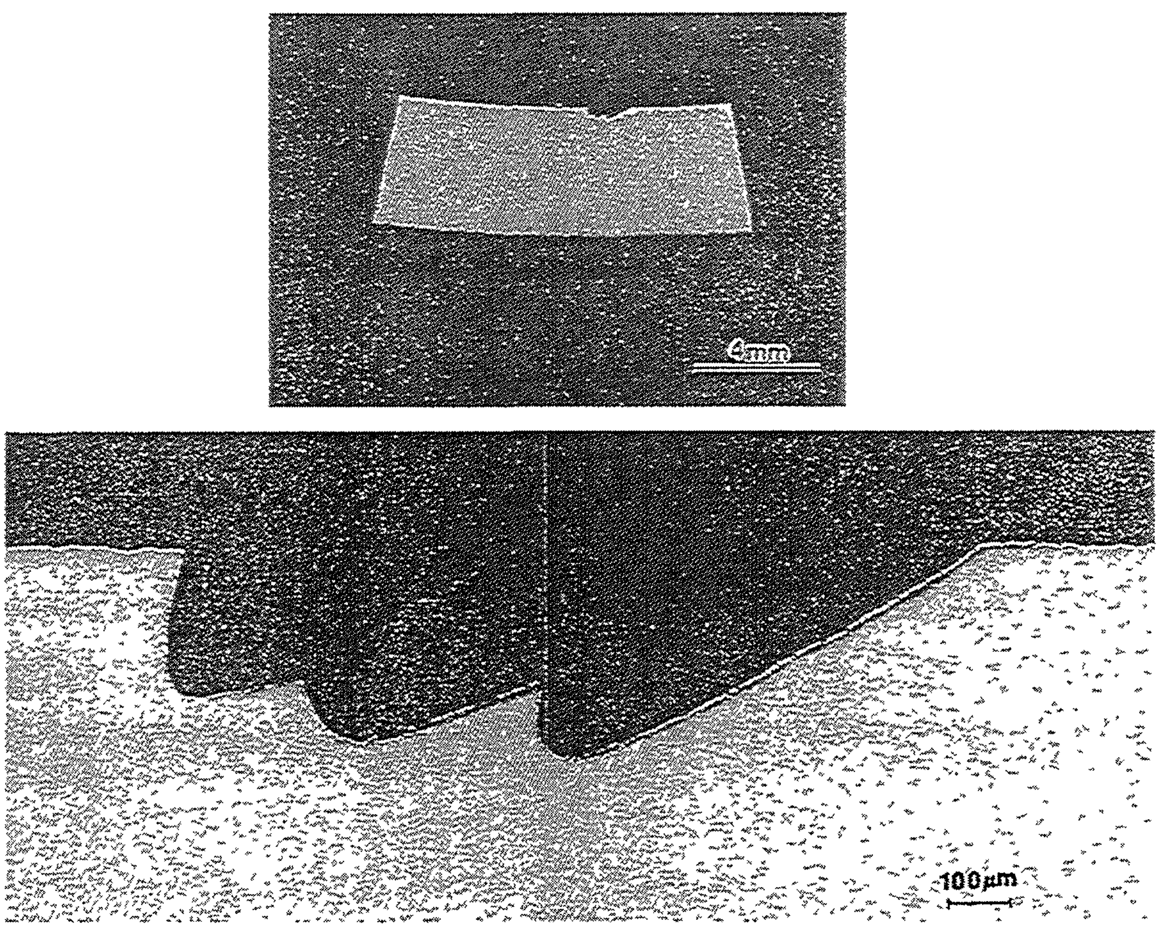


FIG. 21. Transverse section of a flaw in the pressure tube removed from a Bruce unit 1 channel [3].

lead to fretting damage can be minimized during construction by a high level of cleanliness, and during commissioning by the use of strainers at the inlet end of channels.

Channels which have been subjected to significant pressure tube debris fretting are believed to have also exhibited fretting of fuel bundle sheaths, resulting in release of fission products to the PHTS. Therefore, records of failed fuel and visual examination of suspect bundles can be used to identify channels with the highest potential for pressure tube debris damage.

The majority of debris fretting damage observed to date in CANDU pressure tubes has been localized, shallow (less than 0.5 mm deep), blunt notches with low stress concentrations [3], as illustrated in Figure 21. A few exceptions of much deeper flaws (up to 1.6 mm deep) have occurred in early reactors and tubes with these flaws have been removed for destructive examination. These examinations have shown no evidence of crack initiation and, in most cases, no evidence of hydride formation or reorientation, which implies peak stresses were less than 180 MPa. Growth of cracks from these flaws is considered to be very unlikely as the stress concentration at the flaw tips is relatively low, and these flaws are located in the body of the tube where the H/D concentration is predicted to remain below that required to precipitate hydrides at operating temperature during the 30 year design life for pressure tubes. Although hydrides would be expected to form when such tubes are cold, these hydrides would dissolve at operating temperature. The intent of ongoing and future testing is to establish the maximum depth of debris frets that will not lead to DHC, and therefore can be tolerated without pressure tube replacement.

In case DHC initiates at a debris flaw, a sensitive leak detection system is required, so LBB continues to provide a defence in depth limit for the consequences of any pressure tube cracking that may occur.

4.2. PRESSURE TUBE IRRADIATION ENHANCED DEFORMATION

The hexagonal close packed crystal structure of zirconium results in anisotropic deformation and in neutron irradiation causing both irradiation growth (shape change at constant volume in the absence of an applied stress) and irradiation creep (stress dependent constant volume shape change) [27]. The deformation of operating pressure tubes (sag, diameter increase, elongation and wall thickness decrease) is predicted to be due to the addition of strains associated with thermal creep, irradiation creep and irradiation growth with the strain rate for each of these being a function of temperature, flux, dislocation density, stress, etc. [28]. The design of fuel channels associated with the initial construction of all CANDU and Indian reactors includes allowances for the amount of pressure tube deformation expected to occur during a 30 year operating life.

Early deformation predictions were not large enough because it was not known that irradiation significantly enhances the deformation rate for operating pressure tubes compared to that obtained in out-reactor creep tests. Therefore the early CANDU and Indian reactors did not have adequate design allowances for 30 years of deformation by their initial pressure tubes.

The consequences of pressure tube irradiation enhanced deformation being larger than anticipated are additional inspection outages to monitor the deformation, special maintenance to maximize the deformation allowances, and possibly tube replacement before it has operated for 30 years as the allowances for deformation are finite.

Equations based on periodic measurements taken for the pressure tubes in the initial Pickering and Bruce reactors, as well as on research data and theoretical considerations, have now been developed to predict in-reactor pressure tube deformation. These equations are used to design CANDU fuel channels that will accommodate 30 years of pressure tube deformation. However, since no CANDU reactor has yet operated for 30 years, it is important to maintain material testing to high fluences in fast flux facilities of test reactors, and to continue monitoring the dimensional changes in operating tubes, particularly those in lead units, to ensure it remains as predicted. Ongoing test and monitoring programmes will confirm that pressure tubes behave as anticipated, or provide lead time to manage their deformation.

The four types of pressure tube deformation (sag, elongation, diameter increase and wall thinning) are discussed below.

4.2.1. Sag

During service, the horizontal pressure tubes sag between the spacers that separate each pressure tube from the surrounding calandria tube. The use of four spacers in later units, as indicated in Tables II and IV, should ensure that pressure tubes do not contact calandria tubes for well in excess of the fuel channel 30 year design life.

In addition to the sagging of pressure tubes between the fuel channel spacers, the entire fuel channel assembly also sags during service and its curvature increases with service time. Although this curvature will not impede the passage of fuel bundles, most CANDU reactors have horizontal reactivity control mechanisms located at right angles to the fuel channels such that some channels may eventually sag into contact with these mechanisms, as illustrated in Figure 10. Although this is not expected to occur during the 30 year fuel channel design life, monitoring of the gap between fuel channels and mechanisms is planned, in the last half of the design life for lead units, to ensure that contact does not occur sooner than expected. If necessary, remedial action is available, which would likely involve moving the mechanisms down to increase the gap between them and the calandria tubes above them.

4.2.2. Elongation

As shown in Figure 22, pressure tube elongation is essentially a linear function of time [3]. The peak rate of elongation is about 5 mm/year [3, 29], which is much faster than was anticipated during the design of the early reactors. Because of this, the first seven commercial CANDU units, and the first seven Indian units, do not have sufficient fuel channel bearing length to accommodate all of the pressure tube elongation expected during 30 years of operation. All other CANDU units have at least a 75 mm bearing length at both ends of their fuel channels, plus feeder pipe spacings and fuelling machine offsets, etc., which are adequate to accommodate the 30 year elongation of their pressure tubes.

The fuel channels for the first seven commercial CANDU reactors, and the first seven Indian reactors, have required an ongoing programme of axial repositioning so that they remain supported by their bearings as they elongate.

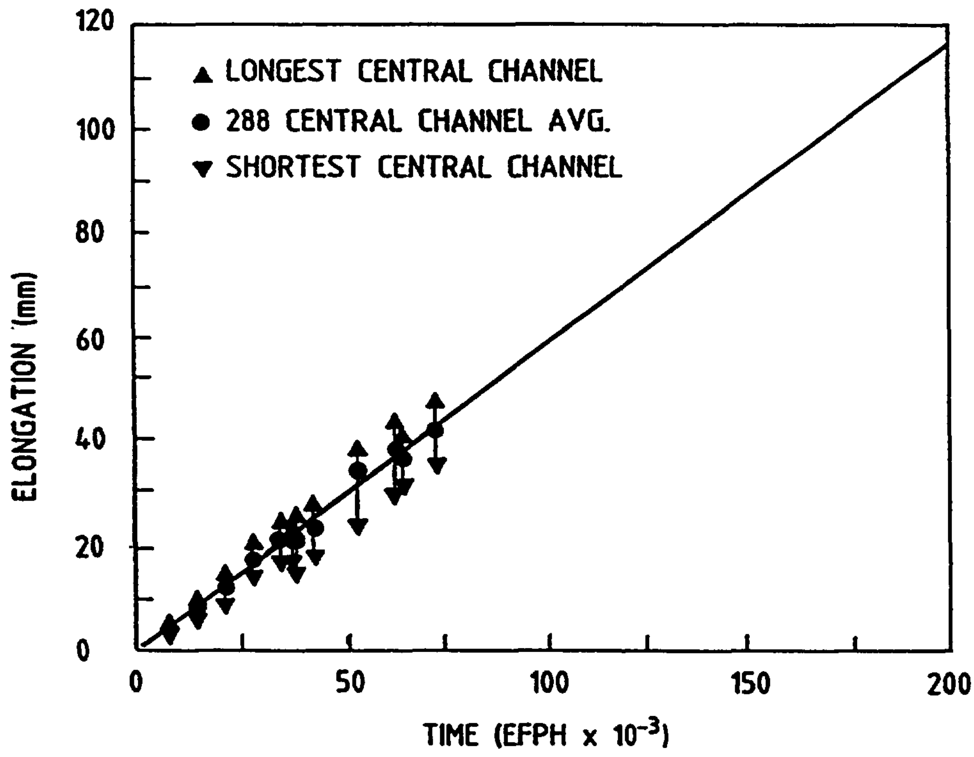


FIG. 22. Elongation of Bruce 'A' pressure tubes as a function of time [3].

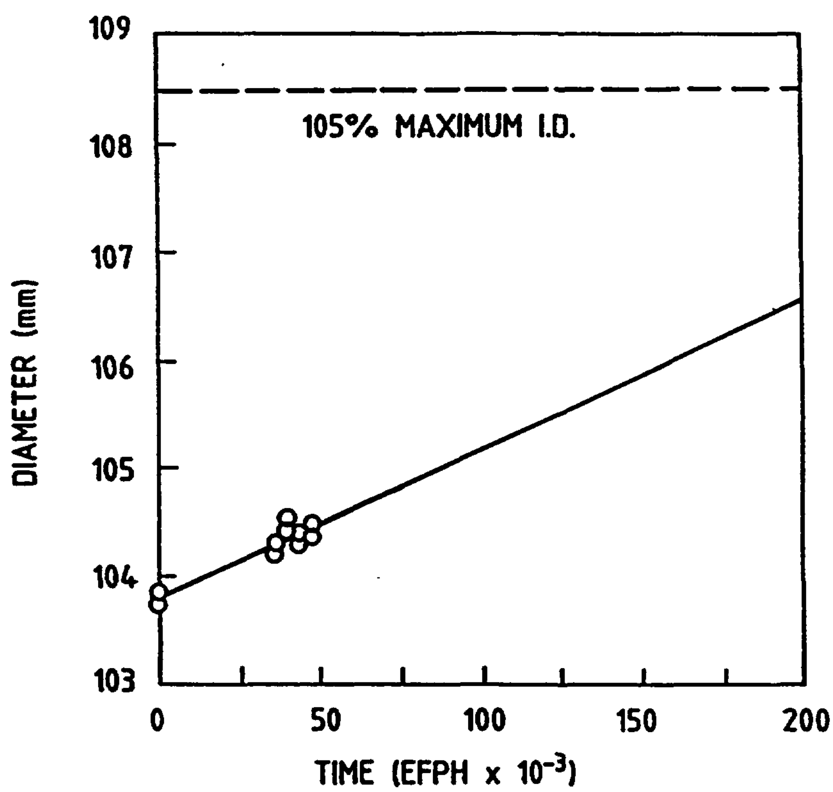


FIG. 23. Increase in diameter of Bruce 'A' pressure tubes [3].

Because there is a large variation in tube-to-tube elongation rates, the in-reactor elongation rates of all tubes in each unit are monitored. This inspection provides accurate elongation rates for each tube, which can be used to establish the correct timing for the mid-life channel reconfiguration (i.e., releasing the initially fixed end of the fuel channels and fixing the initially free end), help identify the possible need for feeder-to-feeder interference checks, and provide lead time for planning corrective action if elongation becomes a life limiting issue for the channels with the fastest elongation rates. There are design solutions available that could be considered to manage this if it occurs, such as defuelling the fastest elongating pressure tubes and then appropriately limiting the flow in these empty channels.

4.2.3. Diametral expansion

Diametral expansion of the pressure tube allows an increasing amount of primary coolant to flow around the fuel bundles, which slightly reduces the critical channel power at constant flow. Although this is offset by an overall increase in flow, and the redistribution of flow from the lower power channels to the higher power ones, it will eventually result in unacceptable fuel cooling. The design allowance for the diametral expansion of pressure tubes is conservatively limited to 5% of the initial pressure tube diameter. This provides a large margin to avoid pressure tube creep rupture [30], and ensures that the garter spring spacers are not squeezed between pressure and calandria tubes. Figure 23 illustrates that the diametral expansion rate of pressure tubes is only about 0.1 mm per year [3].

The size and number of annulus spacers associated with each fuel channel were changed from the values defined for the initial Pickering and Bruce reactors. More (4 vs. 2) smaller diameter spacers have been used in the later fuel channel designs to maximize the space in the fuel channel annulus gap before a garter spring would become squeezed between a pressure and calandria tube due to diametral expansion of the pressure tube.

4.2.4. Wall thinning

The reduction in wall thickness of pressure tubes during operation has been measured in the lead operating reactors and has been found to be well within the value assumed in the design analyses. As a result, this is not expected to be a life limiting issue for any reactor. The periodic inspection programme (PIP) is adequate to monitor this.

4.3. CHANGES IN PRESSURE TUBE MATERIAL PROPERTIES

It is known that irradiation increases tube hardness, yield and tensile strengths, and reduces ductility and fracture toughness. In addition, the susceptibility of Zr-2.5%Nb to DHC increases slightly, and the velocity of DHC increases somewhat, particularly at the inlet end due to its lower irradiation temperature [31]. The consequences of such changes is that pressure tubes become more susceptible to fracture, that is, the margins associated with demonstrating their LBB behaviour are decreased and hence the possibility of tube rupture is increased. Changes in tube properties must not result in an unacceptable safety margin on tube rupture.

Tests on material from tubes removed from the initial Pickering and Bruce units have shown that irradiation damage has essentially saturated at fluence levels of approximately 1×10^{25} n/m^2 and no further change has been seen up to a fluence of 1×10^{26} n/m^2 [3, 32], as

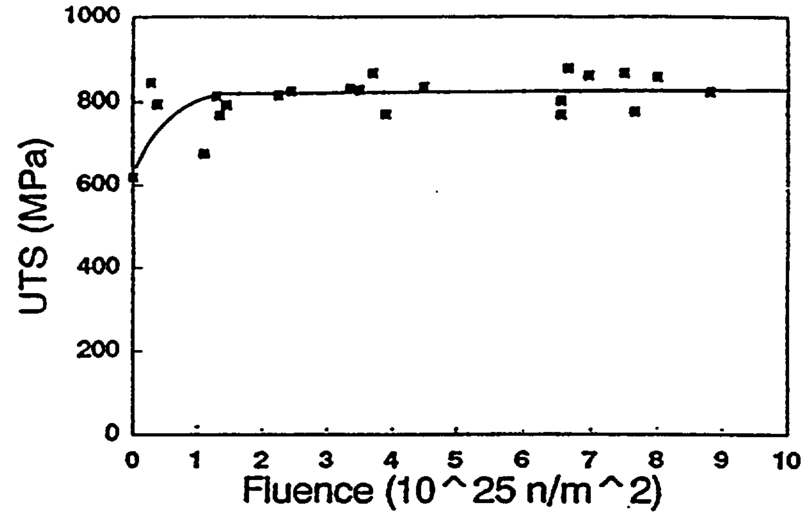


FIG. 24. Irradiation hardening of pressure tube UTS [3].

illustrated in Figure 24. Since this saturation of the effect of irradiation on the material properties of Zr-2.5%Nb pressure tubes occurs during the first few years of operation, the values of pressure tube properties after 30 years of irradiation are expected to be adequate to allow Zr-2.5%Nb tubes to achieve their 30 year design life. Test data from the lead operating CANDU reactors, and material testing to high fluences in fast flux facilities of test reactors, will confirm whether or not fracture toughness and DHC velocity values, etc., remain at their saturated values. In addition, pressure tube Fitness-for-Service Guidelines are being developed to provide criteria and assessment methodologies for addressing all aspects related to pressure tube flaw tolerance and how LBB provides a defence in depth limit for the consequences of any pressure tube DHC that may occur [26].

