Operating performance and reliability of CANDU PHWR fuel channels in Canada

Performance et fiabilité en service des canaux de combustible de réacteurs canadiens CANDU à eau lourde sous pression (PHWR)

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PERFORMANCE ET FIABILITÉ EN SERVICE DES CANAUX DE COMBUSTIBLE DE RÉACTEURS CANADIENS CANDU À EAU LOURDE SOUS PRESSION (PHWR)

par B. Strachan et D.R. Brown

Résumé

A la différence des centrales nucléaires à réacteurs à eau sous pression (PWR) qui comportent une seule grande cuve sous pression, les centrales nucléaires CANDU comportent de nombreux canaux de combustible à haute pression de petit diamètre. La bonne performance en service des canaux de combustible de réacteurs CANDU a contribué pour une large part à l'accession des centrales nucléaires CANDU à la première place mondiale du point de vue de l'exploitation.

En date du 31 décembre 1982, il y avait 7,480 canaux de combustible installés dans 18 réacteurs CANDU d'une puissance supérieure à 500 MW(e). On a déclaré huit de ces réacteurs "en service"; ceux-ci ont accumulé 24,000 années-canaux de combustible d'exploitation.

Les seuls grands problèmes rencontrés en service avec les canaux de combustible ont été des fissures avec fuites dans 70 de ceux-ci et un fluage axial plus important avec les premiers réacteurs; ce fluage a été plus important que celui que avait été prévu au départ lors de la conception.

On a remédié à ces deux problèmes pour tous les réacteurs CANDU qui ont été construits depuis la centrale Bruce 'A'. Les réacteurs plus récents devraient avoir une performance encore meilleure.

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OPERATING PERFORMANCE AND RELIABILITY OF CANDU PHWR FUEL CHANNELS IN CANADA

by B. Strachan and D.R. Brown

Abstract

CANDU nuclear plants use many, small-diameter high pressure fuel channels unlike PWR nuclear plants which have a single, large pressure vessel. Good operating performance from the CANDU fuel channels has made a major contribution to the world-leading operating record of the CANDU nuclear power plants.

As of 1982 December 31, there were 7,480 fuel channels installed in 18 CANDU reactors over 500 MW(e) in size. Eight of these reactors have been declared in-service and have accumulated 24,000 fuel channel-years of operation.

The only significant operating problems with fuel channels have been the occurrence of leaking cracks in 70 fuel channels and a larger amount of axial creep on the early reactors than was originally provided for in the design.

Both of these problems have been corrected on all CANDU reactors built since the Bruce GS 'A' station and the newer reactors should exhibit even better performance.

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1.0 INTRODUCTION

Commercial generation of electricity from nuclear power in Canada started in the province of Ontario. Ontario is located in central Canada, bordering the Great Lakes. It covers an area of 917,000 square kilometres and has a population of 8,700,000. The industrial capacity of the province is large compared to its population because industrially it serves a large part of Canada. Ontario Hydro, an electrical utility owned by the provincial government, is assigned the responsibility for generating electrical power. Electricity is generated via hydraulic, fossil fuel and nuclear power. Nuclear power forms 25% of the total installed electrical capacity, but produces 35% of the electrical power. Atomic Energy of Canada Limited designs the nuclear systems, and Ontario Hydro designs the balance of the plant, and constructs and operates the stations. All Canadian power reactors are of the CANDU Pressurized Heavy Water type (PHWR).

Currently, there are eight CANDU reactors greater than 500 MW(e) in operation in Ontario with twelve more in various stages of construction or planning. The 4 x 540 MW Pickering GS 'A' station is located on the north shore of Lake Ontario, 28 km east of the provincial capital, Toronto. The 4 x 750 MW Bruce GS 'A' station is located on the east shore of Lake Huron, 240 km northwest of Toronto. Repeat stations, Pickering GS 'B' and Bruce GS 'B' are in advanced stages of construction on the same sites. The construction of the 4 x 850 MW Darlington GS station is underway at a location 200 km east of the Pickering station.

The first unit of the Pickering GS 'A' station started producing electricity in 1971, April and the fourth unit in 1973, June. The dates for the Bruce GS 'A' station were 1977, September and 1979, January.

The operating performance of the Pickering GS 'A' and Bruce GS 'A' reactors has been outstanding. Based on life time capacity factors up to the end of 1981, seven of these reactors place in the first ten positions of the world's most reliably operating nuclear power stations. The eighth reactor is in eighteenth place. Table 1 shows the performance rankings and capacity factors for 131 reactors greater than 500 MW, as of 1981 December 31.

The main reason for the very high capacity factors attained by CANDU reactors is their ability to be refuelled while operating at full power and thereby avoid the refuelling shutdowns that occur in other types of reactors. Excellent reliability, however, has also made a major contribution to the CANDU reactor operating record. A measure of reliability can be gained via the incapability factor which is the ratio of 'energy not produced to perfect production' due to equipment incapabilities for a given period of time. Table 2 gives the incapability factors for major power station equipment for the Pickering GS 'A' and Bruce GS 'A' stations, including the fuel channels. Considering the arduous environment in which fuel channels operate their incapability factors compare very well with those of other major equipment. Table 3 gives the incapability factors specifically for the Pickering and Bruce fuel channels, for each year of operation from start-up.

Although the overall performance of the fuel channels has been excellent as Table 4 shows, there have been problems which have caused appreciable incapability factors for the years 1974, 75 and 76 in the Pickering G.S. 'A'. These incapacities were caused by cracks in the pressure tubes.

Another phenomenon which will eventually cause fuel channel incapability in the Pickering G.S. 'A' and the Bruce G.S. 'A' stations is pressure tube creep deformation. The zirconium alloy pressure tubes creep due to the combined influence of stress, temperature and irradiation. Provisions incorporated in the Pickering GS 'A' and Bruce GS 'A' fuel channel designs to accommodate creep are insufficient to last the 30 year design life of the stations. Fuel channel adjustments are expected to restore creep life and permit reactor operation for at least 30 years. Additional life, well beyond 30 years is possible by replacing all fuel channel pressure tubes.

The CANDU fuel channels are designed to be completely replaceable because it was anticipated that pressure tube creep might eventually lead to their renewal. This attention to replaceability at the design stage has made fuel channel replacement a relatively easy operation.[1]

2.0 DESIGN DESCRIPTION

The fuel channels of all CANDU reactors are very similar, and the fuel channel described here is that of the CANDU 600 MW reactor design which incorporates improvements that avoid the operating problems encountered in the Pickering GS 'A' and the Bruce GS 'A' reactors.

In a CANDU reactor, there is no reactor vessel as there is in light water reactors. Instead, the primary heat transport system coolant flows through the reactor core in a large number of identical fuel channels. The high-pressure fuel channels pass through the calandria which is a low-pressure vessel that contains the heavy-water moderator. By using fuel channels, the difficulties of a large volume, high-pressure and high-temperature reactor vessel are avoided.

A second feature of CANDU reactors is that they use natural uranium fuel (without enrichment) and so the use of low neutron absorption materials in the reactor core is an important requirement. All incore structure materials are zirconium alloys and in particular the portion of the fuel channels that pass through the core is made of high strength zirconium-niobium alloy. High-purity heavy water is used for the primary coolant and for the moderator.

Figure 1 shows the CANDU 600 MW calandria is a horizontal cylinder through which the 380 fuel channels pass. Each fuel channel goes through a passage created by the reactor endshield lattice tubes and a calandria tube. The annular space around the fuel channel is filled with an inert gas. This annulus is connected to a process system that incorporates moisture detecting instrumentation to warn of leaks from either the fuel channel, calandria tube or lattice tubes.

