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CANDU-PHW Fuel Channel Replacement Experience

Expérience en remplacement des canaux de combustible du réacteur CANDU-PHW

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*par J.T. Dunn et B.K. Kararia**

Résumé

Une des caractéristiques principales du réacteur CANDU à eau lourde pressurisée est l'emploi de tubes de force au lieu d'une grande cuve sous pression contenant le combustible et le caloporteur. Ceci permet une conception inhérente telle qu'il est possible de remplacer ces tubes promptement sans influencer sérieusement les hauts facteurs de disponibilité des réacteurs. Des huit réacteurs industriels d'Ontario Hydro, il y en a sept (y compris deux dont certains des canaux de combustible ont été remplacés) qui occupent les dix premières places parmi les grands réacteurs électronucléaires du monde, du point de vue de la performance.

Des fissures et fuites consécutives dans la région du joint dudgeonné des tubes de force ont entraîné le remplacement de 70 canaux de combustible dans trois réacteurs, les tout derniers étant ceux du réacteur 2 de la centrale nucléaire de Bruce 'A' en février 1982. On a modifié la conception du joint dudgeonné ainsi que les techniques de dudgeonnage pour ne plus avoir ce problème avec les réacteurs CANDU qui suivront Bruce 'A'.

La présente communication est une description de la performance passée et prévue des tubes de force des réacteurs CANDU, de l'outillage et des techniques employés pour le remplacement des canaux de combustible.

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Abstract

One of the main characteristics of the CANDU pressurized heavy water reactor is the use of pressure tubes rather than one large pressure vessel to contain the fuel and coolant. This provides an inherent design capability to permit their replacement in an expeditious manner, without seriously affecting the high capacity factors of the reactor units. Of the eight Ontario Hydro commercial nuclear generating units, the lifetime performance places seven of them (including two that have had some of their fuel channels replaced), in the top ten positions in the world's large nuclear-electric unit performance ranking.

Pressure tube cracks in the rolled joint region has resulted in 70 fuel channels being replaced in three reactor units, the latest being at the Bruce Nuclear Generating Station 'A', Unit 2 in February 1982. The rolled joint design and rolling procedures have been modified to eliminate this problem on CANDU units subsequent to Bruce 'A'.

This paper describes the CANDU pressure tube performance history and expectations, the tooling and procedures used to carry out the fuel channel replacement.

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1.0 INTRODUCTION

The CANDU-PHW is a unique concept in that it employs multiple fuel channel pressure tubes, rather than one large pressure vessel, to contain the fuel and coolant.

The fuel channels are designed to permit their replacement with relative ease. Three of the Ontario Hydro (O.H.) commercial size CANDU units have experienced cracks in the rolled joint region of the fuel channel pressure tube, necessitating fuel channel replacement. This led to the introduction of improvements in the design and the installation procedure in subsequent units.

To date some 70 fuel channels have been replaced. The recent experience with a cracked pressure tube in Bruce NGS-A, Unit 2, demonstrates that fuel channel replacement can be accomplished with minimal down time and low personnel radiation exposure.

The production performance of eight Ontario Hydro commercial CANDU units has been exceptional. Table 1 shows this performance to the end of 1981 and the ranking of these units in the world's 131 large (greater than 500 MWe) operating reactors.

This paper describes the tooling and procedures used to replace the fuel channel in Bruce NGS-A, Unit 2, in February 1982, which required a unit outage of only 36 days and resulted in a radiation exposure of 10 rem.

2.0 REACTOR AND FUEL CHANNEL DESIGN DESCRIPTION

The CANDU reactor consists of a calandria, which is a large cylindrical tank (typically 8.5 m diameter by 5 m long) containing the heavy water moderator, end shields, and identical fuel channels in a square lattice array, which horizontally traverse the calandria and project through the end shields (Figure 1).

A fuel channel assembly consists of a pressure tube, two end fittings and associated hardware as shown in Figure 2.

The pressure tubes are about 6 m long and 103 mm in inside diameter, with a wall thickness of 4.06 mm and made of cold-worked Zirconium-2.5% wt Niobium alloy. The pressure tubes are subject to an internal heavy water coolant pressure and to bending loads from the weight of the fuel and coolant. The pressure tubes are located inside the calandria tubes and separated by garter springs. The pressure tube is connected to an end fitting at each end by means of rolled joints.