New pressure tubes have a very large margin between their leaking and breaking conditions. However, as they are operated this margin decreases due to irradiation and D ingress. As a pressure tube's operating life increases, there is an increasing concern that a sufficiently large flaw could exist to precipitate hydrides and initiate DHC. In order for tubes to continue operating, it is required that assessments be done to show there continues to be a high confidence both that DHC will not occur, and also that if DHC does occur the result would be a leak that is detected and the reactor safely shut down before this cracking becomes unstable. To ensure this LBB requirement is satisfied, there must be a high confidence level that the time *required* for a crack to grow to an unstable length is less than the time *available* to take appropriate action after leakage is detected. The annulus gas system of CANDU reactors provides a very sensitive detection system for pressure tube leaks, which allows this LBB requirement to be satisfied throughout the 30 year design life for pressure tubes [9, 10]. A corollary of this LBB requirement is that conditions which do not satisfy LBB, such as having a very high H/D concentration in a portion of a pressure tube, must not occur. Therefore, pressure and calandria tube contact must not be allowed to exist long enough for a hydride blister to form and grow at the cold spot of the pressure tube. Since the D ingress rate for Zircaloy-2 pressure tubes is higher than that for Zr-2.5%Nb tubes, the margin on demonstrating LBB for Zircaloy-2 tubes becomes unacceptably small and requires that they be replaced before they have operated for 30 years. This was the reason all of the Zircaloy-2 tubes that were originally installed in Pickering units 1 and 2 were replaced with Zr-2.5%Nb tubes.

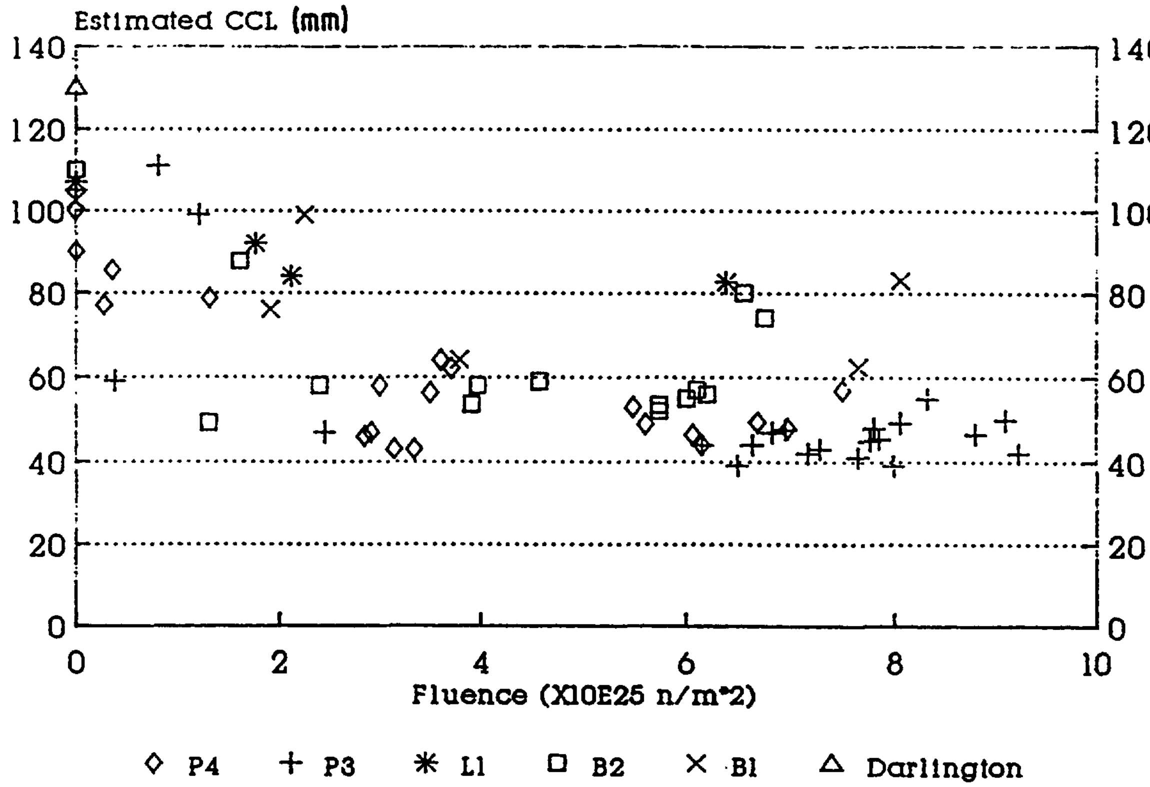


FIG. 25. Effect of irradiation on the estimated critical crack length of Zr-2.5%Nb pressure tubes [3].

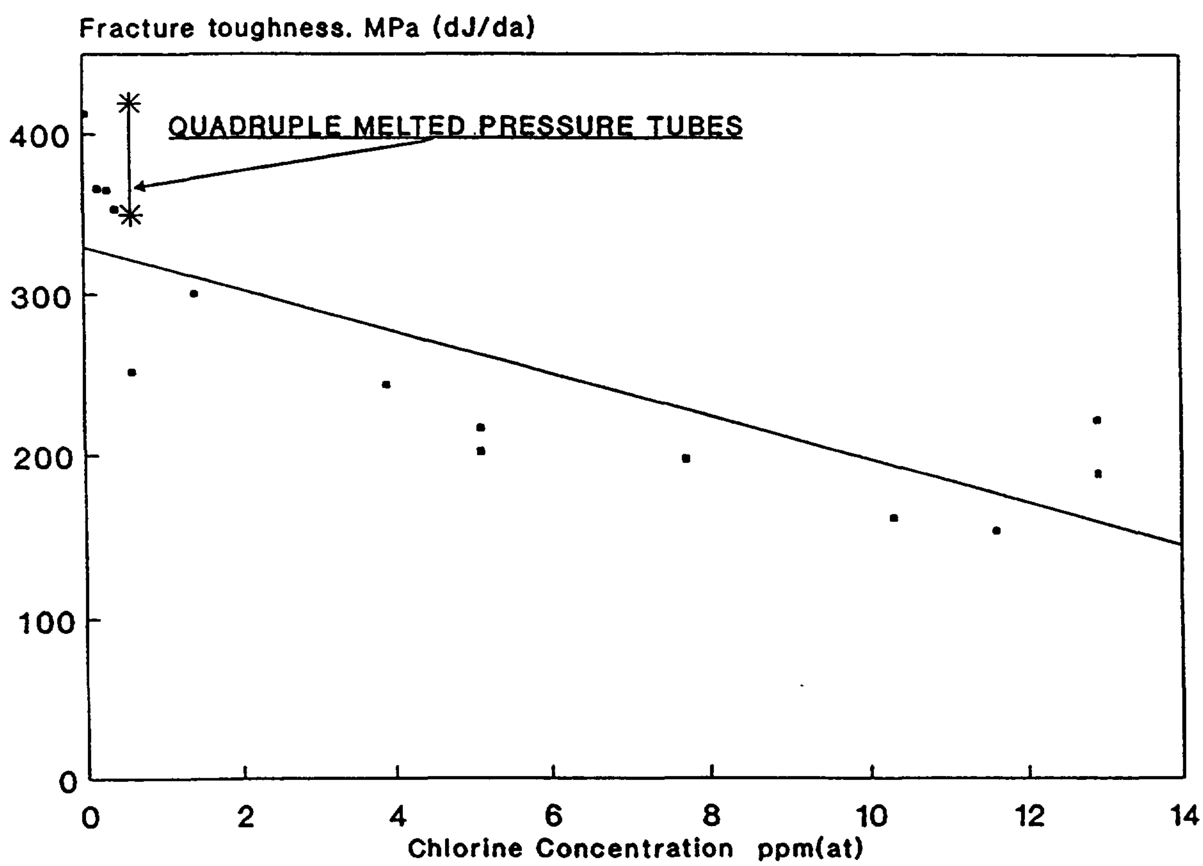


FIG. 26. Effect of chlorine concentration on Zr-2.5%Nb pressure tube fracture toughness [5].

Although degradation of the properties for Zr-2.5%Nb pressure tubes has not been responsible for any operating problems, an improvement in these properties is a highly desirable way to reduce pressure tube rupture concerns. If a crack grows undetected, it will become unstable when it reaches the critical crack length (CCL), which is controlled by the fracture toughness of the material containing the crack. As-fabricated pressure tubes are very tough, but irradiation damages the crystal lattice and changes the tube mechanical properties. There is an initial rapid decrease in toughness to about half its unirradiated value with very little subsequent change [3, 33], as illustrated in Figure 25. The toughness values for irradiated pressure tubes cover a very broad range, with some tubes still having quite a high toughness after 18 years of service [34–36]. Also, the tubes made most recently are tougher than earlier tubes, even though their nominal specification is identical. These observations led to studies to look for subtle microstructural or microchemical differences that may have caused these differences in toughness.

It was recently observed that tubes containing low concentrations of chlorine have high fracture toughness. Thus the range of fracture toughness is partly related to trace amounts of chlorine, which is a residue of the process used to refine zirconium. The fabrication records revealed that the very tough Zr-2.5%Nb tubes were made from 100% recycled material that had been melted four times. Such quadruple melting of ingots reduces the chlorine concentration to small values and consistently produces tough tubes [5, 13, 14, 36], as illustrated in Figure 26.

4.4. SUMMARY

Feedback from the operation of pressure tubes in early reactors and the knowledge gained through an extensive CANDU pressure tube research and development programme, has identified the key pressure tube degradation issues and the appropriate techniques for managing their ageing. The ageing effects that caused pressure tube problems which were not anticipated during the design of the initial CANDU fuel channels are:

- Irradiation enhanced deformation,
- Delayed hydride cracking, and
- Changes of pressure tube material properties.

The consequences of irradiation enhanced deformation are additional outages to monitor the deformation and to axially reposition channels, plus their life may be shortened since the design allowance for deformation is finite. The consequence of DHC (if pressure tube properties have not become unacceptable due to the effects of irradiation and D ingress) is leakage and reactor shutdown for tube replacement (LBB). The consequences of material property changes is that the probability of having Break-Before-Leak behaviour could become unacceptably large because a tube may not leak before it ruptures, or leakage from a crack may not be detected before the crack grows to the unstable length.

All CANDU and Indian reactors, except the earliest ones that were designed before the irradiation enhancement of pressure tube deformation was known, are expected to be able to accommodate 30 years of pressure tube deformation.

The material properties of Zr-2.5%Nb pressure tubes after 30 years of operation are expected to be acceptable, since these property values primarily change only during the first few years of irradiation, and then there is a saturation of the effect of irradiation on them.

Also, the D ingress rate for these tubes is believed to be low enough that it will not have an unacceptable effect on their properties.

As the solubility limit of H/D in zirconium is low, DHC is the main ageing issue for pressure tubes. As a result, an extensive CANDU pressure tube research and development programme is aimed at defining the most appropriate techniques for avoiding DHC in operating tubes.

To minimize the ingress of D from the coolant that flows inside pressure tubes, the coolant chemistry is carefully controlled, as well as the chemistry of the gas in the annulus between each pressure and calandria tube. It is also important to avoid conditions that cause local concentrations of H/D in a tube as the main DHC concerns have been associated with concentrations of H/D at:

- large tensile stress concentrations and
- large temperature gradients through the tube wall.

Tensile stress concentrations are minimized by ensuring significant flaws or defects are not created during tube manufacture, installation, commissioning or operation. Whenever a new type of pressure tube flaw or defect is identified as a significant threat to tube integrity, any existing tubes that may be at risk are inspected and an appropriate improvement is implemented for future tubes. For example, top-of-ingot solidification defects have been eliminated as a concern by a combination of improved manufacturing process controls in recent orders and inspection of tubes which were fabricated before these improvements were made.

Recent dispositions of indications detected in CANDU fuel channels, other than those associated with the fuel handling system used in Bruce/Darlington reactors, have been confined to fretting by debris in the primary heat transport system. Such debris, originating from construction and/or commissioning activities, can become trapped in the fuel and cause fretting damage to pressure tubes during their first year or two of operation. This mechanism for creating flaws usually produces blunt flaws that cause relatively small local stress concentrations in a few pressure tubes, and hence is not a significant concern. In case DHC does occasionally occur, it is important to have a sensitive leak detection system to ensure LBB continues to provide a defence in depth limit for the consequences of any pressure tube cracking.

The main concern associated with operating CANDU pressure tubes currently is the large temperature gradient that exists through the wall in some tubes of early reactors where the displacement of fuel channel spacers has allowed pressure tubes to come in contact with the cooler calandria tubes. This creates the potential for formation of brittle hydride blisters that may eventually cause a long through-wall crack without any prior leakage (break before leak). To avoid this possibility, all CANDU fuel channels that have four potentially displaced spacers need to have their spacers repositioned sufficiently to ensure pressure and calandria tubes remain separated throughout the latter portion of the channel's 30 year design life.

There is a high level of confidence that the Zr-2.5%Nb pressure tubes in all CANDU reactors, except a few early ones, have sufficient residual/remaining life that they will reliably operate for at least their design life of 30 years at 80% capacity, with a large safety margin on tube rupture. It is expected that the more recently fabricated fuel channels, which have Zr-2.5%Nb pressure tubes and four tight-fit spacers, will efficiently achieve their 30 year design life without any special maintenance.

5. INSPECTION AND MONITORING METHODS FOR PRESSURE TUBES

Monitoring of pressure tube performance in operating units at regular intervals is required to ensure that these tubes behave as anticipated, i.e., to provide assurance of pressure boundary integrity. A variety of special purpose in situ inspection techniques have been developed to accomplish this. The inspection data that is being obtained from operating units, combined with predictive models, give the needed confidence that the current generation of CANDU pressure tubes will perform effectively and safely throughout their 30 year design life.

Pressure tube inspection programmes are aimed at either demonstrating that tubes will continue to operate reliably or helping operating staff establish, as early as practical, the occurrence of any potentially unacceptable tube degradation, so that appropriate maintenance activities can be initiated to arrest or mitigate such degradation. In order to ensure that the most appropriate data is used to predict future pressure tube performance, much of the overall pressure tube inspection programme focuses on inspection of lead reactor units (Bruce unit 3, the CANDU-6 unit at Point Lepreau, etc.). Subsequent reactor units are inspected less frequently, using smaller sample sizes, to verify that their pressure tubes behave in a similar fashion, within expected ranges.

The Canadian Standards Association (CSA) Standard N285.4 describes a periodic inspection programme (PIP) which requires that a few pressure tubes in each unit be periodically inspected for flaws, dimensional changes, and D ingress. The PIP objective is to ensure the detection of any unexpected generic problem in order to keep the probability of pressure tube failure at an acceptably low level for the entire design life. This programme also includes the periodic removal of a surveillance tube from a lead unit for destructive examination in a laboratory to confirm that changes in tube mechanical and metallurgical properties (fracture and burst test results, oxide layer thickness, H/D distribution, etc.) are consistent with the predictions from the pressure tube research programme. It is mandatory for reactors operating in Canada to satisfy the PIP requirements described in CS-N285.4. A reactor operator may also perform additional in-service inspection (ISI) in order to supplement the monitoring of generic pressure tube characteristics. ISI is performed either to demonstrate continued long term integrity of pressure tubes with specific known, or suspected, integrity concerns, or to help in the efficient planning of maintenance operations for these tubes. The details associated with such ISI programmes change as the requirements for information by utilities and/or regulators change. They are developed annually and cover all scheduled inspections within planned reactor outages. ISI programmes are derived from consideration of:

(a) Reactor outage schedules,
(b) Results of previous inspections/material sampling,
(c) Engineering assessments of pressure tube integrity,
(d) Commissioning and operating experience,
(e) Fabrication history, and
(f) Results of research programmes into the metallurgical behaviour of pressure tube materials.

These sources identify the issues which affect inspection requirements, inspection sample sizes, and inspection intervals.

Although the primary purpose of a fuel channel ISI programme is to monitor the integrity of pressure tubes which may be at higher than average risk due to known or suspected causes, it also generates engineering data on fuel channel performance for operating, design, and research applications. The engineering data collected from ISI inspections is used to:

(a) Ensure continued reliable reactor operation,
(b) Verify engineering assessments,
(c) Confirm research findings,
(d) Plan future inspection and maintenance strategies, and
(e) Provide input into future research programme direction.

Information on pressure tube material behaviour is expanding rapidly as a result of the extensive in-service inspections, the material sampling campaigns at operating units, the assessments of material conditions in lead units, and the comprehensive research programmes. Since the direction of ISI programmes relies heavily on this information, they must be re-evaluated periodically to ensure their primary objective of demonstrating that continued reliable reactor operation is met.

Inspection of CANDU pressure tubes is an important part of the overall CANDU plant maintenance strategy. In order to satisfy fuel channel inspection requirements while keeping outage duration and radiation exposure to personnel to a minimum, many special techniques and systems have been developed to perform pressure tube inspections. Section 5.1 reviews the measurement techniques that are now available for the inspection and monitoring of pressure tubes. Section 5.2 briefly describes some of the systems that have been developed to perform pressure tube inspections.

5.1. PRESSURE TUBE INSPECTION AND MONITORING METHODS

Pressure tube inspection and monitoring capabilities have been developed progressively as needs have been recognized. AECL instituted monitoring programmes for the NPD and Douglas Point prototype CANDU reactors involving in-channel gauging and also tube removals. When the first commercial CANDU reactors were commissioned at Pickering, the periodic inspection programme required that the inside diameter, sag profile, length and surface profilometry of a small number of tubes be measured every 5 years. The diameter, sag and length were required to monitor the irradiation creep and growth of the pressure tubes. Profilometry was used to assure that the fuel bundle bearing pads were not unduly wearing the tube surface. The equipment used for this inspection, known as the dry channel gauging equipment, is described in Section 5.2.1. Operating experience at Pickering and later reactors rapidly broadened the scope of the pressure tube inspection methods that were required. Although this equipment has primarily been developed for Ontario Hydro's multi-unit stations, virtually all of it is directly applicable to the single unit CANDU-6 stations. Ontario Hydro's primary pressure tube inspection system, CIGAR (Channel Inspection and Gauging Apparatus for Reactors), which is described in Section 5.2.4, routinely performs contract inspections of single unit CANDU-6 pressure tubes for other utilities.

The following sections provide an overview of the pressure tube inspection and monitoring methods.

5.1.1. Pressure tube flaw detection

(a) Volumetric inspection methods

Prior to 1974, in-service volumetric inspection of pressure tubes (inspection of the full volume of the metal for flaws rather than just its surface) was not believed to be necessary. The tubes were volumetrically inspected during manufacture, operating conditions were not thought to be conducive to flaw development and there was no previous experience of flaw growth. However, in 1974, leaks developed in some Pickering units 3 and 4 pressure tubes [21]. The cause was found to be cracks that had developed at their rolled joints due primarily to improper installation procedures. The leaking channels were located using the acoustic emission techniques described in Section 5.1.4. However, many other channels had partial cracks that had to be located to prevent future problems. Ultrasonic rolled joint inspection equipment was developed to locate these flaws. The equipment and its capabilities are discussed in Section 5.2.2.