The fuel channel assembly is made up of three major components: the zirconium alloy pressure tube and two identical stainless steel end fittings. The end fittings provide a connection between the pressure tube and the feeder pipes for the coolant flow and also provide ports through which the fuelling machine can remove spent fuel from the channel and insert fresh fuel into it, while the reactor is operating.

Locked inside each end of the fuel channel is a channel closure (Figure 2) which can be remotely removed by the fuelling machine. The end fitting incorporates a latching groove and a sealing surface for the channel closure and an external latching shoulder and a sealing surface for the fuelling machine. A hardened-steel seal ring shrunk into the end fitting body improves the reliability of the closure sealing surface.

The feeder connection is on the side of the end fitting (Figure 3). Four bolts pass through a flange into holes tapped into the end fitting body. A hub welded to the feeder pipe, is tightly held by the flange against a solid metal seal ring.

The positioning assembly (Figure 4) provides axial restraint for the fuel channel. Only the positioning assemblies on one side of the reactor are engaged at any time, the assemblies on the other side being disengaged to permit axial motion resulting from thermal expansion and pressure tube creep elongation. Each positioning assembly is engaged for half of the reactor life so that the creep elongation of the pressure tube can be shared equally by the two ends of the fuel channel.

The channel annulus bellows (Figure 5) seals the gas filled annulus between the fuel channel and the calandria and lattice tubes, and permits axial motion. A small diameter tube welded to the bellows ferrule connects the annulus to the Annulus Gas System which circulates dry CO_2 gas and monitors the moisture content of this gas.

The outboard sliding journal/bearings can be seen (Figure 5) just inboard of the bellows. Both parts of the bearings are made of hardened tool steel. The shorter component is referred to as the journal ring.

The pressure tube is attached to the end fittings with roll-expanded joints (Figure 6). Although not permitted by the ASME Code in Class 1 applications, their use has been justified for this application by extensive testing and analysis. Immediately beside the pressure tube rolled joint is the liner tube rolled joint. This joint is a mechanical attachment and not a seal, since the liner tube contains flow holes to allow the primary coolant to flow from the fuel passage into the annulus between the liner tube and the end fitting body.

Inside the pressure tube is the fuel, supported axially at the flow outlet end of the fuel channel by the shield plug. The inner end of the shield plug is a tube with flow holes matching those of the liner tube. The outer end of the shield plug is solid to block core radiation and to divert the coolant through the flow holes into the liner tube end fitting annulus.

In the reactor core surrounding the pressure tube is the calandria tube, with garter springs in the gas-filled annulus between them (Figure 7). Outside the calandria tube is the heavy water moderator. The garter springs maintain the annulus between the pressure tube and calandria tube, and prevent the tubes from contacting.

3.0 OPERATING PROBLEMS

3.1 Delayed Hydride Cracking

3.1.1 Crack Mechanism

The delayed hydride cracking phenomenon was first observed in zirconium- niobium material in experiments in 1972. Since the Pickering pressure tube cracks in 1974, it has been intensively studied. We now understand the hydride cracking mechanism sufficiently to know what measures are necessary to avoid it. Remedial measures have been introduced in all CANDU reactors built since the Bruce GS 'A' station, and consequently, delayed hydride cracking is not expected to occur in these newer stations.

Delayed hydride cracking involves the repeated growth and fracture of zirconium hydrides at a stress raiser in the presence of a high tensile stress. The cycle is repeated until the process creates a through-wall crack. Research has established that to initiate delayed hydride cracking there must be hydrides present in the material, and there must also be:

(a) a tensile stress greater than 590 MPa at a smooth surface; or

(b) a combination of a lower stress with a notch or defect that acts as a stress raiser.

Hydrides form in zirconium alloys when the hydrogen content exceeds a level known as the 'terminal solid solubility' (TSS). At concentrations below this level the hydrogen remains in solution and hydrides cannot form. At higher concentrations, the hydrogen in excess of the TSS precipitates as small brittle hydrides. Therefore hydrogen can exist in the material in both forms; in solution and as a separate hydride phase. The amount of hydrogen that remains in solution increases with temperatures according to the solubility relationship for the material.

The hydrogen content of 'as-manufactured' pressure tubes is 10-15 ppm and at this concentration hydrides precipitate only at reactor cooldown temperatures. However, the hydrogen content of the tubes is increased by absorption of deuterium from the primary coolant and by diffusion from the stainless steel end fitting. Absorption from the coolant occurs uniformly along the length of the tube and is very small. Diffusion from the end fitting is greater, but is restricted to the region near the rolled joint because the rate of diffusion is very slow. The increase in hydrogen at the rolled joint raises the temperature at which hydrides can be present and eventually hydrides will be present near the rolled joint at reactor operating temperatures.

On cooling, hydrides precipitate in the circumferential-axial plane of the tube if the tube is not highly stressed. However, on cooling under the influence of a tensile hoop stress greater than about 175 MPa, the hydrides preferentially reorient towards the radial-axial plane. The higher the stress, the greater the degree of reorientation. Reorientation thus places the hydrides normal to the hoop stress and establishes the conditions for possible crack initiation.

3.1.2 Pickering GS 'A', Units 3 and 4

In 1974 August when the Pickering GS 'A' Unit 3 was being returned to service after a scheduled maintenance outage, primary heat transport water was found to be leaking into the annuli surrounding the fuel channels. To identify which fuel channels were leaking, a sensor was held against the end face of each fuel channel in turn and the noise generated by the leaking primary coolant was analyzed on acoustic emission equipment. Seventeen leaking channels were found and subsequent internal ultrasonic inspections revealed that the pressure tubes had cracked in the region immediately adjacent to the rolled joint between the pressure tube and the end fitting. The leaking channels were replaced and the reactor was returned to service.[2]

In 1975 May, Unit 4 was shutdown when similar leakage of heavy water was detected. Fifty-two channels were replaced due to cracks in the rolled joint region.

Examinations on the cracked tubes showed that the cracks had formed by the delayed hydride cracking mechanism as a result of very high residual stress in the portion of the pressure tube immediately adjacent to the rolled joint. An incorrect rolling procedure, which permitted the roll expander to be inserted too far into the pressure tube expanded the tube where it was not supported by the end fitting and in so doing produced very high residual stresses. These joints are referred to as 'overrolled'. The initial assembly clearance between the pressure tube and end fitting was the cause of additional residual stress. This was introduced in the tube as it was roll expanded into contact with the end fitting bore. Figure 8 shows the difference in roll expander positions and typical residual hoop stress profiles between properly rolled and 'overrolled' joints.

Since the Pickering reactors were returned to service in 1975 March and 1976 March, no further fuel channel leaks have occurred.

3.1.3 Bruce GS 'A', Unit 2

The Bruce GS 'A' rolled joints are similar in design to those in Pickering GS 'A' and the installation procedures were identical. The joints in Units 1 and 2 were made in 1972, prior to the discovery of the Pickering cracks and as a result were similarly improperly rolled. Consequently, the residual stresses in the Units 1 and 2 joints were about the same very high level which occurred in Pickering. To reduce the level of residual stress and avoid delayed hydride cracking, the regions of high residual stress in the Bruce GS 'A' Units 1 and 2 rolled joints were stress relieved. The Units 3 and 4 joints were correctly rolled, but they too were stress relieved as a precautionary measure and to reduce the residual stress caused by the initial assembly clearance.

Despite stress relieving, in 1982 February during a start-up of Unit 2 from a two-day shutdown, water was detected in the gas annulus system. Isotopic and tritium checks again confirmed the leak was primary coolant from a fuel channel. An acoustic emission scan established that a single channel was leaking. Ultrasonic inspection again confirmed that the pressure tube adjacent to the rolled joint was cracked. The leaking fuel channel was replaced and the reactor returned to service.

It was not until 1975, when the Pickering GS 'A' cracks were evaluated, that a need for stress relieving was established. As a result the Bruce GS 'A' joints experienced very high residual stresses for about two years. The high residual stresses alone were sufficient to gradually cause the formation of a small number of large radial hydrides. One of these hydrides cracked and the crack grew large enough that the addition of the system operating stress following stress relief was enough to allow it to continue growing during cold, pressurized-reactor shutdowns, until it penetrated the pressure tube wall.

As a result of the delay in stress relieving in Units 1 and 2, it is expected that more rolled joints in these units may eventually crack. The number will be dependent on the extent of hydride formation during the time the joints were highly stressed. The number of additional joints which may crack is thought to be very small.

3.1.4 Solutions to Delayed Hydride Cracking

3.1.4.1 Design

To reduce the high residual stresses experienced in Pickering GS 'A' and Bruce GS 'A' rolled joints, a modified rolled joint has been developed. The clearance between the pressure tube and the end fitting bore has been reduced so that the tube deformation during rolling is much less. The diametral assembly clearance in the modified joint ranges from an interference of 0.18 mm to a clearance of 0.05 mm, whereas the Pickering GS 'A' and Bruce GS 'A' designs had a clearance ranging from 0.18 to 0.53 mm. The new version is referred to as a 'zero clearance' rolled joint and it has significantly lower residual stresses. The very tight assembly fit achieved in the zero clearance rolled joint is accomplished in two steps; first by thermally expanding the end of the end fitting to permit insertion of the pressure tube, followed by roll expansion of the pressure tube into the end fitting.

The improper rolling procedures used for the Pickering GS 'A' and Bruce GS 'A' rolled joints were revised to avoid over-insertion of the roll expander and the rolled joint design was modified to make it less sensitive to roller location. These changes ensure that there will not be any further over-rolling or flaring of the pressure tube.

Figure 9 shows two typical residual hoop stress profiles for the zero clearance rolled joint. The upper profile is for a joint assembled with maximum permissible clearance. The lower profile is for a joint assembled with maximum permissible interference. Comparing these profiles with those in Figure 8 for the Pickering GS 'A' joints confirms the significant reduction in residual stress achieved by the zero clearance joint. Figure 10 gives the combined residual and operating stress for the zero clearance joint and shows that the peak stress is 200 MPa as compared with the stress of 590 MPa required to initiate cracking in a smooth surface.

3.1.4.2 Inspection

The significant reduction in total stress achieved in the zero clearance rolled joint will prevent delayed hydride cracking in pressure tubes with smooth surfaces. However, since the stress required to initiate cracking at a flaw is lower, it is important that pressure tubes are as defect-free as possible.

Pressure tubes manufactured to date have been inspected using an automated ultrasonic technique set to detect flaws greater than 0.08 mm deep. The tubes were inspected using normal and shear waves. Each tube was inspected continuously along its full length, in a spiral pattern. The inspection was conducted twice in opposite directions to increase coverage. Recently however, these inspection techniques were carefully re-examined. It was found that due to the nature of the ultrasonic reflections it was possible for large flaws to escape detection if they lay in a band 0.6 mm from the inside surface and at an angle of 5-10° to the surface. To ensure detection of such flaws a supplementary eddy current inspection was added to the inspection plan. This inspection was conducted on installed pressure tubes in all reactors under construction and will be integrated with the ultrasonic inspection for future reactor pressure tubes. The only significant flaw found was in the Cordoba reactor and the pressure tube containing the flaw was replaced.

With the introduction of the eddy current inspection, we are confident that pressure tubes containing flaws which could be susceptible to delayed hydride cracking, will be found and rejected.

3.2 Pressure Tube Creep

3.2.1 Creep Phenomena

During operation CANDU fuel channels are subjected to high pressure, elevated temperature and fast neutron flux. These conditions cause the zirconium alloy pressure tubes to creep, which results in increases in diameter, length and sag. These changes occur very gradually with operating time.

Pressure tube creep actually comprises three components, which are:

(a) thermal creep, which is dependent on stress and temperature;
(b) irradiation creep, which is predominantly dependent on fast neutron flux, but also on stress and temperature;
(c) irradiation growth, which is dependent on fast neutron flux and temperature, but not stress.

For the purpose of this presentation the term 'creep' will include all three of these components.

Creep in the three principal directions can be expressed by strain rate equations that include factors which account for anisotropic zirconium properties and changes in pressure tube dimensions can be calculated. The rates of these changes vary in proportion to fast flux, stress and exponentially with temperature.

Early creep predictions were made from empirically derived equations based on limited in-reactor tests. These equations were subsequently found to under-predict dimensional changes. New equations have since been developed which are based on periodic measurements taken on the fuel channels in the Pickering GS 'A' and Bruce GS 'A' reactors. These later equations have been substantiated by many such measurements taken in subsequent years. They are accurate within + 25% which includes scatter in the measurements and variations in metallurgical properties between tubes. A more creep-resistant zirconium alloy is now being developed which will have significantly less creep.

Creep allowances incorporated in the Pickering GS 'A' and Bruce GS 'A' designs were based on the earlier empirical equations. Consequently, fuel channels in these reactors cannot accommodate currently predicted creep changes for the 30 year design life of the station. The new equations became available late during the construction of the Bruce GS 'A' station and it was possible to make modifications to increase the creep life in Unit 4.

3.2.2 Effects of Creep

Pressure tube creep affects the fuel channels in different ways. For example, diametral creep increases the coolant flow area in the tube and axial creep creates channels of unequal length. All regions of the fuel channels which are affected by creep were analysed and where necessary tested to establish that no intolerable operating situations were created. The analyses and tests were conducted for the Pickering GS 'A' and Bruce GS 'A' reactors since these are the only reactors adversely affected by creep. The analyses and tests confirmed that the fuel channels are very tolerant of creep and in only two regions are remedial actions needed to keep the reactors operating for their full design life. These regions are the journal/bearings which are adversely affected by axial creep elongation and the spacings between feeder pipes which are adversely affected by differences in elongation.

The effect of creep on these regions and the remedial actions planned to maintain reactor operation are discussed below.

(a) Journal/Bearings

In Pickering GS 'A' and Bruce GS 'A', the fuel channel bearings are only long enough to accommodate about 15 years of creep. However, the fuel channels can be axially repositioned to restore bearing travel. This is accomplished by disconnecting the fuel channel from the reactor structure, pulling it back towards the fixed end and reconnecting it in this new position. This will reposition the bearings towards their originally installed positions and permit continued operation.

The fuel channel and reactor geometries will permit the bearings to be repositioned three times, which will give an additional creep life of about 15 years. This will extend the fuel channel operating life of both stations to 30 years.

(b) Feeder Pipe Spacings

Differential creep gradually reduces some feeder pipe installed spacings and if permitted to continue, will eventually lead to contact between adjacent feeder pipes in the Pickering GS 'A' and Bruce GS 'A' reactors.

Most cases of feeder pipe contact can be avoided by optimizing the fuel channel re-positioning scheme mentioned for restoring journal/bearing travel. Repositioning schemes studied for the Pickering GS 'A' reactors indicates that in all but a very few cases, feeder pipe contact can be avoided for the 30 year reactor design life. Spacings in doubtful cases can be restored by selectively defuelling the faster creeping channels, for short periods of time, to stop them from creeping. The loss in reactor power associated with limited defuellings is estimated to be less than 1% for all cases of potential feeder contacts.

3.2.3 Solutions

Once it was realized that the Pickering GS 'A' and Bruce GS 'A' creep provisions were inadequate, the fuel channel design was modified. Later stations now incorporate greater allowances and can accommodate currently predicted creep for 30 years. Designing for creep is a very easy task and to ensure adequate allowances are provided it is necessary only that the designers have accurate creep predictions with which to work. Current creep equations now predict creep accurately since they are based on many measurements taken on fuel channels in operating reactors. They are sufficiently accurate that we can now closely predict creep in any fuel channel before the measurements are taken.