The initial Bruce units were under construction at this time and pressure tubes had already been installed in units 1 and 2 using similar procedures to those used for Pickering units 3 and 4. Remedial measures (stress relieving of the rolled joints) were undertaken and a reinspection of the full length of the pressure tubes was performed using eddy current inspection techniques delivered by the STEM delivery system (see Section 5.2.3).

During the pre-service eddy current inspection of Bruce units 1 and 2 tubes, a lamination defect was discovered in one tube. Further investigation revealed that it was introduced during manufacture of the tube and had not been detected by the manufacturer's inspection. Changes were subsequently made to the tube manufacturing and inspection procedures to reduce the possibility of this type of flaw re-occurring (see item (c) of Section 4.1.3).

The development of rolled joint cracks, and the discovery of a manufacturing flaw in a tube in reactor, highlighted the need for a new inspection system capable of performing pressure tube volumetric flaw detection in addition to other periodic inspection requirements. The system had to operate quickly and require very little radiation exposure to operating personnel. To meet these needs, the CIGAR system was developed [46, 47]. This equipment and its capabilities are discussed in Section 5.2.4. It came into service in 1985 and has become the primary inspection and gauging system for operating CANDU pressure tubes since then.

The CIGAR drive, a complex automated system, was just entering its commissioning phase when the rupture of pressure tube G16 occurred at Pickering unit 2 in August 1983. CIGAR's capabilities were required, but attempting to use such a complex system untested in a radioactive environment was out of the question. A small manually installed drive system was quickly developed that operated with the CIGAR inspection heads, drive rods and inspection instrumentation. The system, christened CIGARette, fulfilled the immediate inspection need by determining that the conditions that had caused the failure of G16 were present in many other fuel channels in both Pickering units 1 and 2. The CIGARette equipment and its capabilities are described in Section 5.2.5.

Another tube rupture led to further developments. In 1986, Bruce unit 2 pressure tube N06 started to leak, and the unit was safely shut down, but the tube later failed during low temperature pressurization. Investigation showed that the leak was caused by a crack that grew from a manufacturing lamination defect of similar origin to that mentioned earlier. Concern was raised about the possibility of other similar manufacturing defects being present in other Bruce units 1 and 2 and Pickering units 3 and 4 pressure tubes. (Pickering units 1 and 2 were being retubed and the tubes in later reactors had been subjected to improved manufacturing inspection). Therefore, a new system was developed to rapidly inspect only the rolled joint region of pressure tubes in Ontario Hydro's older reactors. The CIGAR system can inspect only 2 channels/day. The new system, PIPE (Packaged Inspection ProbE), can inspect approximately 24 rolled joints/day. The system was first used in Bruce unit 2 in 1987. A description of this equipment and its capabilities is given in Section 5.2.6.

(b) Surface profilometry

The dry channel gauging equipment used for periodic inspection of pressure tubes prior to the introduction of the CIGAR system had a strain gauge stylus type surface profilometer. Its primary function was monitoring the depth of fuel bundle bearing pad fret marks and fuelling scratches on the bottom of the tubes. By the time CIGAR was being developed, it had been determined that fuel scratch marks were not a serious problem and that monitoring of them could be carried out by eddy current methods. Consequently, a profilometry capability was not developed for CIGAR at that time. Later, concern arose about possible fretting of some pressure tubes in Pickering unit 5 by debris that might have escaped from a broken strainer during commissioning of the unit. An LVDT (linear variable differential transformer) stylus type profilometer that can be used in a water filled channel was developed for CIGAR and used successfully to measure the depth of fretting damage located by the ultrasonic flaw detection system.

A more widespread fretting problem in Pickering unit 8 led to the development of a number of other surface inspection systems. During the commissioning of Pickering unit 8, cast iron simulated fuel bundles made with special bearing sleeves were placed in the pressure tubes to prevent vibration from moving the garter spring spacers. Some of the simulated fuel bundles vibrated sufficiently that the sleeves caused fretting damage to the inside surface of the pressure tubes. The affected tubes were located using eddy current methods, then the extent of damage at each location was determined using one or more of the following three tools developed for assessment of this problem:

(1) The mini-gauging system is a stylus type surface profilometry head attached to the STEM delivery system. This unit can measure depths up to 0.8 mm.

(2) The super mini-gauging system is similar to the mini-gauging system but contains a Welch-Allyn Video Probe to allow the operator to view the area being profiled by the stylus.

(3) CIRT (circumferential in-service replicating tool) is a device which allows a 150 mm long full circumferential section silicon rubber replica to be made of the inside surface of a pressure tube. Excellent surface detail is obtained from which accurate depth measurements can be made.

These three tools developed for use in Pickering unit 8 have only been used in a pre-service reactor. However, with some modification they should be capable of being used in operating reactors, if the need arises.

(c) Flaw replication

If a blunt notch open to the inside surface of an installed pressure tube was detected prior to reactor startup, a mold was occasionally made to define its geometry for use in a refined flaw assessment. Recently, techniques have been developed for doing this flaw replication for operating tubes with fret marks caused by fuel bearing pads or by debris in the primary heat transport system.

(d) Pressure tube visual inspection

A visual inspection capability based on a radiation resistant hermetically sealed CCTV (Closed Circuit TeleVision) system has been developed and used for inspection assessment of a number of in-service pressure tubes. The camera head is mounted to the STEM delivery system and must be used in an isolated and drained channel. Both axial and radial viewing heads are available. Head positioning, light intensity, focus adjustment and data recording are controlled from a location outside containment.

A camera system that can operate under water and can be delivered by the CIGAR system was recently developed. This allows wider use of visual inspection because it does not require the time and radiation exposure of personnel to isolate and drain the channels to be viewed.

5.1.2. Pressure tube geometry monitoring

(a) Diameter and wall thickness measurement

The periodic monitoring of pressure tube diameter is important to assure that diametral creep and growth stay within their design allowances. CIGAR uses an ultrasonic gauging system that measures wall thickness as well as diameter at 60 positions around the tube at 3 mm axial intervals along its length to confirm that these deformations continue to be acceptable.

The small wall thickness reduction that occurs during reactor operation is of no concern as it is more than compensated for by irradiation strengthening. Diametral expansion of the pressure tube will eventually result in unacceptable fuel cooling, but this is not expected to cause a significant concern during the 30 year design life for CANDU fuel channels. If necessary, fuel management and design changes can be used to mitigate the fuel cooling related concerns.

(b) Sag measurement

Measurement of the pressure tube displacement profile and the maximum vertical deflection (sag) are also a periodic inspection requirement. CIGAR uses a servo inclinometer to measure the slope at many positions along the tube and calculates the displacement profile from a single integration.

Another method that has been used to provide a quick, but less accurate, estimate of maximum pressure tube sag is laser sag measurement. In this method, a laser beam is aligned to pass through the centre of each end fitting. The beam is then raised until it just grazes the top of the sagged pressure tube. The magnitude of sag is calculated from the difference in elevation of the laser at the two positions. This technique requires a dry channel.

A key use of sag measurements is to confirm that fuel channels will not contact the horizontal reactivity control mechanisms located below them. Although this is not expected to occur during the 30 year design life for fuel channels, measurements of fuel channel sag (and of the gap between fuel channels and mechanisms) are performed for lead units, to ensure that contact will not occur. If necessary, the horizontal mechanisms can be moved downwards to increase the gap between them and the calandria tubes above them.

(c) Pressure tube elongation

After only a few years of operation, length measurements indicated that pressure tubes in the initial Pickering units were elongating somewhat faster than had previously been predicted. The dry channel gauging equipment required one to two days to perform measurements on each channel and a faster method of measuring length was required. Hand operated “spot face tools” were designed to measure the distance from the end of each endfitting to the spotfaces on the reactor endshields. Since the fuel channels were fixed to the endshield at the west face of the reactor, and free to expand on the east face, spotface measurements on the east end quickly gave a relative measure of the elongation of many Pickering channels. These tools, and the equivalent "stud tools" developed for Bruce and Darlington reactors, are still used occasionally and have undergone considerable improvements since the original design.

The immediate concern, in 1976, about increased elongation at Pickering was the need to adjust the “yoke nut gap”, a clearance between the fastening nuts on the studs of the energy absorbers attached to the end shields and the yokes attached to the end fittings, that were designed to limit the outward movement of the end fittings should a guillotine separation of the channel occur. The clearance was necessary to allow the fuel channel to freely expand when heated to operating temperature. Manually operated “yoke nut gap” tools were developed to allow this clearance to be measured before and after adjustment.

After a large number of elongation measurements were obtained, it was apparent that elongation rates vary considerably from channel to channel. Making manual measurements was time consuming and subjected personnel to a large radiation exposure. To allow the elongation of all channels to be measured quickly and without radiation exposure, a new device, STEM (scanning tool for elongation measurement) was developed. This tool is attached to the fuelling machine and can be operated remotely. Measurements are made periodically and can be made either with the reactor shut down (cold) or on power (hot). Being able to measure the change in length of the Pickering channels when they were hot provided assurance that the yoke nut gaps were properly set and that nothing was constraining the channels from expanding as they should.

Since there is a large variation in tube-to-tube elongation rates, the in-reactor elongation rates of all tubes in each CANDU unit are monitored. Some units obtain the required data about fuel channel length from the fuelling machines each time a channel is refuelled. Such data not only confirms that the fuel channel design is adequate to allow 30 years of elongation, but also provides an accurate elongation rate for each tube, which can be used to establish the correct timing for the mid-life channel reconfiguration (i.e., releasing the initially fixed end of the fuel channels and fixing the initially free end), and provides lead time for planning corrective action if elongation becomes a life limiting issue for the pressure tubes with the fastest elongation rates. There are design solutions available that could be considered to manage this if it occurs, such as defuelling the fastest elongating pressure tubes and then appropriately limiting the flow in these empty channels.

5.1.3. Detection and measurement of spacers, blisters, contact, gap and hydrogen/deuterium

(a) Spacers

Shortly after the failure of Pickering unit 2 pressure tube G16 in 1983, the cause was determined to be the displacement of the "garter spring" spacers that separate the hot pressure tubes from the cooler calandria tubes. The displaced spacers allowed the two tubes to contact. Within a few weeks, an eddy current device capable of detecting the location of these spacers was developed. It is now used on CIGAR for in-service and periodic inspection, and with STEM equipment for dry channel and reactor construction checks.

(b) Contact

After the G16 failure, the spacer location coils and CIGAR ultrasonic flaw detection system were the primary inspection tools used during the inspection of the Zircaloy-2 pressure tubes in Pickering units 1 and 2. When a spacer was found to be significantly out of position, ultrasonic inspection usually detected a response on the bottom of the tube midway between the spacers. Cracks in hydride blisters gave fairly high amplitude response. Patches of lower amplitude response around and between the blisters were later shown to coincide with surface roughness and white oxide in the contact zone.

Inspections performed in other reactors (all of which have pressure tubes made of a newer alloy, Zr-2.5%Nb) have located some fuel channels with spacers significantly displaced from their design location. In these cases, low amplitude ultrasonic indications, interpreted as roughness and oxide, have been observed on the pressure tube outside surface at expected locations of pressure and calandria tube contact. Some of these tubes have been removed and it has been metallographically confirmed that no blisters were present.

(c) Blisters

Although no blisters have yet been found in reactors with Zr-2.5%Nb tubes, laboratory tests have demonstrated that they can occur. Hence, misplaced spacers causing contact between pressure and calandria tubes is considered undesirable. Consequently, a Spacer Location And Repositioning (SLAR) System has been developed by the CANDU Owners Group [48]. While SLAR's primary function is not inspection, a number of new fuel channel inspection systems have had to be developed in order for it to perform its mission.

The SLAR system can introduce significant bending stress in the pressure tube during the spacer repositioning operation. If blisters were already present in a tube, the bending might cause them to crack, or enhance existing cracking. A volumetric inspection system capable of rapidly detecting cracked blisters during a single axial pass through the pressure tube was required. The fast-scan blister detection system uses 6 line focused ultrasonic transducers that inspect the outside surface of the bottom 60 degrees of the pressure tube during a single axial pass through the tube [49].

More recently, a new inspection system dedicated to rapidly locating cracked blisters and spacers has been developed for Pickering reactors. Known as BLIP (Blister and Spacer Location Inspection with PIPE), it utilizes the SLAR type fast-scan blister detection system combined with CIGAR spacer detection methods. A description of BLIP is given in Section 5.2.7.

(d) Gap

In order for the SLAR system [48, 50] to reposition a spacer, it must centralize the pressure tube in the calandria tube to unpinch the spacer. A method was required to monitor the gap during the centralizing operation [51]. A gap measurement system was developed that uses both eddy current and ultrasonic inspection equipment. The primary gap measurement is made by a send-receive eddy current method. The result is sensitive to small variations in the pressure tube wall thickness. An ultrasonic pressure tube wall thickness measurement system is used to measure these variations and compensate the gap measurement for them. A version of this gap measurement system has now been developed for use with the CIGAR inspection system.

(e) Hydrogen (H) and deuterium (D) measurement

D ingress is difficult to simulate in laboratory tests, so a statistically significant amount of data is needed from the lead reactors since this is the most important parameter used to predict the formation of brittle hydride blisters at the contact between a pressure and calandria tube. Until recently, this could only be done by the removal and testing of surveillance tubes, which is very expensive. As illustrated in Figure 16, a sampling tool is now available that scrapes thin slivers of metal from operating tubes, leaving a smooth, rounded groove whose depth is much smaller than the tube's corrosion and wear allowance [17]. These samples are analyzed for H and D using a specially developed vacuum extraction technique.

The Periodic Inspection Programme specified in CSA Standard N285.4 requires that six pressure tubes be scraped about 10 years after generation of first power for the lead unit at a power station. One of these tubes is then scraped at intervals of about three years. Each tube whose D concentration is being determined has scrapes taken at a few axial locations to indicate how D ingress varies along the length of the tube.

The scraper tools first remove the surface oxide layer, and then scrape a 0.1 mm thick sliver of clean material from a small area of the pressure tube. The present "dry" scraper tools are manually operated. They require the pressure tube that is being inspected to be defuelled, isolated by freeze plugs in the feeders, blanked, drained and swabbed before the samples are taken. An outage duration of about a day is required for each tube being scraped and there is some radiation exposure to personnel. A new "wet" scraper tool is being developed that can be operated by a fuelling machine so a shorter outage duration will be needed and there will be less radiation exposure to personnel [18].

The number of scrape samples, and therefore the number of axial positions at which H/D can be determined using this method, is limited in practice by time. Also, although the samples are small, a method that does not remove material from the tube would be preferable. Investigation has shown that electrical resistivity and ultrasonic shearwave velocity are parameters that might be related to H/D content. Development of H/D measurement methods based on these concepts is underway, but they are not yet available for in-reactor use.

5.1.4. Detection and location of leaks

A CANDU reactor's primary heat transport system (PHTS) might develop a leak in one of its few hundred pressure tubes, or in one of its other pressure boundary components (fuel channels have inlet and outlet bolted feeder pipe connections and closure plugs for on-power refuelling, etc.). In addition, there are heavy water moderator and various light water cooling systems within the reactor that could also possibly develop leaks. Despite the apparent potential for leaks, careful design, construction, operation and maintenance of these reactors has resulted in very few leaks occurring. However, it is very important to be able to detect a leak if it occurs and to quickly determine the location of the leak.

Leak detection systems are designed into the reactor. For example, dry CO_2 gas flows through the annulus between each pressure and calandria tube with the moisture content of this gas being constantly monitored during reactor operation using very sensitive leak detection equipment. This almost immediately identifies if any pressure tube leakage starts to occur.

Once a pressure tube leak is detected, it is important to quickly determine which tube is leaking. The following techniques can do this.

(a) Monitoring channel outlet temperatures

Channel outlet temperature monitoring is the primary method for locating which pressure tube is leaking, or at least defining that a leak is coming from one of a small group of tubes. As water enters the annuli of a leaking tube, more heat is lost to the moderator and/or end shield cooling system, so the outlet temperature of the leaking channel is slightly depressed. Because many annuli are interconnected as a string of annuli, this method may only determine the locality of a leak (one string of annuli rather than the specific leaking tube).

Two separate strings of annuli were located in Bruce unit 2 in 1982 using these methods. One leaking tube in each string was then identified by inspection for cracks.

(b) Ultrasonic technique

The "pigtail" tool is a pole with a pair of ultrasonic transducers on the end. When manually positioned on an annulus gas pigtail (small tubes joining one fuel channel annulus to another to form a string of annuli) it detects the presence or absence of water in the tube. Using these tools, the areas within the reactor where leaks may be occurring can be localized.

(c) Acoustic emission techniques

An acoustic emission (AE) system has also been periodically used to locate a leaking pressure tube [52]. It consists of instrumentation located outside containment connected via cables and preamplifiers to a sensor head mounted on the front of a fuelling machine. The sensor head consists of a round metal plate to which is attached a low frequency ultrasonic leak detection horn and one or more piezoelectric AE transducers with different frequency ranges. The round plate is acoustically isolated from the fuelling machine.

Location of the instrumentation outside containment allows the equipment to be used either during shutdown or full power operation. To check if a pressure tube is leaking, the fuelling machine presses the sensing plate hard against the end of a fuel channel endfitting. This allows high frequency acoustic activity from leaks remote from the sensors, like leakage from a cracked pressure tube, to be detected. The acoustic frequency spectra of a leak usually distinguishes the leak source within the reactor. Experience has shown that feeder connection leaks > 1.0 kg/h and pressure tube leaks > 0.5 kg/h can be located with this system.

(d) Other techniques

Another system developed but as yet not used for detecting small pressure tube leaks is the annulus gas "sniffer". It consists of flexible devices that are threaded into the inlet and outlet annulus gas system (AGS) headers to isolate and allow a gas sample to be drawn from a single AGS string. The sample would be monitored for D_2O content. By drawing the sample alternately from each end of a string, the position of a leak in the string might be located.

5.2. SYSTEMS FOR PRESSURE TUBE INSPECTION

5.2.1. Dry channel gauging equipment

The dry channel gauging equipment developed for early CANDU units is capable of measuring:

(a) The local curvature at many points along the tube from which the sag profile of the tube can be calculated. In some cases, the location of the "garter spring" spacers can also be deduced from the curvature information.

(b) The tube minimum and maximum inside diameter at 12.7 mm intervals along the tube length.

(c) The profile of the tube inside surface from which the depth of scratches or fret marks can be determined.

The use of this equipment initially required time consuming procedures during which operating personnel are exposed to radiation. Its delivery and data collection systems have been improved to minimize these undesirable features. Since its capabilities for pressure tubes have been superseded by later equipment (CIGAR - Section 5.2.4) the dry channel inspection equipment is kept operational primarily for occasional calandria tube gauging.