To correct the journal/bearing travel problem it is necessary only to provide longer bearings. Similarly, to prevent feeder pipes from contacting it is necessary only to increase their initial spacings. These increases were minimized by arranging to accommodate half the total creep at each end of the reactor, whereas in the Pickering GS 'A' and Bruce GS 'A' stations it was intended that axial creep be accommodated all at one end. The positioning assemblies permit the creep to be accommodated equally at each end of the reactor.

As a result of these and other minor modifications the fuel channel design can accommodate currently predicted creep for the full 30 year design life of the station. Further modifications are practical which could extend the creep life further and it is expected that the next generation of CANDU reactors will have additional creep capacity, which will permit a longer design life.

4.0 SUMMARY

The CANDU reactor fuel channels have made a significant contribution to the very high capacity factors attained in the Pickering GS 'A' and Bruce GS 'A' reactors. This has been achieved despite delayed hydride cracking of the pressure tubes and replacement of seventy fuel channels. The delayed hydride cracking was caused by very high residual stresses introduced by the rolling operation. Newer reactors incorporate a modified rolled joint which has much lower residual stresses and will not be susceptible to delayed hydride cracking.

Provisions for accommodating pressure tube creep in the Pickering GS 'A' and Bruce GS 'A' reactors are insufficient to last the 30 year design life. New creep equations have been derived and their accuracy has been validated by frequent in-reactor measurements. Critical creep allowances in the Pickering GS 'A' and Bruce GS 'A' reactors will be restored by axially repositioning the fuel channels. Creep allowances have been increased in newer reactors and are adequate for the design life without the need for axial repositioning.

Because the fuel channels were designed to be replaceable, the fuel channels replacements in Pickering GS 'A' and Bruce GS 'A' were relatively simple operations. These replacements have confirmed that it is practical to completely retube a CANDU reator. This means that the CANDU reactor design life can be significantly extended and perhaps even doubled.

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TABLE 1

CANDU PHWR RANKING WORLD'S COMMERCIAL REACTORS
500 MWe AND LARGER TO 1981, DEC. 31.

UNIT	GROSS CAPACITY FACTOR (%)*	WORLD RANK
BRUCE 3	84	1
PICKERING 2	82	3
BRUCE 4	81	4
PICKERING 1	80	5
PICKERING 4	80	6
BRUCE 1	79	8
PICKERING 3	78	10
BRUCE 2	73	18

$$\text{*GROSS CAPACITY FACTOR \%} = \frac{\text{ENERGY PRODUCED IN PERIOD}}{\text{PERFECT PRODUCTION IN PERIOD}} \times 100$$

TABLE 2

LIFETIME INCAPABILITY
TO 1981, DEC. 31

8 UNITS — 53.1 UNIT YEARS

CAUSE OF INCAPABILITY	PICKERING GS 'A' INCAPABILITY (%)	BRUCE GS 'A' INCAPABILITY (%)
ON–POWER FUELING	0.8	0.8
FUEL	0.1	0.0
HEAT TRANSPORT PUMPS	0.2	0.2
PRESSURE TUBES	4.9	0.3
BOILERS	0.5	2.4
TURBINE AND GENERATORS	5.8	6.6
INSTRUMENTATION AND CONTROL	0.7	1.7
HEAT EXCHANGERS	0.9	0.0
VALVES	0.4	4.5
OTHER	5.6	0.0
TOTAL INCAPABILITY	19.9	16.5

$$\text{INCAPABILITY FACTOR \%} = \frac{\text{ENERGY NOT PRODUCED DUE TO EQUIPMENT INCAPABILITY IN PERIOD}}{\text{PERFECT PRODUCTION IN PERIOD}} \times 100$$

TABLE 3

CANDU FUEL CHANNELS
ANNUAL INCAPABILITY FACTORS

YEAR	PICKERING NGS–A				BRUCE NGS–A			
	UNIT				UNIT			
	1	2	3	4	1	2	3	4
1971	2.6	0.0						
1972	0.05	0.0	0.0					
1973	0.0	0.0	0.84	0.0				
1974	2.7	0.0	36.7	1.2				
1975	0.0	0.0	23.2	53.3				
1976	4.5	4.0	4.3	25.8				
1977	4.1	1.8	0.0	3.5	0.0	0.0		
1978	0.0	0.0	0.51	0.0	0.0	0.0	0.0	
1979	0.0	2.4	0.8	1.3	0.0	0.0	0.0	0.0
1980	0.003	0.0	2.8	2.9	2.7	0.0	0.56	1.3
1981	0.0	0.0	1.6	0.0	0.0	0.0	0.0	0.0
WEIGHTED AVERAGE	1.3	0.8	7.4	10.9	0.6	0.0	0.14	0.44

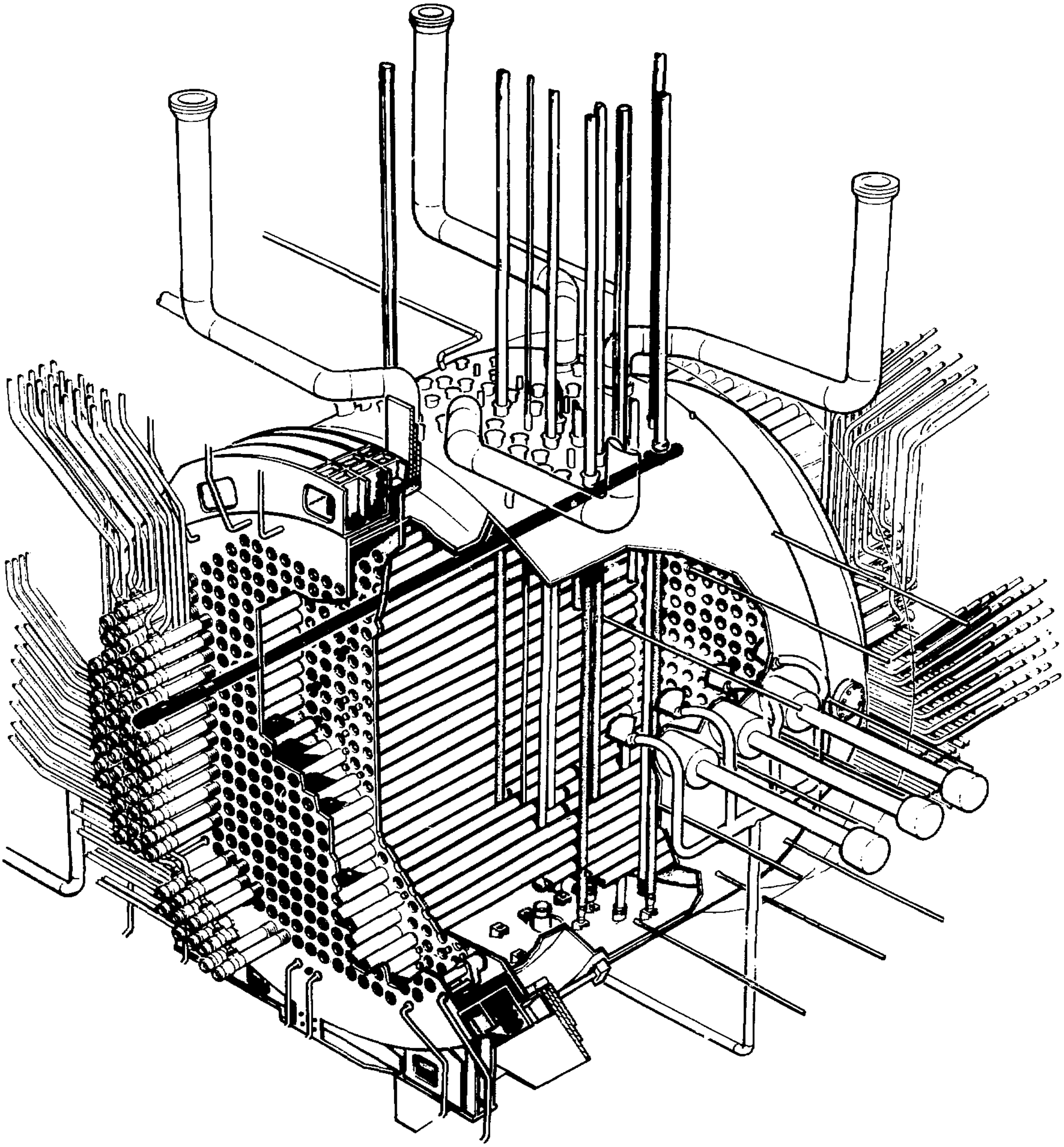

FIGURE 1 CANDU 600 MW REACTOR (ONE FUEL CHANNEL EMPHASIZED)

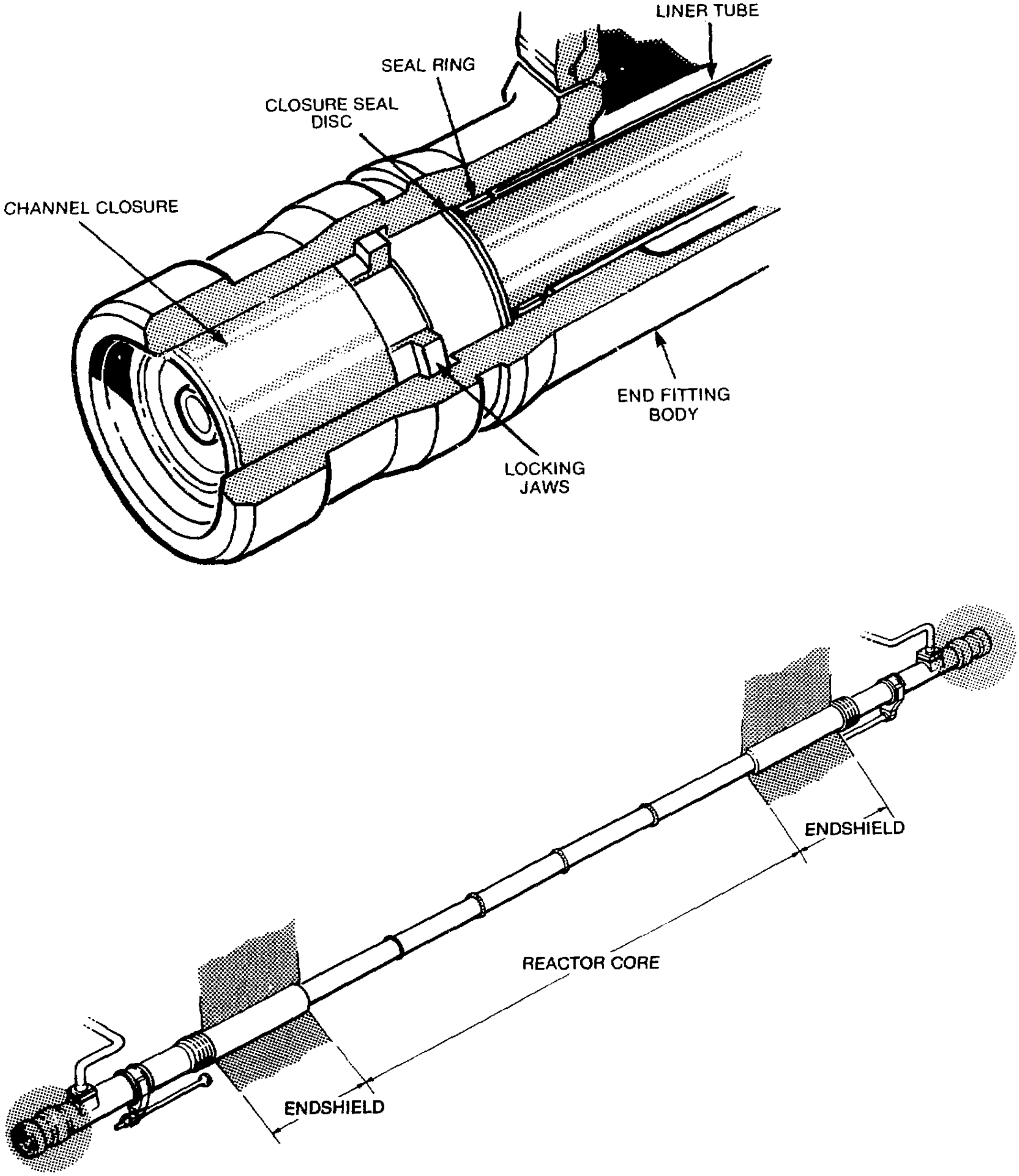

FIGURE 2 CHANNEL CLOSURE

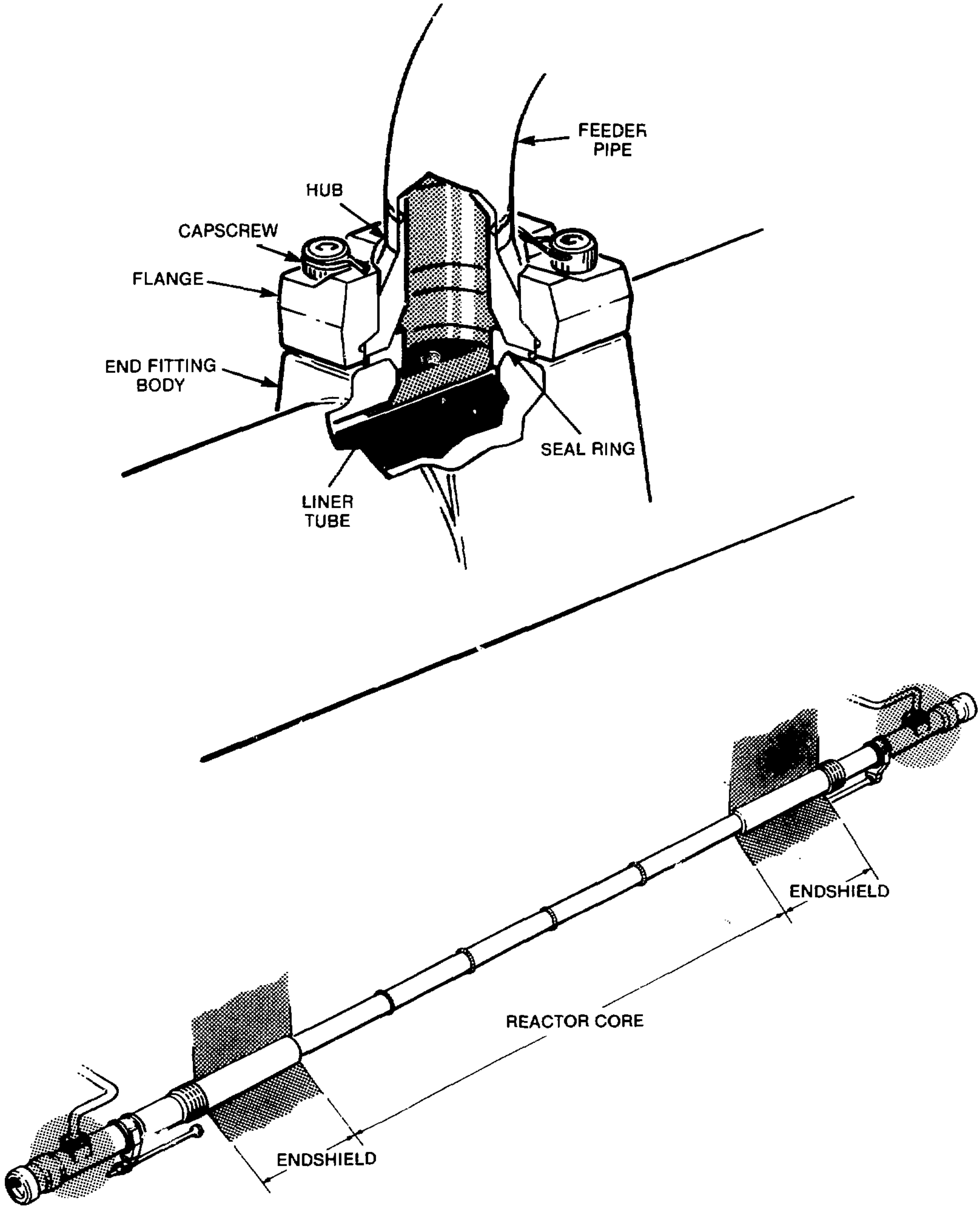

FIGURE 3 FEEDER COUPLING

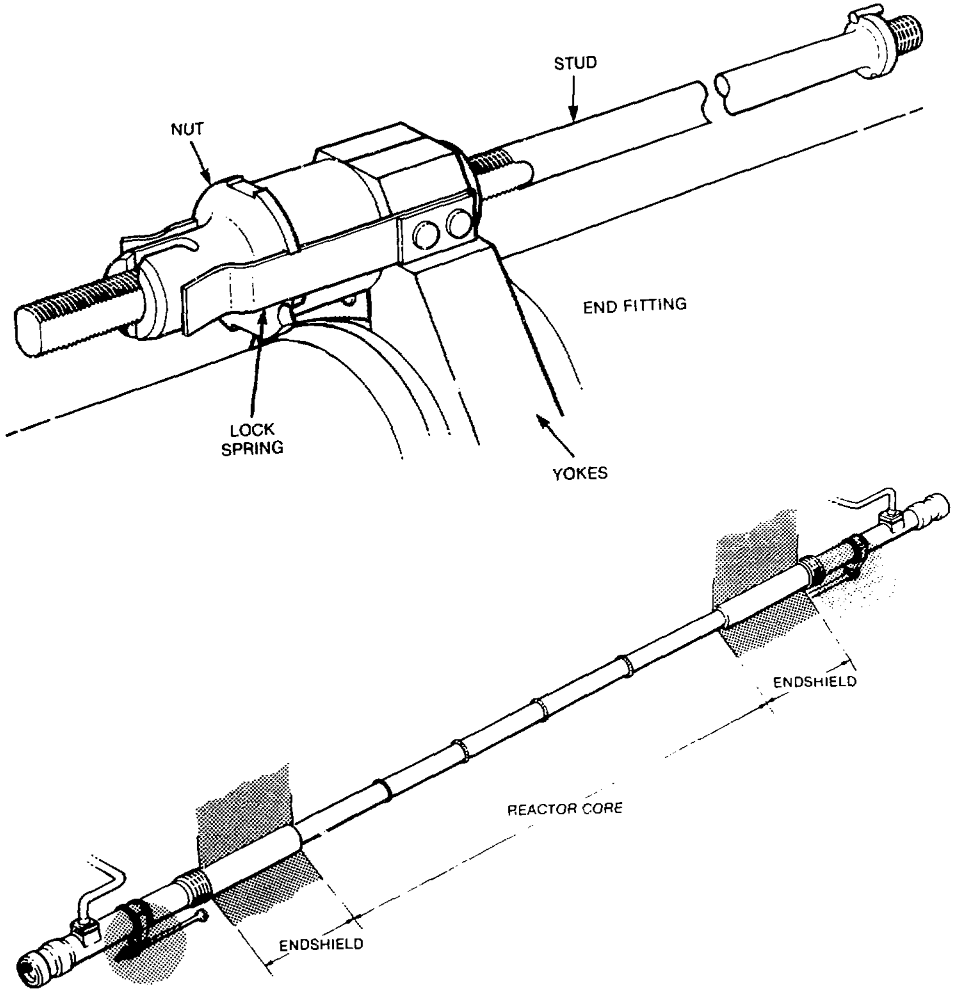

FIGURE 4 POSITIONING ASSEMBLY