5.2.2. Rolled joint ultrasonic inspection equipment

In 1974 cracks, initiated by improper rolling procedures during installation, developed in some Pickering units 3 and 4 pressure tube rolled joints. The rolled joint ultrasonic inspection equipment was developed to determine which joints contained cracks. It is capable of performing a two transducer circumferential shearwave ultrasonic volumetric inspection of the rolled joint region of the pressure tube. The 5 MHz, 6.25 mm diameter unfocused transducers are scanned in a circumferential 3 mm pitch from the pressure tube end to 150 mm inboard.

The use of this equipment required the fuelling machine to install the transducer probe into one of the end fittings of the channel to be inspected. The probe drive was then manually mounted onto the end fitting and connected to the probe. The control console was located in the reactor vault and data collection was rudimentary. The limited capabilities and the radiation exposure to operating personnel have rendered this equipment obsolete.

5.2.3. STEM inspection delivery equipment

The STEM inspection delivery equipment was developed for pre-service eddy current evaluation of Bruce units 1 and 2 pressure tubes. This delivery system is capable of scanning an inspection probe axially through the full length of a dry pressure tube. Probes with rotational capability may be fitted. This system has been used to perform eddy current volumetric inspection with one or more probes of various types, "garter spring" spacer location scans, and CCTV video visual examinations of the insides of pressure tubes and calandria tubes.

This equipment has been used extensively for pre-service inspections. Because its use on in-service reactor pressure tubes requires the time and exposure intensive process of isolating and draining the channel, it is used only for special in-service inspections.

5.2.4. CIGAR

CIGAR (Channel Inspection and Gauging Apparatus for Reactors) was developed in recognition of the need for a system capable of performing a wide range of measurements quickly and with minimum radiation exposure to operators [46, 47]. The system has been operational since 1985 and has inspected a few hundred pressure tubes in Ontario Hydro's reactors, as well as some tubes in Quebec Hydro's CANDU-6 unit at Gentilly, New Brunswick's CANDU-6 unit at Pt. Lepreau and the Republic of Korea's first CANDU-6 unit at Wolsong.

CIGAR, which is a highly automated system, is the primary inspection tool now used for most operating CANDU pressure tubes. This system consists of an in-channel inspection head, drive mechanism, computer controls and full data collection, processing and playback facilities. The drive mechanism, which is mounted on the reactor fuelling machine bridge, as illustrated in Figure 27, moves the inspection head (see Figure 28) along the fuel channel under remote direction from a control console located outside containment. CIGAR's inspection head performs pressure tube flaw detection and measures its diameter, wall thickness, and sag profile. It can also measure the gap between the pressure tube and the calandria tube, and can determine the position of the spacers that keep the two tubes separated.

To perform a CIGAR inspection of a pressure tube, the reactor must be in its shutdown state, with its PHTS cold and depressurized. The fuel channel must be defuelled, but does not need to have the primary coolant drained from it or to be isolated from the PHTS. The inspection head is attached to a special fuel channel closure and installed by the reactor fuel handling system into the fuel channel to be inspected. After the CIGAR equipment has been installed at the reactor face, the complete inspection and data analysis for one pressure tube requires only about four hours. Most of the data acquired from the inspection systems are processed and evaluated before the equipment leaves the channel being inspected.

CIGAR is capable of the following:

(a) Flaw detection — Ultrasonic volumetric inspection of the pressure tube using four 45 degree shearwave transducers arranged as a circumferential and an axial pair, and one normal beam transducer. This flaw detection system was designed to detect flaws 0.15 mm deep (6% of the pressure tube wall thickness) and larger.

(b) Diameter measurement — Ultrasonic gauging of the pressure tube inside diameter is performed at 18 degree intervals around the circumference at approximately 2000 axial positions along the tube. Tables and graphs of minimum and maximum diameter, their orientations and tube ovality can be obtained.

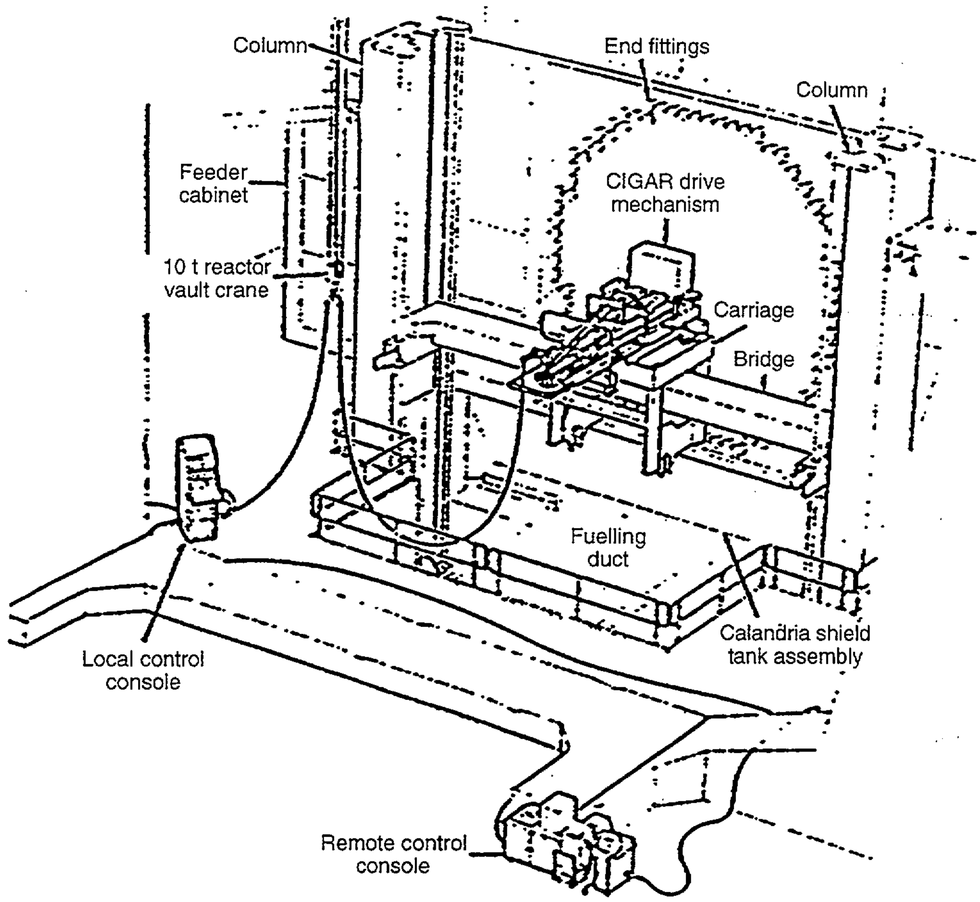


FIG. 27. Schematic illustration of the CIGAR system in operation [46].

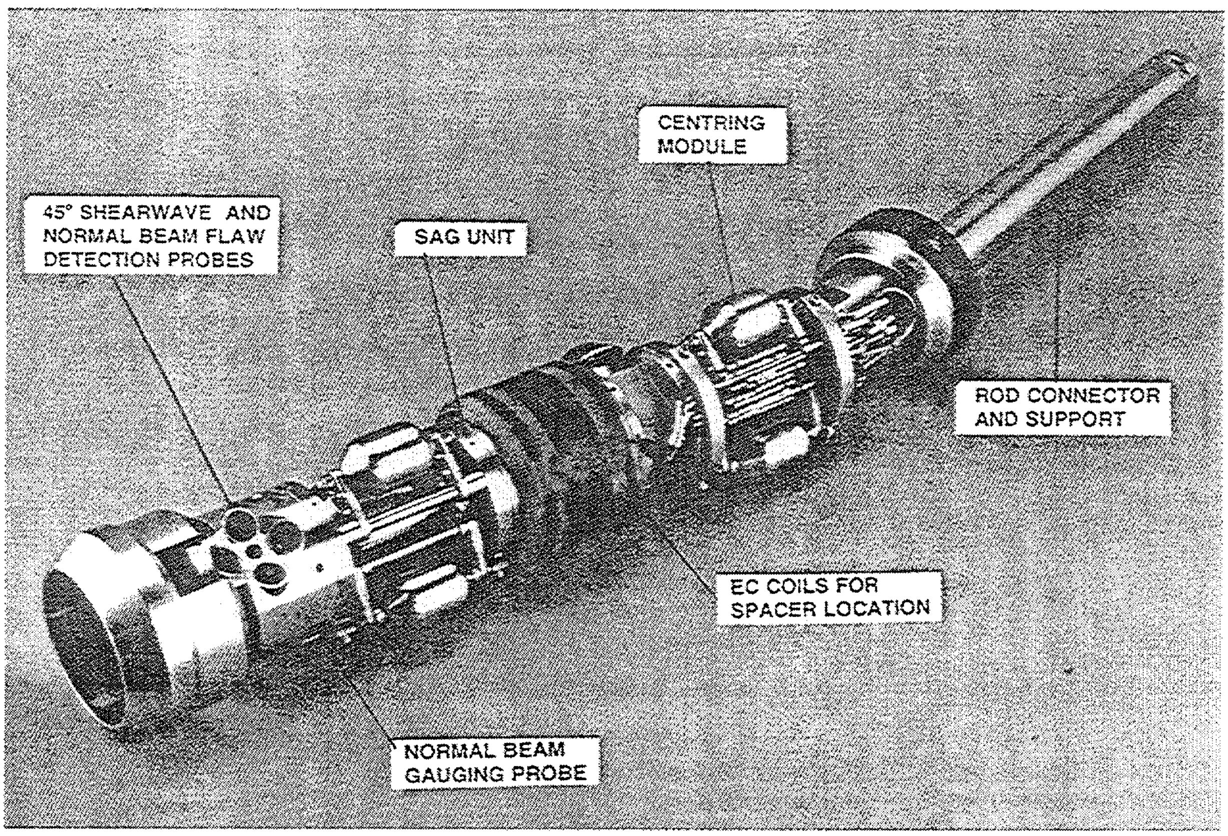


FIG. 28. CIGAR inspection head [46].

(c) Wall thickness — Ultrasonic gauging of pressure tube wall thickness is performed at the same time as diameter gauging and similar outputs are available.

(d) Sag measurement — The sag profile of the pressure tube is calculated from servo inclinometer slope measurements made at 30 mm intervals along the tube.

(e) Spacer location — The axial position of the "garter spring" spacers are located using a send-differential-receive eddy current coil.

(f) Inside surface profilometry — A stylus profilometer for measuring the depth of inside surface imperfections such as fuelling scratches, fuel bundle fret marks or fretting due to debris has been developed. It requires a special inspection head configuration and is used only for special inspections as required.

(g) CCTV video visual inspection head — A dedicated inspection head containing an underwater television camera and associated lighting has recently been developed to interface with the CIGAR drive unit to provide visual inspection capability.

Additional capabilities that are being added to the CIGAR system are:

(a) Gap measurement - A system for measuring the gap between the pressure and calandria tubes has been developed. It utilizes a send-absolute receive eddy current coil and requires ultrasonic gauging for wall thickness compensation.

(b) Eddy current flaw detection - This capability will complement the ultrasonic flaw detection for inside surface defects.

CIGAR is capable of inspecting an average of two fuel channels per day with approximately half the time being required for the fuel handling system to install inspection heads. Installation of the drive mechanism on the fuelling machine bridge and maintenance of equipment require minimal radiation exposure of personnel.

5.2.5. CIGARette

The CIGARette system uses a CIGAR inspection head and drive rods with a manually mounted and connected friction drive [53]. Because a CIGAR inspection head is used and the system provides access to the whole channel, CIGARette is theoretically capable of doing any inspection that can be done by CIGAR. However, the rotational drive is much slower than CIGAR's and continuous rotation is not possible. Also, on inspections requiring many repeated rotations, slipping of the friction drive can lead to loss of position reference.

The CIGARette system has been used primarily for limited area ultrasonic examinations and "garter spring" spacer location scans. Because of the requirement of manual mounting and rod connection at the reactor face, its use is time consuming and results in radiation exposure to personnel. Its use has been superseded by CIGAR and it is kept primarily for emergency use.

5.2.6. PIPE

The PIPE (Packaged Inspection ProbE) equipment was designed in 1987 to rapidly perform ultrasonic inspection of pressure tube rolled joint regions to assure the absence of unacceptable manufacturing flaws or delayed hydride cracks [47]. Separate systems have been built for Bruce and Pickering. They have been used to inspect rolled joints at an inspection rate of 10 channels (20 rolled joints) per day.

The system is capable of performing ultrasonic flaw detection with sensitivity and coverage equivalent to that of the CIGAR system. The primary inspection is performed by scanning circumferential looking 45 degree shearwave transducers in a circumferential 1 mm pitch. A normal beam and two axial looking 45 degree shearwave transducers are also provided for flaw characterization. The length of tube inspected is 100 mm extending approximately 50 mm on either side of the end of the rolling transition.

PIPE performs a very specific, limited function but does it quickly and with very low radiation dose to operating personnel. The system, which uses modified fuelling machines to manipulate the inspection head, has demonstrated the feasibility of getting instrumentation cables through the fuelling machine pressure boundary and making underwater connections of signal cables. A concern about this concept is that the modification and use of the fuelling machines limits the ability of a shared fuel handling system, such as that at Bruce, to service other reactors. It also exposes the skilled mechanics who maintain and modify the machines to additional radiation dose.

5.2.7. BLIP

The BLIP (Blister and Spacer Location Inspection with PIPE) system for Pickering reactors was developed in 1988 in anticipation of the need to be able to rapidly survey many fuel channels to find those with displaced spacers and to determine whether cracked blisters have formed. Spacer location is performed using the same eddy current method used by CIGAR. Cracked blister location employs line focused axial looking 45 degree shearwave ultrasonic transducers capable of inspecting the bottom 60 degrees of the tube during axial movement of the inspection head. They are identical to those used on the SLAR system.

Like the PIPE system, BLIP utilizes the reactors' fuelling machines modified to provide greater travel and to allow instrument cables to penetrate the pressure boundary. The machines at both ends of the fuel channel must be modified if the whole length of the pressure tube is to be inspected.

BLIP has been designed to perform a very specific, limited function but to do it more quickly than other existing systems. However, conversion of the fuelling machines takes a significant length of time and involves radiation exposure to skilled personnel. These drawbacks dictate that the system only be used if the inspection of a very large number of pressure tubes is required.

5.2.8. Scrape tool

Small pressure tube samples (0.1 mm slivers) are removed for analysis to determine their hydrogen and deuterium content, as illustrated in Figure 16 [17]. Once the sample is taken, the tool is removed from the fuel channel so the sample can be retrieved and analyzed.

Many hundred pressure tube samples have been successfuly taken from various CANDU reactors.

5.3. SUMMARY

The fuel channel design facilitates the non-destructive inspection of pressure tubes, and also the relatively easy removal of individual tubes to monitor their material condition. A wide variety of techniques and equipment have been developed and used to obtain defect, dimensional, and compositional information, in-situ, during periodic and in-service inspections for a sample of the irradiated pressure tubes in CANDU nuclear reactors. This data can be referenced to results from the pressure tube research and development programme and to the condition of surveillance tubes that are periodically removed from lead reactors. This allows accurate assessments to be done to demonstrate that operating pressure tubes have an adequate material condition. Research and development continues to enhance and extend these inspection and monitoring capabilities.

6. ASSESSMENT METHODS AND FITNESS-FOR-SERVICE GUIDELINES FOR PRESSURE TUBES

The general requirement for assessments that ensure the continued reliable operation of the pressure tubes in CANDU and Indian reactors is to show that:

- their deformation does not exceed allowable limits (using data primarily obtained by periodic measurements of tube deformation and models developed for predicting this deformation [29]),

- their service induced deterioration does not exceed allowable limits,

- their material remains in an adequately tough condition (using data primarily obtained by periodically removing and testing surveillance tubes), and

- there are no mechanisms which could cause significant crack growth.

It must be demonstrated that delayed hydride cracking (DHC) cannot occur in a pressure tube. Furthermore, as a defence in depth, it must also be demonstrated that if DHC does occur, the resulting leak can be detected and the reactor safely shut down before the crack becomes unstable. The leak before break (LBB) behaviour for pressure tubes is dependent on the leak detection capability of the annulus gas system, on the manner in which a reactor will be shut down in response to a pressure tube leak, and on the various factors associated with the growth of DHC in pressure tubes, such as their material toughness, etc. Assessment of the LBB behaviour of a pressure tube that is assumed to be exhibiting DHC requires knowledge of at least the following:

- the maximum length at which a crack penetrates the pressure tube wall,
- the crack velocity,
- the critical crack length, and
- the leak detection time.

The first three parameters are material properties determined primarily from tests on surveillance tubes. The last parameter is a function of reactor design and operating procedures.

New pressure tubes have a large margin between the initiation of leaking and unstable rupture. However, as they are operated this margin decreases due to irradiation and deuterium (D) ingress. As a pressure tube's operating life increases, there is an increasing concern that a sufficiently large flaw could cause precipitation of hydrides and initiate DHC. In order for tubes to continue operating, it is required that assessments be done to show there continues to be a high confidence both that DHC will not occur, and that pressure tubes will exhibit LBB, if DHC does occur.

6.1. FLAW ASSESSMENT REQUIREMENTS

CSA Standard N285.4 requires both the periodic inspection of pressure tubes in operating CANDU nuclear reactors and the occasional removal of surveillance tubes for laboratory testing, to ensure that unacceptable degradation in component quality is not occurring. If indications exceeding the acceptance standards of CSA-N285.4 are detected, an evaluation is required to determine if the pressure tube is acceptable for continued service.

Fitness-for-Service Guidelines are being developed within the Canadian nuclear industry to provide acceptance criteria for flaws found in pressure tubes, and to define a standardized approach for assessing the integrity of operating zirconium alloy pressure tubes. The main objective of the assessment process is to ensure that the probability of tube failure remains acceptably low throughout the operating life of the tubes [26]. The Guidelines complement the rules of Section XI of the ASME Code and the requirements of CSA-N285.4 on the periodic inspection of pressure tubes in operating CANDU nuclear reactors. The Guidelines provide the methodology for the evaluation of manufacturing flaws, in-service generated flaws, hydride blisters, and in-service degradation of zirconium alloy pressure tube properties. Trial application of the draft Guidelines has enabled the key pressure tube integrity issues to be more clearly defined and has caused the pressure tube research programmes to be more closely focused on these issues.

The draft Guidelines consist of three sections. The first section describes the requirements that must be met to qualify pressure tubes for continued service. It contains evaluation procedures and acceptance criteria for assessing crack-like flaws, notch-type flaws, pressure tubes in contact with calandria tubes and generic changes in material fracture toughness. The second section contains the material properties database needed to carry out the assessments for Zr-2.5%Nb tubes. The third section provides the technical basis for the acceptance criteria and evaluation procedures, as well as the justifications and descriptions of the databases. The draft Guidelines were issued to CANDU reactor operators for trial use, and released to the Atomic Energy Control Board of Canada for review and comment, in May 1991.

The content of these draft Guidelines is summarized below. In addition to describing the procedures and criteria being developed, an overview of the research programmes under way to address the key pressure tube integrity issues is presented. The research programmes outlined are intended to remove the unnecessary conservatisms built into the current methodology.

6.2. SHARP FLAW ASSESSMENT

To develop flaw evaluation procedures and acceptance criteria for pressure tubes, it is first necessary to define appropriate mechanisms for crack initiation, crack growth and crack stability. For cold-worked zirconium alloy pressure tubes, the two potential mechanisms of stable crack growth are fatigue and DHC. The fatigue properties of irradiated Zr-2.5%Nb indicate that failure by fatigue is much less likely than failure by DHC and all pressure tube failures to date [2, 3] have been due to DHC. Therefore, DHC is considered to be the dominant mechanism for the growth of flaws, and it is important to know the H/D content of pressure tubes. Until recently, data about the ingress of D into pressure tubes from the heavy water coolant that flows through them could only be obtained by removing tubes from operating reactors. However, such data can now be obtained by a non-destructive examination of operating tubes. This procedure involves the removal of a thin sliver of metal from an operating tube [17], as illustrated in Figure 16.

For the evaluation of pressure tubes found to contain sharp or crack-like flaws, it must first be shown that stresses are small enough so DHC is not expected to occur at operating conditions. Then, a conservative estimate must be made for the sub-critical crack growth that may occur during the evaluation period. Analytical procedures are defined for calculating the DHC flaw growth that could occur during startup/shutdown transients and the sub-critical fatigue flaw growth that could occur during loading cycles. Acceptance criteria have been developed to ensure there is adequate protection against failure by unstable fracture, plastic collapse, and DHC. If the H/D concentration is high enough to permit DHC at operating temperature, it must also be demonstrated, to the satisfaction of the appropriate regulatory authority, that if DHC does occur the postulated crack would penetrate the tube wall, resulting in a detectable leak and allowing sufficient time for the reactor to be safely shut down before the crack becomes unstable, i.e., LBB must occur, not rupture [9, 10].

The subcritical growth of a sharp flaw is conservatively calculated to determine the maximum possible size of the flaw at the end of the evaluation period. It must then be demonstrated that this flaw size is stable with adequate margins of safety for the various loading conditions. The concept of ASME Boiler and Pressure Vessel Code service level categories is used to adjust the margin of safety based upon the likelihood of an event. The categories are: Service level A – normal operating conditions; Service level B – upset conditions; Service level C – emergency conditions; and Service level D – faulted conditions.

The flaw evaluation sequence is:

(a) Characterize the flaw shape and size.

(b) Identify the design basis and operating transients for levels A, B, C and D service conditions.

(c) Perform a subcritical flaw growth analysis based on flaw growth due to fatigue and DHC to define a maximum flaw size (a_f).

(d) Evaluate the flaw based on the calculated maximum flaw size (a_f).

The acceptance criteria outlined in the following sections define whether or not a pressure tube containing a flaw is acceptable for continued operation.

6.2.1. Acceptance criteria for level A and B conditions

(a) Demonstrate that fracture will not occur. The following method is recommended:

$$K_I < K_I / \sqrt{10}$$

where:

K_I = the maximum applied stress intensity factor for the loading condition being evaluated and the calculated flaw size, a_f,

a_f = the maximum depth to which a detected flaw is calculated to grow during the evaluation period, and

K_I = the critical stress intensity factor for fracture initiation based on temperature, neutron fluence, and the predicted hydrogen/deuterium concentration.

The margin of safety recommended above for the level A and B conditions is consistent with that contained in ASME Section XI, IWB-3610, for reactor pressure vessels.

The fracture toughness, K_I, is given by the plane-stress relation:

$$K_I = \sqrt{J_I E}$$

where E is Young's modulus and J_I is the J-integral at fracture initiation. The value of J_I is determined from small specimen tests using standard fracture toughness test procedures developed for pressure tubes [37]. The draft Guidelines recommend that lower bound values be used. Data used to produce a toughness curve were obtained from numerous pressure tubes removed from various CANDU reactors. To date, over 200 small specimen fracture toughness tests have been performed on material that has seen operating temperatures ranging from 30°C to 300°C and a neutron fluence of up to 1×10^{26}n/m^2. To determine the reference curve for use in analysis, the effects of neutron fluence and temperature on K_I are evaluated by determining the relationship between K_I and neutron fluence and then the relationship between K_I and temperature for appropriate fluence levels.

(b) Demonstrate that failure by plastic collapse will not occur. The following method based on limit load expressions developed for piping is recommended [38, 39].

Circumferential flaws:

$$\sigma_a < \sigma_m / 3.0$$

where:

σ_a = the maximum applied primary axial stress for the loading condition being evaluated. Note that diametral expansion and wall thinning due to irradiation enhanced pressure tube creep and growth must be taken into account in conjunction with corrosion and wear allowances when determining the stresses.

σ_m = the membrane stress corresponding to plastic collapse for the crack depth and angle being evaluated. The value of σ_m is to be calculated as follows:

$$\sigma_m = \sigma_f \, (1 - (a / t)(\theta / \pi) - 2\gamma / \pi)$$

$$\gamma = \text{Arcsin}\left(\frac{a \sin \theta}{2t}\right)$$

The definitions of a, t,θ and σ_f are illustrated in Figure 29.

Longitudinal flaws:

$$\sigma_h < \sigma^1{}_h / 3.0$$

where:

σ_h = the maximum applied primary hoop stress for the loading condition being evaluated. Note that diametral expansion and wall thinning due to irradiation enhanced pressure tube creep and growth must be taken into account in conjunction with corrosion and wear allowances.

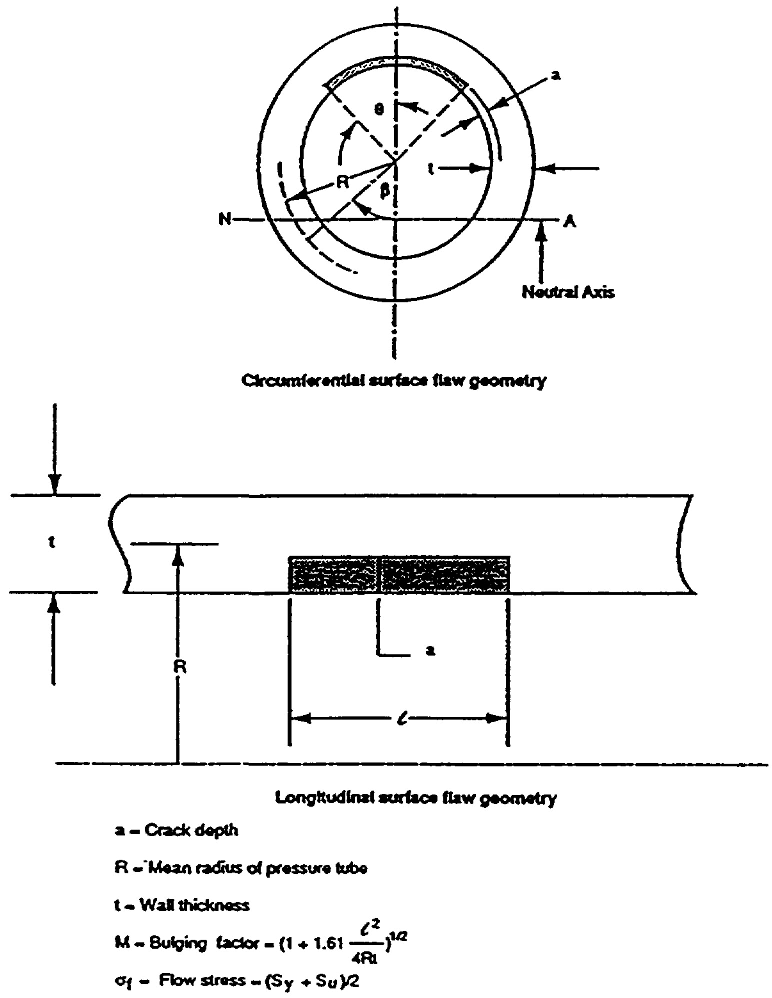


FIG. 29. Illustration of the symbols used in the platic collapse stress calculation for pressure tubes.

σ^1_h = the hoop stress at plastic collapse for the crack depth and length being evaluated. The value of $[^1_h$ is to be calculated as follows:

$$\sigma^1_h = \sigma_f [(1-a/t) / (1 - a/tM)]$$

The definitions of a, t, M and σ_f are illustrated in Figure 29.

The minimum margin of safety recommended is consistent with that contained in ASME Section XI, IWB-3610, which states that the primary stress limits must be satisfied for a local area reduction of the pressure retaining membrane equal to the area of the characterized flaw.

For normal operating conditions, the minimum factor of safety provided by the primary membrane stress limits is 3.0 because the ASME Code requires $P_m < S_m$ where S_m is the lower of 2/3 of the yield strength (S_y) or 1/3 of the ultimate strength (S_u). The margin of 3.0 recommended in the draft Guidelines is, therefore, consistent with the ASME Code requirements. The same margin of safety has been conservatively applied to both the level A and B conditions.

In determining the collapse load, the flow stress (σ_f), which is defined as the average of the yield and ultimate stresses, is required. The yield and ultimate strength of cold-worked Zr-2.5%Nb pressure tube material increases due to neutron irradiation. The draft Guidelines state that this increase in strength may be used in determining σ_f, but lower bound values should be used.

6.2.2. Acceptance criteria for level C and D conditions

(a) Demonstrate that fracture will not occur.

$$K_I < K_I / \sqrt{2}$$

(b) Demonstrate that failure by plastic collapse will not occur.

Circumferential flaws:

$$\sigma_a < \sigma_m / 1.5$$

Longitudinal flaws:

$$\sigma_h < \sigma^1{}_h / 1.5$$

These safety margins are consistent with the requirements of ASME Section XI, IWB-3610.

6.2.3. Additional acceptance criteria if TSS is exceeded at normal full power conditions

If the calculated equivalent hydrogen concentration at a flaw location at the end of the evaluation period exceeds the Terminal Solid Solubility limit for dissolution of hydrides (TSSD), but is below the solubility limit for the precipitation of hydrides (TSSP) at normal full power conditions, the following two additional criteria must be satisfied:

(a) Demonstrate that DHC will not occur:

$$K_I < K_{IH}$$

where:

K_I = the applied stress intensity factor for normal full power conditions and the final flaw size, a_f, and

K_{IH} = the threshold stress intensity factor for DHC.

The draft Guidelines recommend that a lower bound value of K_{IH} should be used. K_{IH} has values between 4.5 and 10.0 Mpa$\sqrt{m}$ and is insensitive to temperature, hydrogen concentration, and neutron fluence [26].

(b) Demonstrate that even if the flaw could propagate through-wall by DHC, LBB is assured [9, 10]. This defence in depth demonstration which shows that, even if a tube does crack, it will leak at a rate sufficiently high that the leak will be detected and the reactor shut down before unstable crack propagation occurs, is a key part of the draft Guidelines for CANDU pressure tubes.

To assure LBB in CANDU pressure tubes it is required that:

(i) the crack length at wall penetration be less than the CCL for unstable propagation, and

(ii) the leak from this through-wall crack be detected and the reactor put into a cold, depressurized condition before the crack length exceeds the CCL.

Although the nuclear industries outside Canada are also developing procedures for the application of LBB, these are for application to stainless and ferritic steel piping in light water pressure vessel reactor circuits and are not directly applicable to Zr-2.5%Nb pressure tubes owing to several reasons, such as:

(i) The consequences of failures in steel vessels and piping in light water reactor circuits can be different from those of a single pressure tube failure in a CANDU reactor.

(ii) The annulus gas monitoring system in CANDU fuel channels provides a very sensitive leak detection capability as soon as a pressure tube leak starts to occur.

Calculations to demonstrate LBB currently are deterministic assessments, although techniques for performing probabilistic LBB assessments are being developed [40]. The proposed acceptance criterion for a deterministic LBB assessment is:

$$t > T$$

where:

t = the lower bound calculated time for an assumed through-wall crack to grow from its length at penetration to the CCL (i.e., the time *available* to detect the leak and take action);

T = the upper bound response time of the leak detection system, which includes the time required to confirm the presence of a leak at normal operating conditions, and the time required to reach a cold depressurized condition (i.e., the time *required* to detect the leak and take action).

This criterion assures that leaking coolant would be detected, and the reactor shut down, before a through-wall DHC becomes unstable.

As illustrated in Figure 30, the methodology for calculating the time t is [9, 10, 41]:

$$t = \frac{CCL - L}{2V}$$

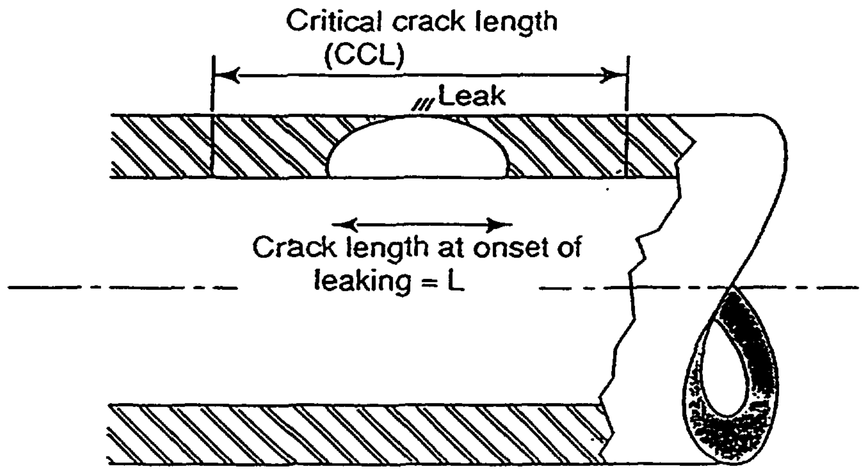
where:

L = the upper bound length of the crack at initial leakage, which is about five times the pressure tube wall thickness,

a. Provide assurance that hydride cracking (DHC) will not occur.

$$K_I < K_{IH}$$

b. Demonstrate that leak-before-break (LBB) is assured for DHC.



$$\text{Time available to take action} = \frac{CCL - L}{2V} > \text{Station response time}$$

V = CRACK FACE VELOCITY

FIG. 30. Use of pressure tube leak before break in fuel channel design as a defence in depth for preventing pressure tube rupture.

CCL = the lower bound critical crack length from slit burst tests on tube sections (this is the minimum length of an axial through-wall crack that would be unstable at the temperature and pressure being evaluated), and

V = the upper bound DHC growth rate in the axial direction at the temperature being evaluated.

This deterministic LBB assessment is not the primary defense against unstable pressure tube rupture, however, it still incorporates a conservative approach by using the worst combination of parameters. Each of the parameters used in an LBB assessment is discussed in the following.

Crack length at penetration (L): Evaluation of up to two hundred measurements of crack shapes in pressure tubes at different stages of their growth in reactor by DHC has indicated that the average crack length at wall penetration was 3.6 times the thickness of the pressure tube [10]. In the draft Guidelines, it is recommended that for CANDU pressure tubes L conservatively be taken to be the greater of 20 mm or the calculated maximum length of the flaw being assessed when it breaks through the wall as a result of crack propagation by DHC [26].

Critical crack length (CCL): The critical crack length has been determined using slit burst tests on tube sections and fracture tests on small specimens. To date over 70 burst tests and over 200 small specimen fracture tests have been performed to estimate the CCL. In the draft Guidelines, it is currently recommended that the lower bound CCL obtained from burst tests be used in the analysis for the material temperature, neutron fluence, and loading conditions under consideration, since the constraint conditions in the small specimens result in overly conservative values.

DHC growth rate (V): Over 400 crack growth rate tests have been carried out using small specimens prepared from Zr-2.5%Nb pressure tubes removed from reactors after irradiation to fluences up to 10^{26} n/m^2 [10]. Pressure tube sections have also been used to measure crack growth rates [41]. An upperbound crack growth rate is used in analyses.

6.2.4. Conservatisms in the evaluation of sharp flaws

The draft pressure tube Fitness-for-Service Guidelines contain many conservatisms in both the evaluation methods suggested for the assessment of flaws and in the interpretation of the material properties databases. For example:

(1) Flaws often have to be assessed as being sharp because of an inability to characterize them as blunt. NDE ultrasonic methods currently in use often do not have the ability to accurately determine the root radius of the flaws. In-core flaw replication techniques have been developed and are in use, but there are limitations to their use and they are currently very dose intensive.

(2) In both sharp and blunt flaw assessments, the stress state at the flaw tip is currently calculated with no allowance being made for relaxation of flaw tip stresses due to creep and growth. This could result in unnecessarily restrictive thermal cycle limitations on the allowable operation for tubes with detected flaws.

(3) In the subcritical flaw growth evaluation approach for sharp flaws, it is currently assumed that DHC initiation and growth starts, during cooldown cycles, as soon as the TSSD temperature is reached. No allowance is made for the significant incubation time required for hydrides to grow to the critical size required for initiation.

(4) The DHC velocity that is used in the subcritical flaw growth analysis for sharp flaws, during a cooldown cycle, is currently assumed to be the upper (95%) confidence level obtained from the DHC velocity database. The database exhibits wide variability and work is in progress to identify the separate effects of fluence, irradiation temperature, and hydrogen concentration so that use of the database can be more specific to the appropriate conditions.

6.3. BLUNT FLAW ASSESSMENT

For a flaw to be characterized as blunt, the geometry (length, depth, width, and root radius) must be known so that a stress analysis to determine the peak value of the notch tip stress concentration can be performed.

To assess if pressure tubes containing notch-type flaws may be exempted from the crack-like flaw evaluation criteria, it is necessary to demonstrate both that the flaw is not sharp, and that it will not become sharp during the evaluation period. For CANDU pressure tubes, initiation of a crack at the root of an existing notch-type flaw may occur because of fatigue crack initiation or DHC.

In the draft Fitness-for-Service Guidelines for CANDU pressure tubes, whose content is being summarized here, the assessment of blunt-type flaws utilizes a two-step approach. The first step involves an evaluation to determine if crack initiation by fatigue or DHC will occur due to any loading conditions that may occur during the evaluation period. If crack initiation is predicted, then the flaw must be treated as crack-like.

If the flaw remains blunt, and TSS (see Figure 13) is not exceeded at operating temperature, then it is only necessary to show that the flaw has adequate protection from failure by plastic collapse. If the hydrogen/deuterium concentration is high enough for DHC at operating temperature, then LBB must also be demonstrated.

As in the sharp flaw assessment, the concept of ASME service level categories is used to define the loads and adjust the margin of safety based upon the likelihood of the loading event occurring.