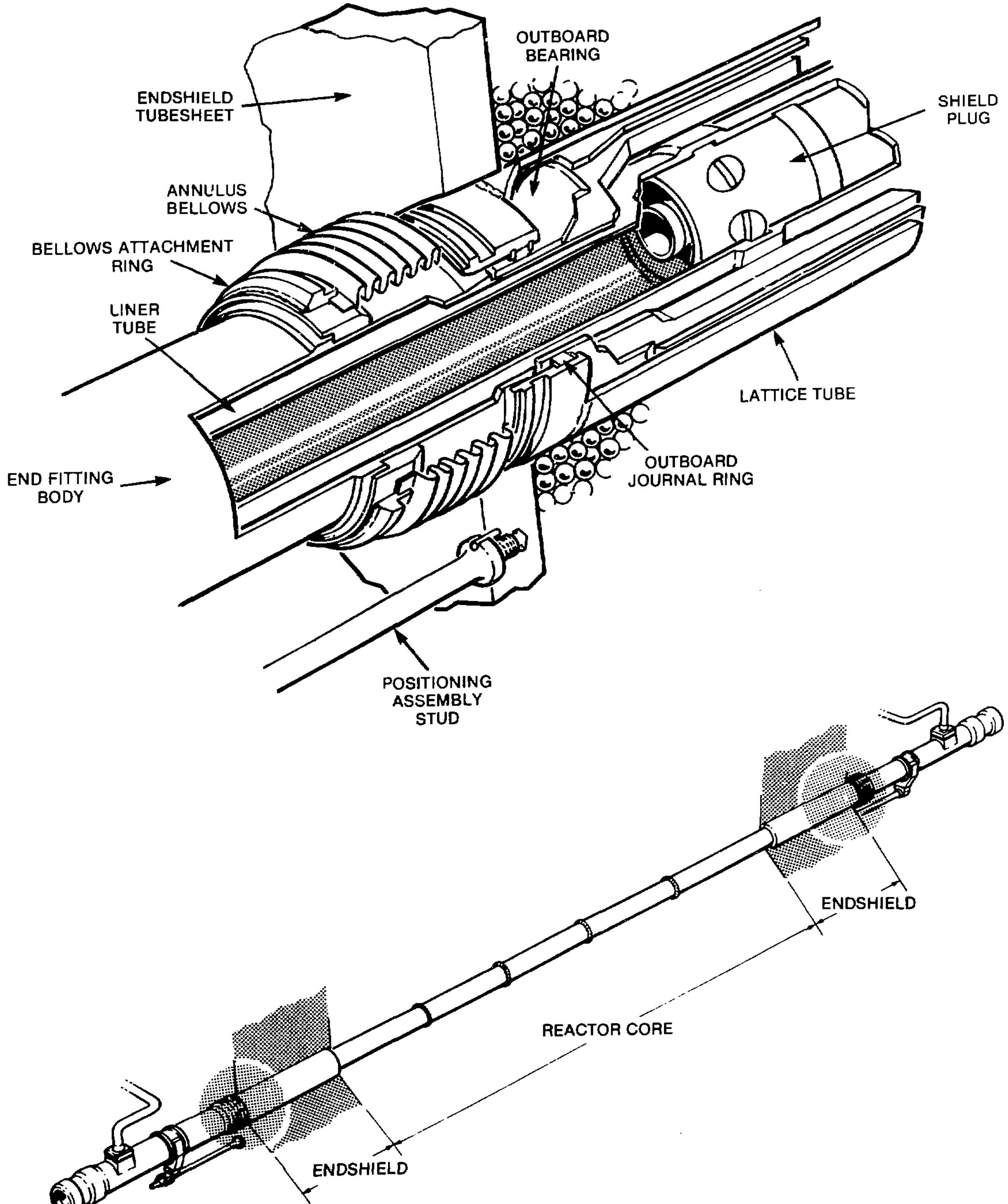

FIGURE 5 BELLOWS ASSEMBLY

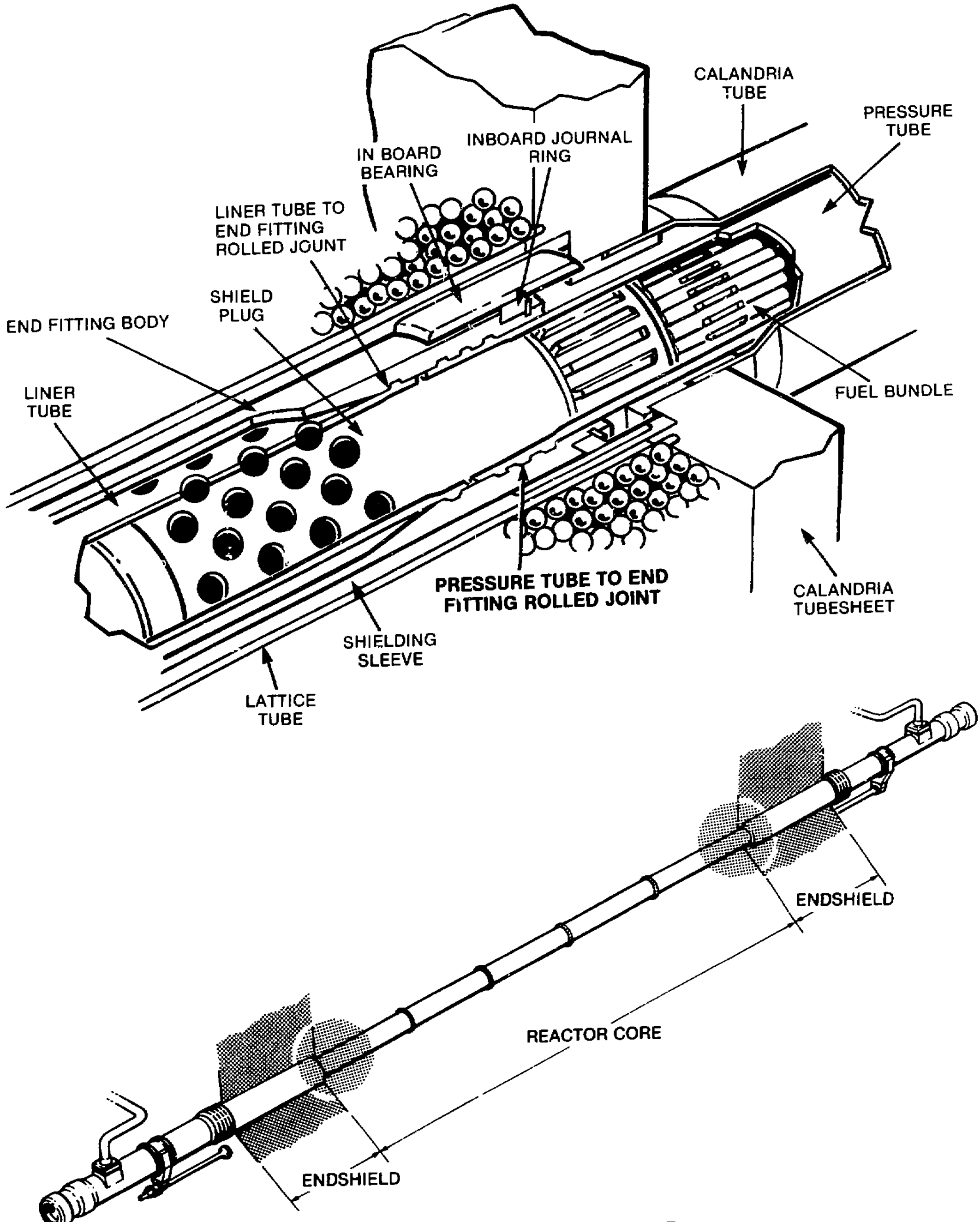

FIGURE 6 ROLLED JOINT

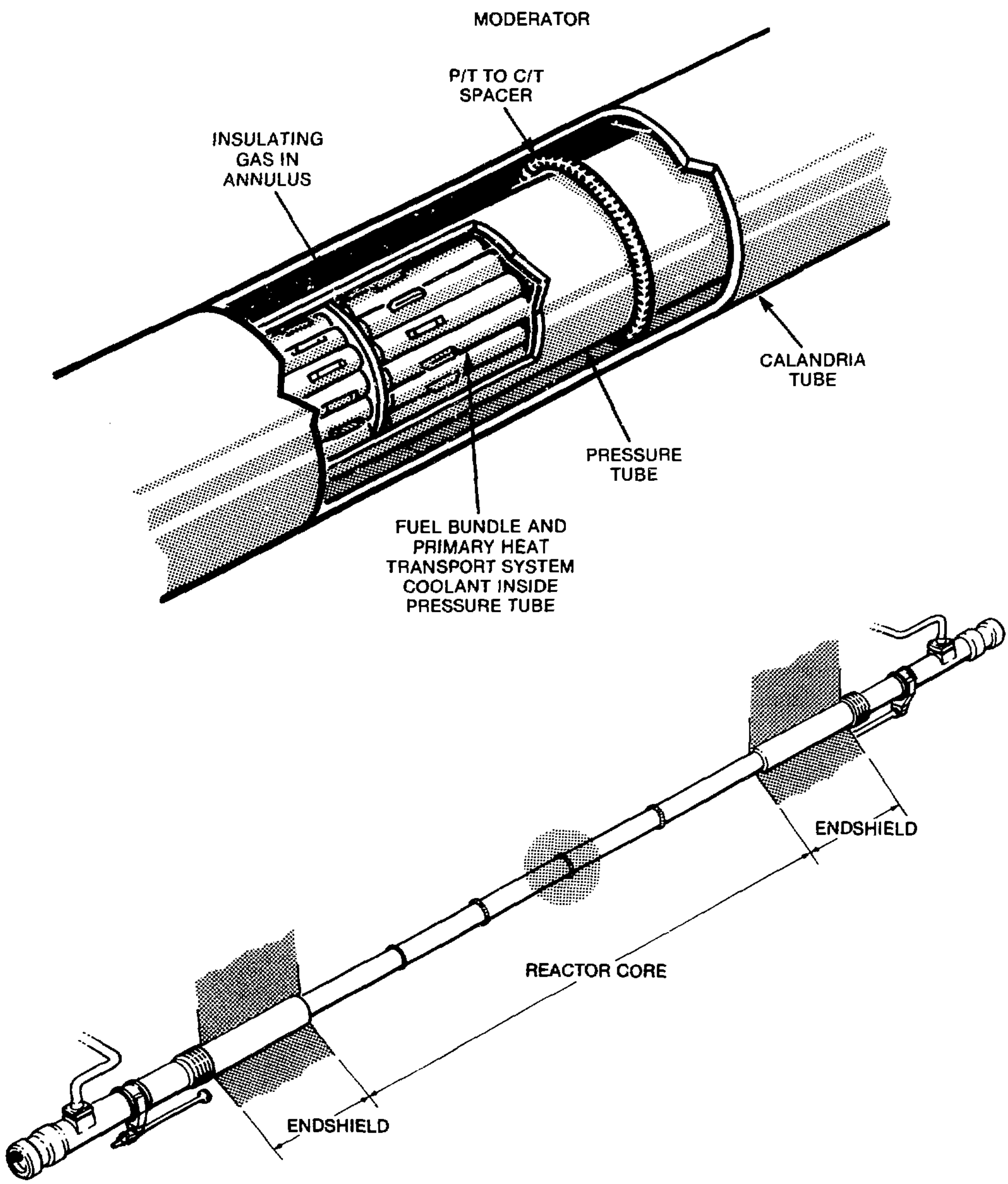

FIGURE 7 SPACER AND FUEL