The flaw evaluation sequence is:

(1) Characterize the flaw shape and size.
(2) Identify the design basis and operating transients for levels A, B, C and D service conditions.
(3) Evaluate the stability of the flaw based on the following three acceptance criteria.

6.3.1. Acceptance criteria for fatigue crack initiation

To demonstrate that crack initiation due to cyclic loading will not occur, either of the following two criteria must be satisfied.

(a) Maximum alternating stress criteria

$$S_{max} < S_{th}$$

where:

S_{max} = the maximum alternating peak stress at the flaw tip (one-half of the stress range) for all Service levels A and B loading conditions. The peak stress at the flaw tip can be determined from numerical stress analysis techniques or from an experimental study, such as photoelastic analysis.

S_{th} = the threshold alternating stress for fatigue crack initiation as determined from the fatigue crack initiation curve, which utilizes a factor of 2 on stress or 20 on cycles, whichever gives the more conservative result.

To determine S_{th}, fatigue tests have been performed at 300°C on unnotched ring specimens taken from unirradiated and irradiated Zr-2.5%Nb pressure tubes. Values of alternating stress versus cycles to crack initiation were determined, with the alternating stress being equal to one-half the total stress range over a load cycle. Fatigue tests were also performed at room temperature on notched ring specimens taken from an irradiated Zr-

2.5%Nb pressure tube. Values of the peak notch tip alternating stress versus cycles to crack initiation were calculated.

A fatigue crack initiation design curve was derived by first fitting a mean through the data points. The curve was then shifted by applying, at each alternating stress on the curve, the most conservative of a factor of 2 on stress or 20 on cycles. This approach is consistent with that used in the ASME Boiler and Pressure Vessel Code.

(b) Cumulative damage criteria

$$CUF < 1.0$$

where:

$CUF = \Sigma \, n_i / N_i$

n_i = the number of load cycles up to the end of the evaluation period for each Service level A and B transient, i.

N_i = the allowable number of load cycles for the peak alternating stress at the flaw tip, associated with transient i. N_i is to be determined from the fatigue crack initiation curve, which utilizes a factor of 2 on stress or 20 on cycles whichever gives the more conservative result.

6.3.2. Acceptance criteria for DHC initiation

It must be demonstrated that DHC will not initiate during all level A and B service conditions.

DHC initiation is controlled by two main processes: the formation and growth of a hydride and the presence of a sufficiently high stress across the hydride to cause fracture [11, 12]. The stress state close to the tip of sharp notches and cracks can be characterized by the elastic stress intensity factor, K_I [42]. However, the stress state at blunt notches cannot be correctly characterized by a simple stress intensity factor, nor by the applied far field stress. Characterizing the stress state at a blunt notch by K_I would overestimate the stress level whereas characterizing it by the far field stress would underestimate it. Based upon current experimental data, the maximum stress calculated to exist at a notch tip stress, as determined from an elastic-plastic analysis, is used to characterize the stress state at a blunt notch. Extensive experimental data has been produced to provide a threshold stress intensity factor (K_{IH}) and a threshold stress (σ_{TH}) for the initiation of DHC at sharp and blunt flaws respectively.

If the calculated hydrogen equivalent (Heq) concentration at a blunt notch at the end of the evaluation period exceeds the terminal solid solubility limit for dissolution of hydrides (TSSD) at normal full power conditions, the draft Guidelines require that the following criteria be satisfied to provide assurance there will be no DHC initiation:

$$\sigma_{max} < \sigma_{th}$$

where:

σ_{max} = the peak tensile stress at the flaw tip for all service level A and B loading conditions. The peak stress at the flaw tip may be calculated from numerical stress analysis techniques or experimental studies.

σ_{th} = the threshold stress for DHC determined from the peak notch tip stress for DHC initiation in notched cantilever beam test specimens. The test specimens were tested at 250°C and were subjected to periodic temperature cycles between 60 and 290°C.

For flaws with peak notch tip stresses greater than σ_{th}, when the calculated Heq concentration is below the solid solubility limit for the precipitation of hydrides (TSSP), the following criteria, which imposes thermal cycle restrictions, may be used to demonstrate that there will be no DHC initiation:

$$N_t < N_{all}$$

where:

N_t = the total number of thermal cycles up to the end of the evaluation period. A thermal cycle is defined as a cooldown from normal operating temperatures to room temperature and then a return back to normal operating temperature.

N_{all} = the allowable number of thermal cycles for the calculated peak notch tip stress, σ_{max}.

The relationship between N_{all} and σ_{max} was derived from experiments performed to determine the number of thermal cycles corresponding to the first failure in notched specimens. This data was plotted as peak notch tip stress versus number of thermal cycles. A lowerbound straight line of the number of thermal cycles to DHC initiation as a function of the peak notch tip stress was then determined. A factor of 8.0 on thermal cycles was then applied to this lowerbound relationship to define the allowable curve and relationship between N_{all} and σ_{max} for use in the assessment [26].

This thermal cycle approach is based on the current understanding that hydrogen (H) and deuterium (D) migrate to the notch tip due to the stress gradient, which acts as the driving force. Precipitation of hydride can only occur during cooldowns because the Heq concentration is below TSSP at normal steady state operating temperatures. When the temperature is returned to the normal operating temperature, not all of the hydride that may have precipitated during the cooldown will dissolve. Hydrides may therefore be present at a blunt flaw during the steady state normal operating period. The extent of hydride growth will increase with an increase in the number of thermal cycles.

It has been observed experimentally using metallography that the critical peak notch tip stress required to fracture a hydride decreases with an increase in the hydride length. The total peak stress acting on a hydride is the sum of the tension stress due to the applied load, plus the compressive stress due to expansion of the hydride relative to the matrix when the hydride forms. As the length-to-thickness aspect ratio of the hydride increases (due to thermal cycles), the compressive stress due to expansion will decrease. The total peak tension stress acting on the hydride therefore increases as the hydride grows longer with increasing thermal cycles.

This means that the critical number of thermal cycles to initiate DHC decreases with an increase in the peak notch tip stress. This relation has been demonstrated in the tests performed and used to derive the relationship between N_{all} and σ_{max}.

Note that the thermal cycle limitation applied to blunt flaws that have a Heq concentration above TSSD assumes the same amount of H/D accumulation, or hydride growth, with every thermal cycle. This assumption is very conservative and research work is under way to quantify this conservatism.

6.3.3. Acceptance criteria for prevention of failure by plastic collapse

To demonstrate that failure by plastic collapse will not occur, the criteria previously specified for the assessment of sharp flaws must be satisfied. These criteria are briefly summarized as follows:

Level A and B service conditions:

Circumferential flaws:

$$\sigma_a < \sigma_m / 3.0$$

Longitudinal flaws:

$$\sigma_h < \sigma^l_h / 3.0$$

Level C and D service conditions:

Circumferential flaws:

$$\sigma_a < \sigma_m / 1.5$$

Longitudinal flaws:

$$\sigma_h < \sigma^l_h / 1.5$$

6.4. BLISTER ASSESSMENT

For the evaluation of a pressure tube in contact with the cooler calandria tube surrounding it, analytical procedures and acceptance criteria are being developed for assessing the possibility that brittle hydride blisters will develop at the point of contact [43– 45]. Before there is sufficient H/D for blister formation and growth, the contact must be eliminated.

The necessary conditions for blister formation are pressure tube contact with the calandria tube, and sufficient accumulation of H/D isotopes in the bulk of the pressure tube in the contact region. Both of these conditions must exist before hydride blisters can form.

The inspection and maintenance strategy for early CANDU units is currently focused on the prevention of blister formation. This strategy will be reviewed when additional information is obtained from operating tubes and the ongoing blister modelling work. To prevent blister formation, susceptible channels are identified analytically by performing conservative blister formation assessments. The potential for blister formation is assessed using:

- Upper bound D ingress rates for pressure tubes in which no specific data (from tube scraping [17]) is available.

- Initial (as manufactured) H concentration for each tube, as obtained from analysis of their offcuts.

- Garter spring spacer locations, where they have been obtained from inspections. For channels where garter spring information has not been obtained, contact is conservatively assumed to have occurred at the most vulnerable locations.

- Thermal boundary conditions (moderator temperature, moderator saturation temperature, and temperature along each pressure tube) based on unit specific information.

- Conservative blister modelling assumptions (contact area, contact conductance, TSS, H/D diffusion rates, etc.).

6.5. GENERIC ASSESSMENT

To assess the acceptability of pressure tube fracture properties, only changes in the material properties due to fast neutron fluence are considered when hydrides are not present in the tube during nominal full power conditions. To assess the acceptability of material fracture properties when hydrides are present at normal full power temperatures, the effects of both the increased hydride concentrations and neutron fluence must be considered.

It is proposed that an evaluation procedure similar to that of Appendix G of ASME, Section III, be used. This procedure requires that a conservative flaw be postulated based upon the manufacturing inspection and in-service inspection techniques applicable to the pressure tubes of the reactor being evaluated. It must then be demonstrated that the stress intensity factor at the postulated crack tip is sufficiently less than the fracture toughness of the material.

For conditions in which the H/D concentration is *greater* than TSS (Figure 13) at nominal full power operating conditions, it must be demonstrated that there is a low probability of having manufacturing flaws susceptible to DHC. Furthermore, it must be shown that there is an equally low probability of initiating DHC at in-service induced flaws such as crevice corrosion marks, fret marks or fuel scratches, and that LBB is assured. Calculations must be performed to demonstrate that detection of a pressure tube leak by the annulus gas system results in LBB behaviour, in case DHC ever does occur in a pressure tube. These calculations must use appropriate operating/shutdown procedures and irradiated pressure tube properties (H/D concentration, critical crack length, DHC velocity, the anticipated distribution of flaws in the tubes being assessed, etc.) obtained from inspections on operating pressure tubes, tests conducted on tubes removed from operating units, and appropriate pressure tube research programmes.

Such generic assessments would define an allowable pressure vs. temperature operating envelope similar to that shown in Figure 31.

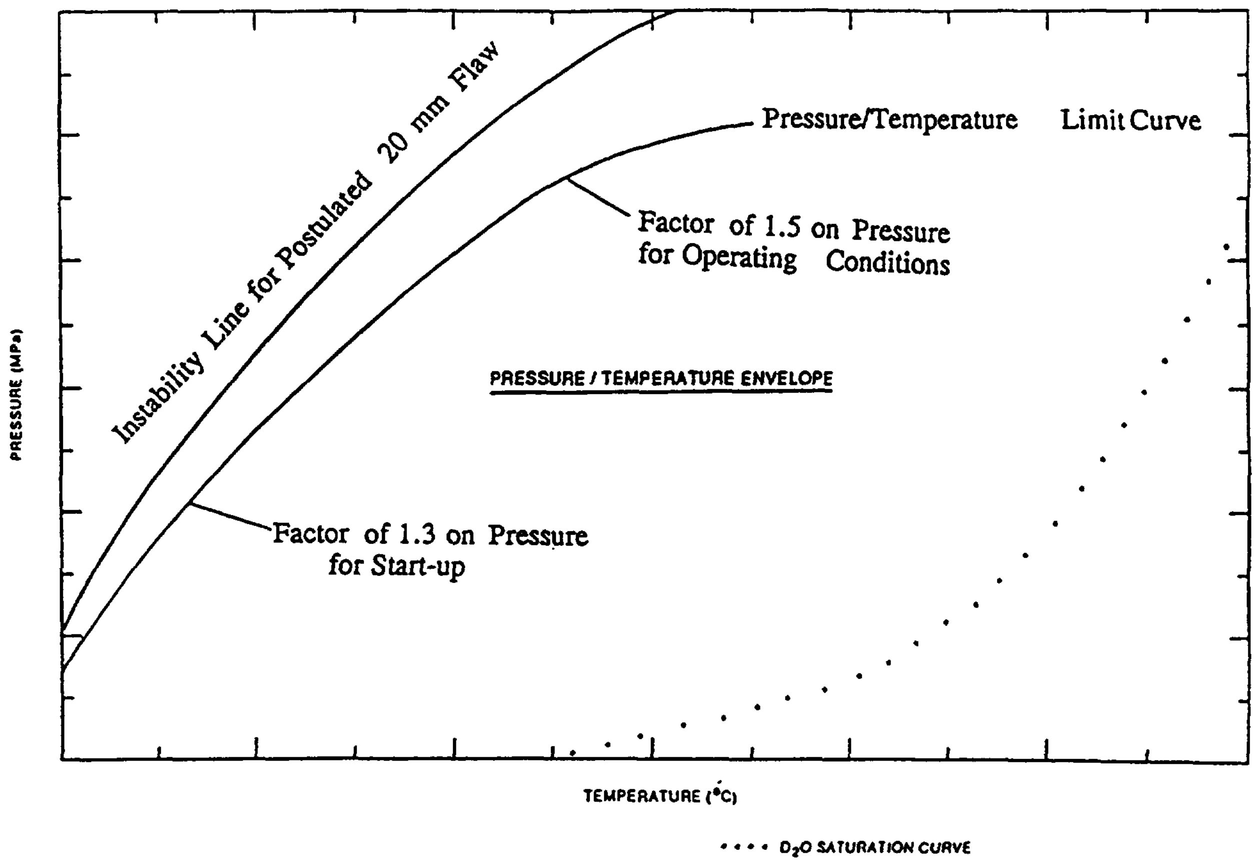


FIG. 31. Typical pressure/temperature operating envelope for pressure tubes.

6.6. SUMMARY

There is a large safety margin associated with having a pressure tube rupture due to DHC. Conservative assessments using upper/lower bound property values show that the probability of DHC is low. Furthermore, defence in depth is provided by conservatively demonstrating that even if DHC did occur, its consequence would be LBB.

A large scale engineering effort has produced comprehensive Fitness-for-Service Guidelines for the analysis of indications detected in cold-worked Zr-2.5%Nb pressure tubes. These Guidelines are presently in draft form and are undergoing evolutionary refinements. Nevertheless, they are currently being applied to standardize pressure tube fitness-for-service assessments. Analyses performed using the Guidelines have identified several conservatisms in both the data and the analytical methodologies utilized. Ongoing research programmes are aimed at improved interpretation of the databases and improved understanding of the failure mechanisms, in an effort to reduce unnecessary conservatisms.

A large number of pressure tube research programmes are currently being funded by the CANDU Owners Group (COG) to address:

(i) Corrosion and deuterium ingress,
(ii) DHC and fracture,
(iii) Irradiation damage and deformation, and
(iv) NDE and testing.

These programmes are focused on developing the necessary data and technical justifications for reducing the unnecessary conservatisms in the pressure tube flaw evaluation methodology. This work involves a significant amount of testing to produce the data required for statistical assessment of properties and improved understanding of the failure mechanisms. Most of the data to be produced in the next few years will come from irradiated tubes recently obtained from the retubing of Pickering units 3 and 4, and from specimens currently in high flux test facilities.

Revisions to the draft Guidelines will be issued in the future, to reflect improved interpretation of the databases and improved understanding of the failure mechanisms, as these improvements become available. The long term validity of pressure tube assessments must be periodically reviewed as the assessment technology, and the relevant data on the effect of irradiation and D ingress on pressure tube properties, is updated.

7. MITIGATION METHODS FOR PRESSURE TUBES

Section 4 of this publication outlined the operating behaviour of zirconium alloy pressure tubes and explained how feedback from the operation of the earliest tubes contributed to the evolutionary improvements in subsequent fuel channel designs and defined the optimum operating procedures for all reactors. Section 4 also mentioned briefly the major mitigation methods that have been used, or are planned, for pressure tubes. These pressure tube mitigation methods are discussed below. Section 7.1 summarizes the operational measures/practices for mitigating ageing mechanisms, and the subsequent sections describe the maintenance methods for mitigating ageing effects.

7.1. OPERATING GUIDELINES TO MANAGE PRESSURE TUBE AGEING

(a) Minimizing debris damage

Construction debris, such as metal turnings left in the primary heat transport system, can get trapped in the fuel and cause pressure tube fretting damage during the first year or two of operation [3]. The amount of debris that could lead to fretting damage can be minimized during construction by a high level of cleanliness, and during commissioning by the use of strainers at the inlet end of channels.

(b) Water and annulus gas chemistry control

Zirconium alloy pressure tubes are subject to corrosion by the slightly alkaline heavy water coolant that flows inside them, with some of the deuterium resulting from this corrosion process being absorbed by them. To minimize this corrosion and deuterium pickup, the pH of the coolant, as well as its deuterium and oxygen contents are tightly controlled.

In addition, O_2 is now added to the dry CO_2 which the CANDU annulus gas system circulates through the annulus of each fuel channel to ensure that a protective oxide is maintained on the pressure tube external surface, as a barrier to hydrogen and/or deuterium ingress into pressure tubes from the annulus. It is also important to ensure that there is a continuous flow of this gas through all channel annuli.

(c) Special operating guidelines

If the pressure tubes of a specific reactor are associated with a unique condition that increases the probability of delayed hydride cracking (DHC) occurrence, its operating procedures may be appropriately modified. Special procedures may be defined to ensure there continues to be a high confidence that tubes will not exhibit DHC, as well as to provide defence in depth by ensuring tubes would exhibit leak before break (LBB), if DHC does occur [9, 10]. As discussed in Section 6, pressure tube Fitness-for-Service Guidelines are being developed to provide criteria and assessment methodologies for addressing any unique conditions that may exist [26]. An example of special, unit specific, operating guidelines is an allowable pressure/temperature envelope such as the one shown in Figure 31.

7.2.MAINTENANCE METHODS FOR MITIGATING AGEING EFFECTS

7.2.1. TUBESHIFT

With the exception of the earlier reactors, whose design did not anticipate the full impact of the irradiation enhanced pressure tube deformation, it is expected that pressure tube deformation will not limit the life of CANDU and Indian fuel channels. Only the early reactors do not have sufficient fuel channel bearing length to accommodate all of the irradiation enhanced elongation that is expected to occur during their 30 year design life. The fuel channels for the first seven commercial CANDU reactors, and the first seven Indian reactors, have required an ongoing programme of axial repositioning (TUBESHIFT) to ensure they remain supported by their bearings as they elongate. The channels for the six earliest commercial CANDU reactors are expected to be replaced before they have operated for their entire 30 year design life. The four oldest reactors have already had all of their fuel channels replaced (see Section 7.4 for a discussion on retubing), which has extended the life of these reactor cores 10 to 15 years beyond their initial 30 year design life.

TUBESHIFT extends pressure tube life prior to retubing. It involves releasing the fuel channel from its axial restraint, axially shifting it on its bearings, and then locking the axial restraint to accommodate continued pressure tube elongation. Such repositioning was done for Pickering units 3 and 4 in 1984 (both units were subsequently retubed in 1989–1992), for Bruce units 1 and 2 in 1987, and for Bruce unit 3 in 1988. Although additional axial repositioning is possible for the Bruce units 1 and 2, by shifting the fuel channels off their bearings at one end and designing a new support for them, no additional TUBESHIFT is expected to be done prior to retubing. Fuel channels in these units have only two spacers, and hence contact between pressure and calandria tubes can be prevented for only 15 to 20 years of operation. However, the channels in Bruce unit 3 have four spacers, and may undergo additional axial repositioning around 1998.