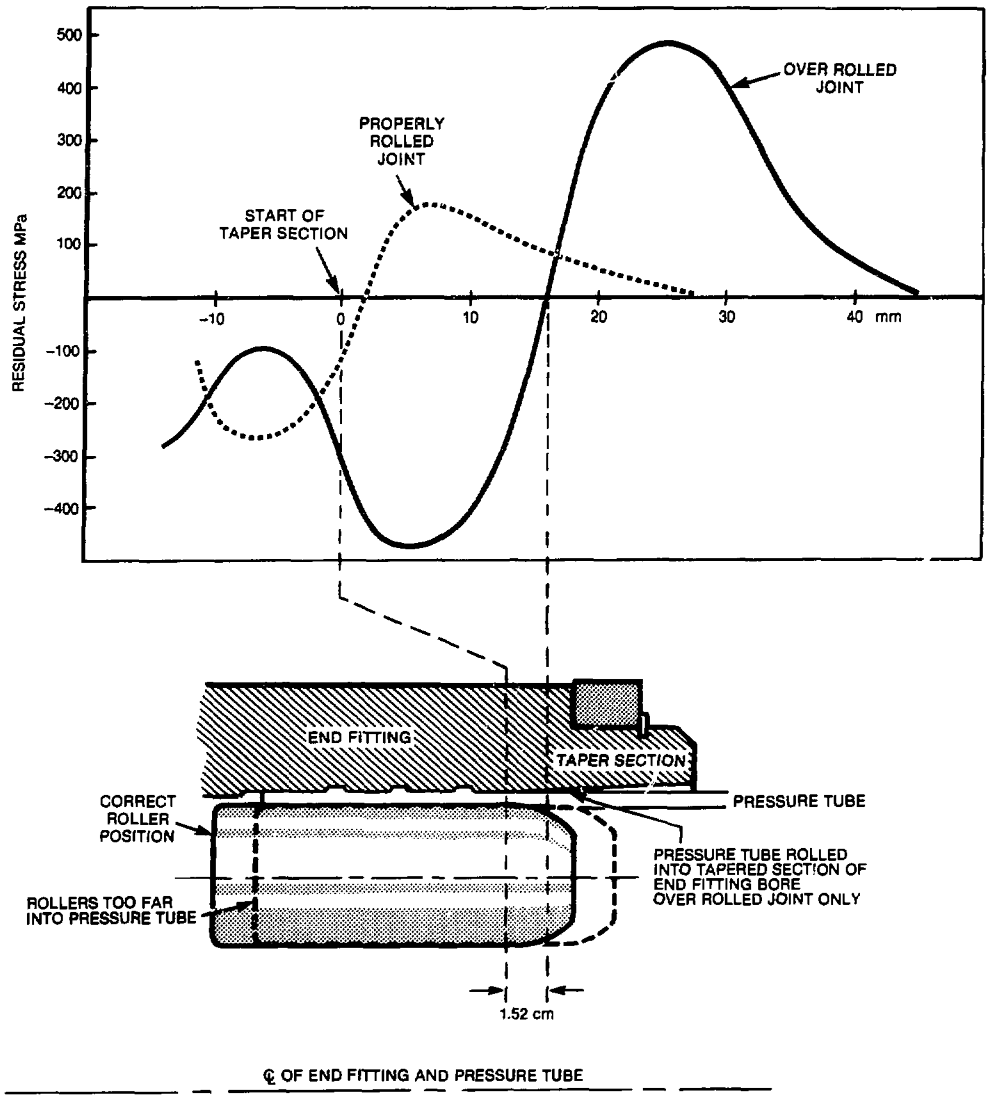

FIGURE 8 RESIDUAL HOOP STRESS IN PRESSURE TUBE

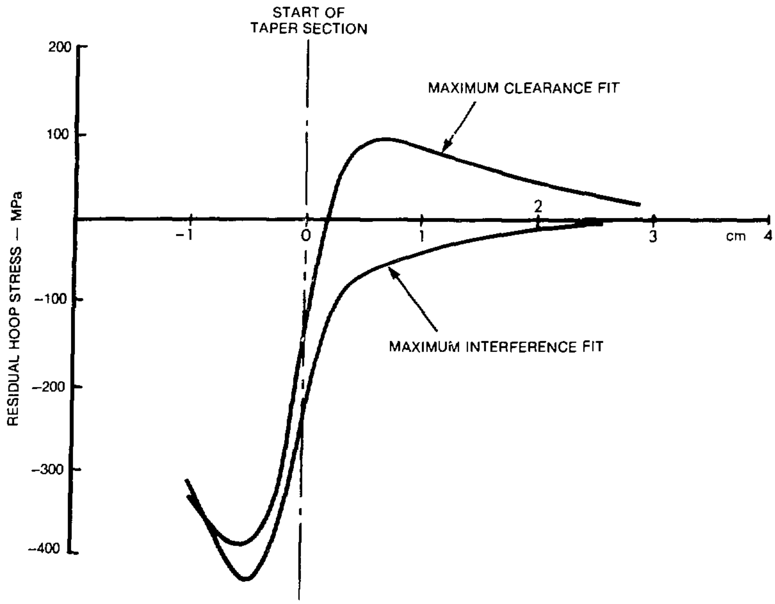

FIGURE 9 ZERO CLEARANCE ROLLED JOINT PRESSURE TUBE RESIDUAL HOOP STRESS