7.2.2. Spacer location and relocation (SLAR)

Of particular concern are the pressure tubes in the early units as there are known to be many displaced fuel channel spacers in these units. Many pressure tubes in these units are in contact with the cooler calandria tubes, giving rise to pressure tube cold spots where brittle hydride blisters may form. A major mitigation method developed for these units is SLAR, which is planned to be performed before hydride blisters start to form in pressure tubes that are in contact with their calandria tubes. To ensure the prevention of blister formation, susceptible channels are identified analytically by performing conservative blister formation assessments. This conservative strategy will be reviewed when additional information is obtained from operating tubes and the ongoing blister modelling work.

SLAR is planned at eight CANDU units whose fuel channels have four spacers that may be displaced from their design positions. This SLAR process [48, 50] can be carried out on all channels in a reactor core during a single outage, by using SLAR tooling and a specialized delivery system that are remotely operated, but requires a significant time to be installed at the reactor face. Alternatively, spacers can be relocated on a smaller scale without such a specialized delivery system by using SLARette tooling to relocate the spacers in a selected batch of fuel channels during normal maintenance outages. (Many units have already started this spacer relocation work, and in 1995 it was completed for the CANDU-6 unit at Point Lepreau in New Brunswick, Canada.)

7.2.2.1. SLAR

The SLAR programme has developed a process based on existing fuelling machine technology that is used on a continuous basis in a fully automated mode to locate and reposition the spacers that separate pressure and calandria tubes in the early CANDU fuel channels. The original requirements for SLAR were that it must be possible to perform this maintenance operation on channels that have operated for up to 100 000 EFPH (effective full power hours). The SLAR system has more than satisfied this requirement as unit specific assessments have shown that it can be used on channels with higher EFPH values. It is expected that the SLARing of a complete reactor will take 3–8 months (depending on the amount of spacer movement needed).

SLAR requires modification of one fuelling machine so that it becomes a SLAR delivery machine, as illustrated in Figure 32. This SLAR delivery machine then works in

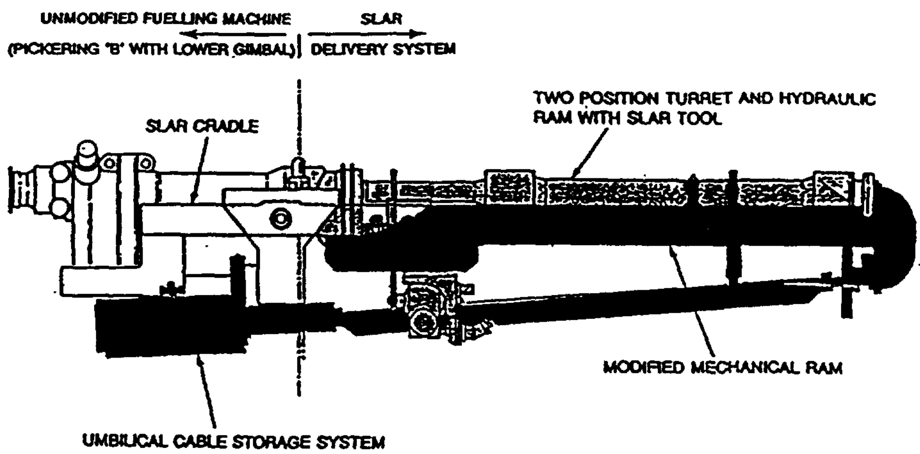


FIG. 32. SLAR delivery machine [54].

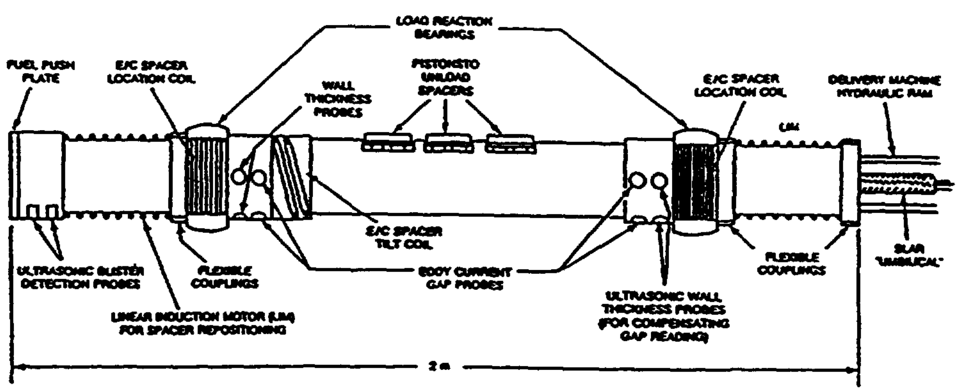


FIG. 33. SLAR tool [54].

conjunction with the other fuelling machine to defuel the channel and in effect becomes part of the Fuel Handling system with its operation run from the Fuel Handling Control Console. The SLAR delivery machine contains a mechanical ram which removes a channel closure and shield plug, and also contains a telescopic hydraulic ram which deploys the SLAR tool (see Figure 33) into the fuel channel. These two rams are indexed by means of a turret which is attached to a conventional fuelling machine magazine and snout assembly. A large drum is located beneath the magazine to store the umbilical cable which supplies the SLAR tool.

The SLAR tool contains all of the required inspection probes and is capable of moving spacers by remote control. The SLAR inspection computer system translates all eddy current and ultrasonic signals from the in-channel tool into various graphic displays. The SLAR tool, which is approximately 185 cm long, 8 to 10 cm in diameter and weighs about 80 kg, consists of the following components:

- A central bending tool for unpinching the spacers.
- Two LIMs (linear induction motors), one located on each end of the bending tool, for moving unpinched spacers.
- A spacer tilt eddy current coil mounted inboard of the free end location probe.
- Four pressure tube to calandria tube gap eddy current detection probes, one mounted at each of the 6 and 9 o'clock positions, adjacent to the location probes.
- Four ultrasonic pressure tube wall thickness probes, one mounted immediately inboard of each of the gap eddy current probes, for compensation of the gap readings.
- A cluster of six line focused ultrasonic blister detection probes, mounted on the free end of the tool, which scan the bottom 60° of the pressure tube, between 5 and 7 o'clock, for indications of cracked blisters.
- Articulated joints at the ram end of the tool, and between each of the LIMs and the bending tool to allow the tool to be sufficiently flexible to pass through a sagged pressure tube.
- The umbilical cable, a hybrid cable containing the power conductors for the LIMs, the signal cables for the inspection probes, and a hydraulic hose to supply pressurized D_20 for the operation of the pistons of the bending tool.
- The coupling which attaches the tool to the SLAR delivery machine push tubes.

Signals from the probes in the tool are transmitted through the umbilical cable, processed by the inspection system computers, and then displayed to the inspection system operators. Information is provided on spacer location, spacer tilt, pressure tube to calandria tube gap, pressure tube defects, wall thickness, blisters, etc.

7.2.2.2. SLARette

The SLARette system [54–56] evolved from the SLAR programme in response to a need by some utilities to avoid the long outage associated with SLAR and achieve the same results over several annual outages. The basic requirements for SLARette were therefore to make maximum use of developed SLAR technology, and to be quick and easy to install and remove. SLARette utilizes essentially the same in-channel tool and inspection system as SLAR, with the exception that the umbilical cable is considerably longer. However, the SLARette delivery system is quite different. The SLAR delivery machine consists of a modified fuelling machine and is capable of operating under totally remote control in automatic or semi-automatic mode, while the SLARette delivery machine is a smaller, less automated version which was designed to be used quickly on a limited number of fuel channels during regular annual maintenance outages.

The SLARette delivery system consists of a carriage platform assembly to support the delivery machine from the fuelling machine bridge, an elevating platform to position the delivery machine, a turret to provide access to non-isolated fuel channels, axial and rotary drives to control the position of the tool in the channel, a calibration tube to calibrate the functions of the tool and a D_2O supply system to fill and drain the delivery machine.

The primary and secondary turrets and transition tube are the main components of the SLARette delivery system which allows it to access a non-isolated fuel channel. The primary turret has a housing which attaches to the end fitting and is securely clamped by four mechanical jaws, which draw the turret housing against the end fitting "E" face. An O-ring provides the seal between the end fitting and the delivery machine. The rotor of the primary turret contains the channel closure removal tool and also allows the transition tube to slide axially through it into the end fitting.

When the channel closure is removed and stored in the primary turret, the turret rotors and transition tube are rotated to align the transition tube with the fuel channel. Then the transition tube, secondary turret, calibration tube and drive unit assembly are all advanced forward approximately 300 mm to allow the transition tube to enter the end fitting. A manually driven ball screw arrangement is provided to accomplish this axial movement. As the transition tube has the same internal diameter as the calibration tube, end fitting liner and fuel channel pressure tube, clear access to the fuel channel is provided for the SLARette tool. The pressure boundary of the primary and secondary turrets, transition tube, calibration tube and main seal assembly all are nuclear Class 1 components.

The axial drive is capable of moving the push tubes at between 5 and 55 mm/s. The rotation drive is capable of rotating the tool at a speed of 0.3 to 3 rpm. The rotational drive, and therefore the push tubes, can be rotated plus and minus 200 degrees, at which point limit switches are contacted to stop further rotation. Mechanical stops are provided at plus and minus 205 degrees in case of failure of the limit switches.

The calibration tube is a reactor grade pressure tube which has been specifically selected to have a circumferential wall thickness variation of at least 0.25 mm and have a sinusoidal wall thickness change. The calibration tube also has several machined features which can be used for tool calibration. These include a slanted groove, ultrasonic notches and flat bottom holes. This tube is required to calibrate the SLAR tool which is located inside it, as well as to provide a carrier/storage location for the tool. One end of the calibration tube, which forms part of the Class 1 pressure boundary, is attached to the axial and rotary drives.

The gap calibration device consists of a movable section of calandria tube which is held round by a series of nylon rings. The annular gap between pressure tube and calandria tube is measured by an LVDT.

The complete SLARette delivery and control system can be set up and operational in a CANDU-6 station in less than 48 hours. It can be dismantled and removed for storage in less than 24 hours. This system can perform SLAR activities on one channel during a 12 hour shift. It has been used successfully on several fuel channels during short maintenance outages for CANDU-6 reactors. It can also be used for the Pickering reactors. With some modifications, the system could be adapted to the Bruce end fitting/closure configuration.

7.2.3. Pressure tube replacement

The original design requirements for the CANDU fuel channel recognized that since knowledge of the long term behaviour of zirconium alloys was limited, the fuel channel should be replaceable. All CANDU fuel channel designs meet this requirement, and fuel channels have been replaced (in varying numbers) in all CANDU reactor types [24, 57].

There are two primary reasons for fuel channel replacement: either surveillance or maintenance. The latter can be subdivided into two further categories: Maintenance for individual fuel channels, and large scale fuel channel replacement for generic (life limiting) reasons.

The following gives an overview of the history of fuel channel replacement in CANDU reactors, followed by descriptions of the replacement process on an individual channel basis and for a complete reactor retubing. Design changes to the fuel channel to make it even more easy to install and replace are being incorporated in the design of future CANDU reactors.

7.2.3.1. Overview of fuel channel replacement history

(a) Single fuel channel replacement (SFCR)

It has occasionally been necessary to replace an individual fuel channel to correct a problem with it. Although the design, manufacturing, installation and inspection improvements associated with the current generation of fuel channels minimize the possibility of problems occurring for the more recently fabricated fuel channels, proven effective tooling and procedures for performing single fuel channel replacements (SFCR) are available.

One of the first fuel channels replaced in a CANDU reactor was at the Douglas Point Generating Station in Ontario. At this 200 MW prototype station, fuel channel replacement was required several months after the unit started up in 1964 as a result of an improperly installed vertical reactivity mechanism tube fretting against an adjacent calandria tube, eventually wearing a hole in it. Although the fuel channel had been designed for replacement, no tooling was available to do the job. Thus the reactor was shut down for several months while the tooling was designed and fabricated. Both end fittings, the pressure tube, calandria tube, and spacers were eventually replaced.

The first pressure tube removal and replacement at the NPD prototype station was in 1967 using tooling that was designed and fabricated for use in this reactor.

In 1974, LBB was exhibited by fuel channels in the Pickering Generating Station that had been assembled using a procedure where the rolling tool was inserted too far inboard. 17 pressure tubes in unit 3, and 52 tubes in unit 4 were replaced [24]. Tooling and equipment were designed with the units shut down so that pressure tubes, end fittings, and spacers could be replaced.

In order to prevent excessive outage durations for unplanned individual fuel channel replacements, the SFCR programme was put in place 15 to 20 years ago to develop, and provide stations with the necessary tooling and equipment to replace individual fuel channels.

(b) Large scale fuel channel replacement (LSFCR)

In 1983, the failure of a Zircaloy-2 pressure tube in Pickering unit 2 led to a generic concern for the integrity of all Zircaloy-2 pressure tubes and the decision was made to replace all of the fuel channels in Pickering units 1 and 2 to remove their Zircaloy-2 pressure tubes. Unit 2 was shut down from August 1983 (the time of the incident) until October 1988. Unit 1 was shut down in late 1983, and returned to power in September 1987.

Four other early CANDU units will also have all of their pressure tubes replaced before they have operated for their 30 year design life. This retubing has already been completed at the four earliest CANDU commercial units (Pickering units 1 to 4), with these units now being able to operate for 10 to 15 years beyond their original 30 year design life. The work on Pickering unit 3 began in 1989, and work on unit 4 started in 1991, after the return to power of retubed unit 3.

Bruce unit 2 was shut down in 1995. In a few years, Bruce unit 1 will either be retubed, or shut down if it is also not needed for production of electricity in the near term, with retubing being delayed until Ontario Hydro needs additional electricity.

The fuel channels of all other operating CANDU units have Zr-2.5%Nb pressure tubes, sufficient bearing lengths for the elongation of these pressure tubes, and four spacers, so it is expected that they can be operated for their full 30 year design lives.

Large scale fuel channel replacement is an essential part of plant life extension for CANDU reactors. Other plant components, both in the NSS and the balance of plant have much less severe service conditions than the pressure tubes. Hence the useful life of a CANDU plant can easily be extended to about sixty years with pressure tube replacement. In order to maximize the capacity factor of CANDU stations, the requirement for fuel channels has been changed from "can be replaced" to "be quickly replaced".

7.2.3.2. Fuel channel replacement process

The steps involved in both single and large scale fuel channel replacement are very similar, however the type of tooling and equipment varies greatly in terms of ease of use and degree of automation. The preparation for a planned single channel replacement is in the order of weeks, whereas it requires two to three years of lead time to adequately prepare for a large scale fuel channel replacement.

(a) SFCR

The first step in an SFCR operation involves the defuelling of the channel being replaced and its isolation from the rest of the PHTS. Defuelling is accomplished using the fuelling machines. Flow restrictors are then installed in the channel to limit the flow through its feeders so that freeze plugs can be formed in the feeders by locally cooling them with liquid nitrogen. This allows the channel to be drained, and permits removal of the capscrews which attach the feeder pipe to the end fitting. Blanking flanges are installed on the feeder pipe so that the freeze plug can be released. The channel is then ready for removal.

Crews, working from a shielded cabinet installed on the fuelling machine bridge, use manual tools to remove the fuel channel positioning assembly. The welds between the bellows and the attachment ring on the end fitting are cut by machining. The pressure tube is cut at its midpoint, and adjacent to each end fitting. The end fittings are then withdrawn into shielded flasks. The two halves of the pressure tube and the spacers are then pushed into a separate shielded flask.

The new pressure tube, protected by a sleeve, is inserted into the calandria tube, and the protective sleeve is withdrawn. Four spacers are inserted, two from each end of the reactor. Tooling is used to hold the pressure tube in position while the first end fitting is installed and the rolled joint is made. The second end fitting is then installed, and the second rolled joint is made. The welds are then made between the existing bellows and the new end fitting attachment rings. Next new positioning assemblies are installed. The final step is to re-establish the freeze plugs in the feeders so that the feeder blanking flanges can be removed, and the feeders can be reconnected. Closure plugs are inserted and the channel is reflooded and refuelled.

During the replacement of the fuel channel many quality control checks are made, including visual and dimensional inspections. The joints are leak and pressure tested after they are made.

(b) LSFCR

In principle an LSFCR operation is the same as performing many SFCRs. In practice, however, the jobs are quite different due to the amount of radioactive components to be handled, the number of personnel involved, and the need to minimize the radiological exposure to as low as reasonably achievable. As the outage is of considerably longer duration, many systems have to be placed in a "lay up" mode. For example, the moderator system is drained and dried, and placed under a helium cover gas. This protects the system as well as minimizing the risk of exposure of the work crews to tritium.

There are four distinct stages to a large scale fuel channel replacement operation: Preparation, removal, installation, and re-commissioning.

Preparation begins with the defuelling of the reactor, which is accomplished using the fuelling machines. During this phase the activity of the primary heat transport system is lowered as much as possible to reduce the gamma fields. Mild chemical decontamination (CANDECON) has generally been used. The heavy water is then drained from the upper half of the PHTS (down to the level of the inlet and outlet headers). Each channel is then re-visited by the fuelling machines in order to drain the water out of each channel and its associated feeders. The system is then vacuum dried to recover as much of the heavy water lying in the low points of the channel and other portions of the system as possible. This process saves on the expense of the replacement heavy water, and minimizes any tritium hazard. The system is flushed several times with light water to ensure that as much as possible of the (tritium containing) heavy water has been removed.

The removal phase begins with the installation of shielded cabinets on the fuelling machine bridge; installation of services in the vault (power, air, communications, etc.); and installation of robotic equipment to be used during the removal of the high activity components. The first step is to remove the feeder capscrews on every channel. The end fittings are then cut in half and the low activity outboard portions are removed without the need for shielded flasks. This also minimizes the amount of high level waste to be stored. The

TABLE V. SUMMARY OF CANDU PRESSURE TUBE KEY AGEING ISSUES AND THE MAJOR LIFE MANAGEMENT ACTIONS

Ageing mechanism	Potential consequences	Units affected	Major ageing management action
Irradiation enhanced deformation	Deformation exceeds design allowance.	Pickering 1 to 4, Bruce 1 & 2	All tubes have been (or will be) replaced before achieving their 30 year design life.
		Bruce 3	Axial repositioning.
		All other units	Monitor deformation of operating tubes and material testing in fast flux facilities.
DHC initiating from hydride blister (at PT/CT contact)	Tube rupture	Pickering 1 to 4, Bruce 1 & 2	All tubes have been (or will be) replaced before achieving their 30 year design life.
		Units with 4 potentially displaced spacers	Spacers to be repositioned before blister initiation/growth can occur, which makes the residual/remaining life of pressure tubes sufficient to achieve their 30 year design life with a large safety margin on tube rupture.
		All other units	No special maintenance needed.
DHC initiating from a stress concentration (e.g. a tube flaw)	LBB	Early units	Individual tube replacements to remove specific large stress concentrations.
		Later units	No special maintenance needed since all known sources for significant tube flaws have been eliminated.
Changes of tube properties during operation	LBB or rupture	All units	Monitor properties of tubes in lead unit(s) and material testing in fast flux facilities to ensure that the residual/remaining life of Zr-2.5%Nb pressure tubes is sufficient to achieve their 30 year design life with a large safety margin on tube rupture.

welds between the bellows (which are left in place), and the bellows attachment rings on the end fittings are removed by machining. The inboard stubs of the end fittings are severed from the pressure tubes.

Remote handling techniques are then used to extract the inboard stubs of the end fittings from the lattice tubes and lower them to the vault floor, where they are put into flasks. When a flask is full it is removed from the vault, and the radioactive components are transported to the on site storage area. The pressure tubes and spacers are pushed from one side of the reactor into a cylindrical metallic container and are lowered by remote handling techniques to the vault floor on the other side of the reactor where they are inserted into flasks. Again, the full flasks are transported to the on site storage area.

The first step in the installation phase involves the insertion of fuel channel sub-assemblies. A sub-assembly consists of a pressure tube rolled into one end fitting, with a protective sleeve around the pressure tube. A magazine containing the four spacers is attached to the protective sleeve, at the end next to the end fitting. After the sub-assembly is inserted, the protective sleeve and magazine are withdrawn, which deposits the spacers in their design locations. After verifying the locations of the installed spacers, the second end fitting is inserted and its rolled joint is made. The welds between the existing bellows and the new bellows attachment rings on the end fitting are made, and the positioning assembly is re-installed. The final step is the reconnection of the feeders.

New fuel, shield plugs, and closure plugs are installed in the channels as part of the re-commissioning phase. The systems are restored from their lay up state, and the reactor is re-commissioned, generally using the same procedures that were used for the initial start up of the plant.

7.3. SUMMARY

Feedback from the operation of the earliest CANDU pressure tubes has contributed to the evolutionary improvement of subsequent fuel channel designs, and helped define the inspection/maintenance required to manage the ageing of operating pressure tubes. The ageing effect that has caused the most significant Zr-2.5%Nb pressure tube problems is DHC. Thus, the major actions taken to mitigate ageing for CANDU pressure tubes have been focused on DHC, as indicated in Table V.

Special operating guidelines (for example, allowable pressure/temperature envelopes such as that shown in Figure 31) have been defined whenever necessary to provide a high confidence that CANDU pressure tubes will not exhibit DHC. The AGS chemistry is carefully controlled to maintain a protective oxide on the pressure tube external surface, which creates a barrier against the ingress of hydrogen/deuterium into pressure tubes from the annulus. The gas flowing through each fuel channel annulus is also monitored continuously for moisture using very sensitive leak detection equipment to ensure LBB requirements are met for pressure tubes. Conservative assessments are periodically performed for each CANDU reactor to demonstrate this LBB behaviour. Such assessments use the unit's operating/shutdown procedures and appropriate irradiated pressure tube properties (hydrogen/deuterium concentration, critical crack length, DHC velocity, the anticipated distribution of flaws in the tubes, etc.). These properties are obtained from various sources including inspections on operating pressure tubes, results of tests conducted on removed pressure tubes, and the pressure tube research programme findings.

Of particular concern are the pressure tubes in the early reactors. A major mitigation method developed for these reactors is SLAR, which needs to be performed before 15 to 20 years of operation are exceeded. It can be carried out on a large scale during a single outage by relocating displaced fuel channel spacers in all of the channels of a reactor core using SLAR tooling in conjunction with a specialized delivery system that is remotely operated. However, SLAR requires a relatively long outage. SLARette tooling provides an alternative method to relocate the spacers in a selected batch of fuel channels during normal annual maintenance outages as it does not need a fuelling machine based specialized delivery system.

Large scale fuel channel replacement for a CANDU reactor, which is currently performed using a combination of manual and remote tooling from inside shielding cabinets located on the fuelling machine bridge, takes about one and a half years. Fuel channel designs for future reactors are being designed with an emphasis on ease and speed of replaceability and hence are expected to require a significantly reduced retubing outage duration [5].

The ability to replace pressure tubes, which experience more severe service conditions than most CANDU components, can permit plant life extension to about 60 years, provided other components allow this.

8. PRESSURE TUBE AGEING MANAGEMENT PROGRAMME

The information presented in this publication suggests that pressure tube degradation caused by delayed hydride cracking and other age related mechanisms continues to be a significant safety and economic concern for early pressure tubes. Therefore, systematic pressure tube ageing management programmes are needed.

The preceding sections of this TECDOC dealt with the key elements of a pressure tube ageing management programme (AMP) whose objective should be to maintain the fitness-for-service of the pressure tubes at a nuclear power plant throughout their service life. This section and Figure 34 show how these elements are integrated within a plant specific pressure tube AMP. Such an AMP should be implemented in accordance with guidance prepared by an interdisciplinary pressure tube ageing management team organized at a corporate or owners group level.

A comprehensive understanding of a pressure tube, its ageing degradation, and the effects of the degradation on the pressure tube's ability to perform its design function is the fundamental element of an AMP. This understanding is derived from a knowledge of:

- the design basis (including applicable codes and regulatory requirements),
- the design and fabrication (including material properties and specified service conditions),
- the operation and maintenance history (including commissioning and surveillance),
- inspection results, and
- generic operating experience and research results.

Sections 1.1, 2, 3, and 4 of this TECDOC contain information on important aspects of the understanding of pressure tubes and their ageing.

Two CANDU pressure tube ruptures have occurred to date, one while Pickering unit 2 was operating and another during a cold pressurization test at Bruce unit 2. The plant was properly cooled following the one hot rupture, and radioactive releases were well below regulatory limits for both events. However, these ruptures, as well as the degradation of early tubes reported in Section 4, have been a significant concern to regulators.

In order to maintain the fitness-for-service of a pressure tube, it is necessary to control within defined acceptable limits, the age related degradation of the tube. Ageing degradation control consists of the following elements, based on an understanding of pressure tube ageing:

- prudent operation consistent with operational guidelines aimed at mitigating degradation mechanisms (Section 7.1),
- inspection and monitoring consistent with requirements aimed at timely detection and characterization of any degradation (Section 5),
- assessment of any observed degradation in accordance with appropriate guidelines to determine fitness-for-service (Section 6), and
- repair or replacement to correct unacceptable degradation (Section 7.2).

A pressure tube AMP is a mixture of the above elements and specific ageing management actions designed to minimize, detect and mitigate ageing degradation before pressure tube safety margins are compromised. This mixture reflects the level of understanding of pressure tube ageing, the available technology, the regulatory/licensing

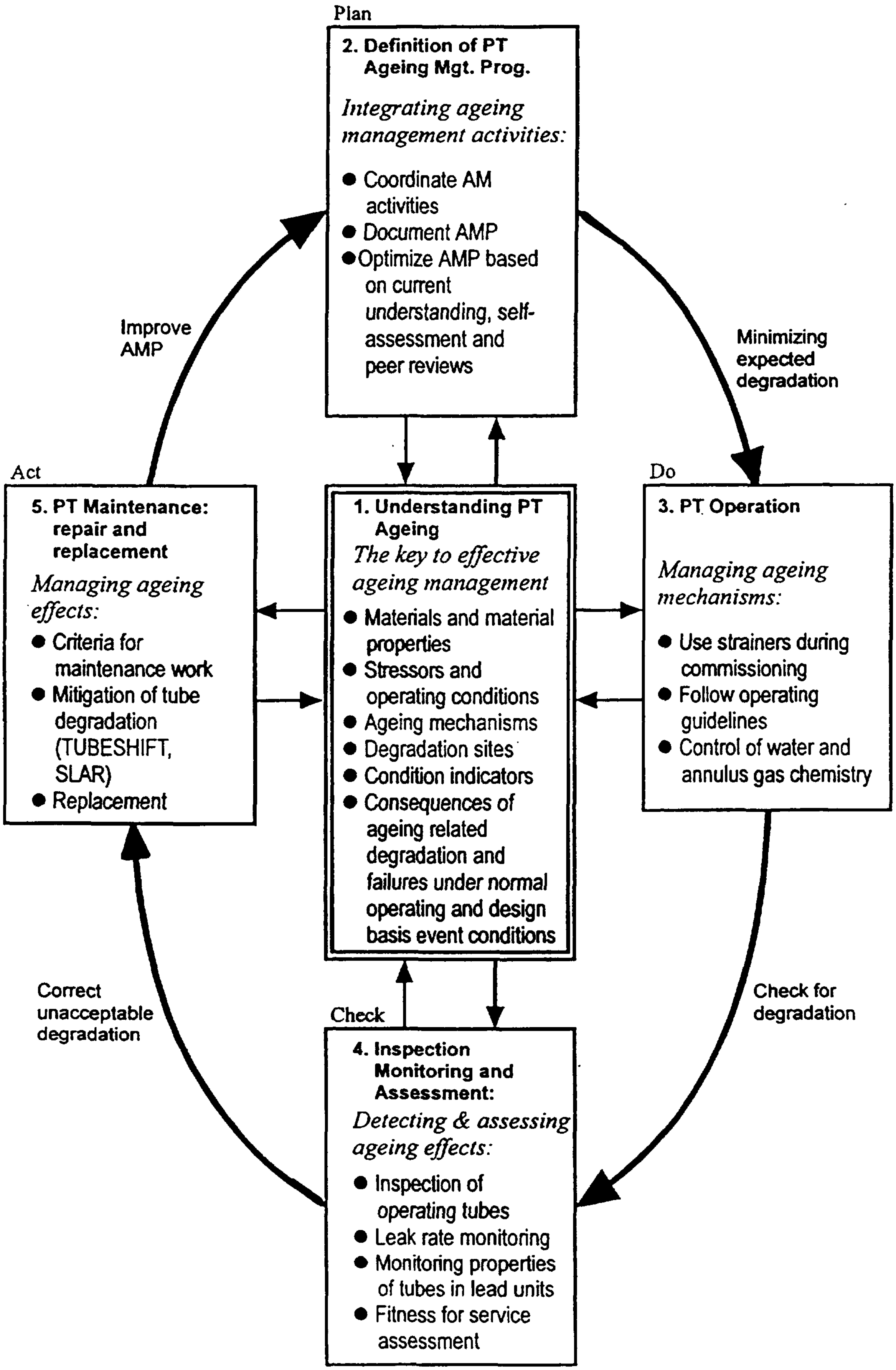


FIG. 34. Key elements of a systematic pressure tube ageing management programme and their interfaces.

requirements, and plant life management considerations/objectives. Timely feedback of experience is essential in order to allow ongoing improvement in the understanding of the pressure tube ageing degradation and in the operation, inspection, assessment and maintenance methods and practices. The following sections address the main features and interfaces of key elements of a pressure tube AMP, as illustrated in Figure 34.

8.1. UNDERSTANDING PRESSURE TUBE AGEING AND FEEDBACK OF OPERATING EXPERIENCE

Understanding pressure tube ageing is the key to effective management of pressure tube ageing. It is the key to:

- integrating ageing management activities within a systematic AMP,
- managing ageing mechanisms through prudent operating procedures and practices,
- detecting and assessing ageing effects through effective and practical inspection, monitoring and assessment methods, and
- managing ageing effects using proven maintenance methods.

This understanding consists of a knowledge of pressure tube materials and material properties, stressors and operating conditions, likely degradation sites and ageing mechanisms, condition indicators/data needed for the assessment and management of pressure tube ageing, and the consequences of age related degradation and failures both under normal operating conditions and design basis event conditions.

The understanding of pressure tube ageing is derived from the pressure tube baseline data as well as operating and maintenance histories. This understanding should be updated on an ongoing basis to provide a sound basis for the improvement of the AMP and operating, inspection, monitoring, assessment and maintenance methods and practices.

The pressure tube baseline data consists of:

- the performance requirements,
- the design basis (including codes, standards, regulatory requirements),
- the original design,
- the manufacturer's data (including materials data), and
- the commissioning data (including inaugural inspection data).

The pressure tube operating history includes pressure-temperature records, system chemistry records and significant event reports. The pressure tube maintenance history includes the inspection records and assessment reports, as well as the type and timing of maintenance performed. Retrievable up-to-date records of this plant specific information are needed for making comparisons with applicable external experience.

External experience consists of the operating and maintenance experience of:

(a) pressure tubes of similar materials of construction and fabrication,
(b) pressure tubes operated under similar water chemistry, and
(c) relevant research results.

This information has to be kept current using feedback mechanisms provided, for example, by owners groups. External experience can also be used when considering the most appropriate inspection method, maintenance procedure and tooling.

8.2. DEFINITION OF PRESSURE TUBE AMP

Existing programmes relating to the management of pressure tube ageing include operations, surveillance and maintenance programmes as well as operating experience feedback, research and development (R&D) and technical support programmes. Experience shows that ageing management effectiveness can be improved by integrating and co-ordinating relevant programmes and activities within a systematic ageing management programme. Safety authorities increasingly require licensees to define such AMPs for selected systems, structures and components (SSCs) by documenting relevant programmes and activities and their respective roles in managing SSC ageing. A definition of a pressure tube AMP includes also a description of mechanisms used for programme co-ordination and continuous improvement. The continuous AMP improvement or optimization is based on current understanding of pressure tube ageing and on results of self-assessments and peer reviews.

8.3. PRESSURE TUBE OPERATION

Operating conditions and practices significantly influence pressure tube degradation and, therefore, are the primary means for the staff of a nuclear power plant to minimize degradation caused by potential ageing mechanisms. These practices include water and annulus gas chemistry control, as indicated in Section 7.1. A chemistry programme must be developed for the specific conditions of the plant and maintained to minimize the corrosion of the pressure tubes. The programme must include adequate sampling and specific action levels, up to and including plant shutdown.

8.4. INSPECTION, MONITORING AND ASSESSMENT

Pressure tube inspection, monitoring and assessment activities are designed to detect and characterize significant component degradation before the pressure tube safety margins are compromised. Together with an understanding of the pressure tube ageing degradation, the results of the pressure tube inspections provide a basis for decisions regarding the type and timing of maintenance actions and decisions regarding changes in operating conditions and practices to manage detected ageing effects.

8.4.1. Inspection and Monitoring

Current inspection and monitoring requirements and techniques for pressure tubes are described in Section 5. The rigor and extent of the non-destructive inspection increases dramatically if pressure tubes develop problems, supplemented by destructive examinations such as metallography on pulled tubes.

It is extremely important to know the accuracy, sensitivity, reliability and adequacy of the non-destructive methods used for the particular type of suspected degradation. The performance of the inspection methods must be demonstrated in order to rely on the results, particularly in cases where the results are used in fitness-for-service assessments. Inspection methods capable of detecting and sizing expected degradation are therefore selected from those proven by relevant operating experience. Current methods used for the inspection of pressure tubes are described in detail in Section 5.1.

8.4.2. Assessment

Safety margins are part of the design and licensing requirements of a nuclear power plant to ensure the integrity of pressure tubes under both normal and accident conditions. If an acceptance level of CSA Standard N285.4 is exceeded, a fitness-for-service assessment is done to assess the capability of a CANDU pressure tube to perform its required function, within the specified margins of safety, during the entire operating interval until the next scheduled inspection.

Fitness-for-service assessments have used a variety of methods in response to the particular conditions and circumstances present at the time of the assessment. Section 6 of this report describes the fitness-for-service guidelines used for CANDU pressure tubes. To date assessments have been deterministic analyses of the specific types of degradation detected. In addition, a probabilistic assessment is allowed to demonstrate fitness-for-service. This method requires probabilistic calculations to assess the conditional probability of tube failures. Risk calculations should take into account the probability that some degraded tubes may be inadvertently left in service and the fact that the probability of non design basis events may increase with time due to other age related degradation.

Although probabilistic assessments of pressure tubes have not yet been used, when appropriate databases exist this method will provide a useful way for handling uncertainty while avoiding excessive conservatism.

8.5. MAINTENANCE: REPAIR AND REPLACEMENT

There are a variety of maintenance actions that can be used to manage any pressure tube ageing effects that are detected by inspection and monitoring (see Section 7.2). Decisions on the type and timing of the maintenance actions are based on an assessment of the observed ageing effect(s), available decision criteria, an understanding of the applicable ageing mechanism(s), and the effectiveness of available maintenance technologies.

Maintenance actions for mitigating pressure tube degradation include preventive methods, such as SLAR and TUBESHIFT, and corrective/repair methods, such as individual tube replacement(s). Maintenance actions for pressure tubes with highly susceptible material, or with other inadequate design features, or exposed to poor water or annulus gas chemistry may have to include replacement of all of the pressure tubes in a reactor before their design life is reached.

8.6. SUMMARY

There is a high level of confidence that the Zr-2.5%Nb pressure tubes in the current generation of CANDU units, and tubes in the earlier CANDU units after SLAR implementation, have sufficient residual life to allow them to operate reliably, with a large safety margin on tube rupture, for at least their design life of 30 years at 80% capacity. However, since no Zr-2.5%Nb pressure tubes have yet operated for 30 years, prediction of their end-of-life conditions comes from extrapolation of data obtained from tubes removed from reactors that have operated for less than 30 years, and from theoretical predictions based on research results. Continued monitoring of pressure tube performance in operating reactors at regular intervals, particularly for the lead units, and continued material testing to high fluences in fast flux facilities of test reactors is required to confirm that Zr-2.5%Nb tubes behave as anticipated, or to provide lead time to manage their ageing.

When pressure tubes do reach their end-of-life, they can be replaced. Pressure tubes have been replaced (in varying numbers) in all CANDU reactor types, for both the removal of surveillance tubes and maintenance purposes. This ability to replace pressure tubes can permit extension of the life for most CANDU power stations to about 60 years, provided other components can operate reliably this long.

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ABBREVIATIONS

AGS	annulus gas system
AMP	ageing management programme
CANDU	Canada deuterium–uranium (reactor)
CCL	critical crack length
DBE	design basis earthquake
DHC	delayed hydride cracking
LBB	leak before break
NPD	Nuclear Power Demonstration (reactor)
PHTS	primary heat transport system
PHWTR	pressurized heavy water tube reactor (India)
SSCs	(plant) systems, structures and components
TSS	terminal solid solubility

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