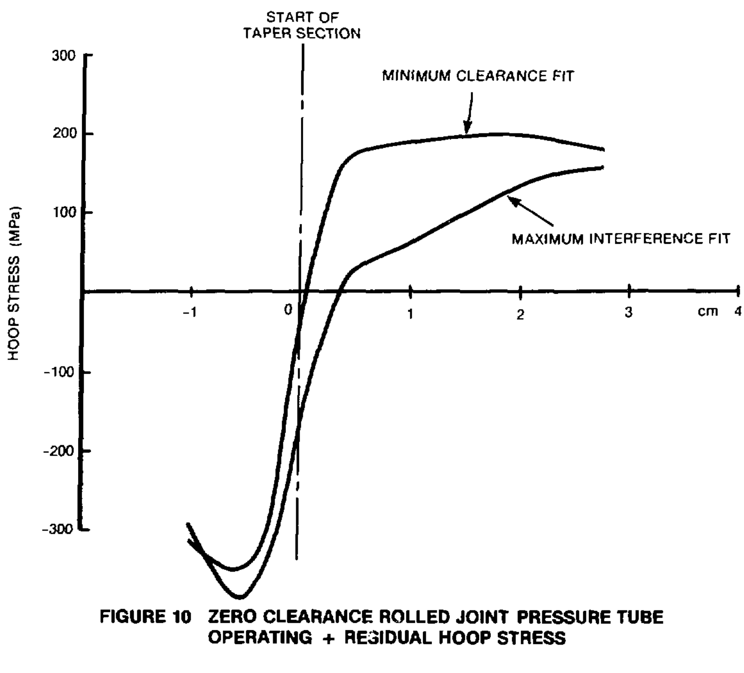

FIGURE 10 ZERO CLEARANCE ROLLED JOINT PRESSURE TUBE OPERATING + RESIDUAL HOOP STRESS

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