Figure 3 shows the cross-section of a rolled joint. Three grooves are machined in the end fitting bore, and the pressure tube is inserted into the end fitting covering the grooves. A tube expander is introduced into the pressure tube through the end fitting and the tube is roll-expanded into the end fitting. The pressure tube wall thickness is reduced by 12-15% and the grooves in the end fitting are partially filled with the pressure tube material. The axial strength and leak-tightness of the rolled joints were established by extreme qualification testing, and have proved very satisfactory in operation.

The thin wall design of the pressure tube ensures that any cracks will result in a leak path through the wall before occurance of an unstable fracture. This type of behaviour is called "leak-before-break". Extensive laboratory tests have shown that the critical crack length at operating conditions is much longer than the length of the longest possible non-leaking crack. The three reactors that have had pressure tube cracks to date have provided in-service confirmation of the leak-before-break concept.

The CANDU-PHW concept permits the replacement of the fuel channels with relative ease.

The annular gap between the pressure and calandria tubes forms a part of the gas annulus system. The gas annulus system provides a dry pressurized inert gas atmosphere in the channel annuli between the pressure tubes and calandria tubes. The gas acts as a thermal barrier and prevents corrosion of the fuel channel components.

Gas annulus system moisture monitoring provides an effective and reliable means of detecting the leakage of heavy water from the fuel channel.

3.0 ONTARIO HYDRO COMMERCIAL CANDU-PHW — FUEL CHANNEL EXPERIENCE

To quantify the effect of equipment performance O.H. measures this in terms of the Incapability caused by specific equipment as a percentage of perfect production in the time period.

Table 2 and 3 identify the equipment causing Lifetime Incapabilities of the Pickering and Bruce NGS.

Table 4 shows each Unit Incapability due to fuel channels during each year of service. The lifetime incapability due to fuel channels has been 4.9% at Pickering NGS-A and 0.3% at Bruce NGS-A, to the end of 1981.

3.1 Pickering NGS-A

Fuel channel pressure tube rolled joint cracking has occurred in Units 3 and 4. Over-extended rolling of the rolled joints during installation resulted in high residual stresses, causing the cracking.

3.1.1 Unit 3

In August 1974, when Unit 3 was being returned to service after a scheduled maintenance outage, routine checks of the gas annulus system showed that heavy water was leaking from the primary heat transport system. As the gas annulus system is connected to each fuel channel via a series of headers and small tubes called pig-tails (Figure 4), it was difficult and time-consuming to establish the exact location of the leaking fuel channel. One leaking channel was eventually identified and replaced, with a personnel radiation exposure of 78 rem.

Alternative methods of leak detection were investigated and the one that proved best was acoustic emission where the noise emitted by the leaking water could be detected, Figure 5. This method was used when further leakage in Unit 3 was confirmed and sixteen leaking channels were identified.

Ultrasonic flaw detection equipment, that could be installed by the fuelling machines, was developed to eliminate the need to defuel and isolate a fuel channel for examination. The rotation of the ultrasonic transducers was manually performed with signal interpretation performed remotely. Seventy pressure tube rolled joints were fully examined, no cracks were found.

The reactor was returned to service in March 1975, some seven months after the initial detection of a leak. Seventeen fuel channels were replaced with a total radiation exposure to the maintenance staff of 420 rem.

The basic procedures and tooling used to replace the fuel channels were similar to those used recently at Bruce NGS-A, Unit 2.

3.1.2 Unit 4

In May 1975, heavy water was found in the gas annulus system of Unit 4. Acoustic emission testing showed that there were two leaking fuel channels. By this time a motorized remotely controlled ultrasonic flaw detection method was available and nearly 400 pressure tube rolled joints were examined.

Cracks were found in 61 joints (57 fuel channels) and varied in depth between 10% and 90% of the wall thickness. Two leaking fuel channels and 50 of those showing partial through-wall cracks were replaced.

Seven fuel channels each with a small crack in the rolled joints were not replaced. The cracks were less than 2 mm in length and did not constitute a threat to the integrity of the pressure tube. (Subsequent examinations have detected no increase in length of these cracks.)

The reactor was returned to service in March 1976, ten months after detection of the initial leak. Fifty-two fuel channels were replaced with a total radiation exposure to maintenance staff of 953 rem.

3.2 Bruce NGS-A

The pressure tubes in Units 1 and 2 were installed prior to the time of the pressure tube leaks in Pickering Units 3 and 4. The fuel channels had been installed and installation practice had allowed the over-extension of the rolled joint as in the Pickering Units.

The pressure tubes in Units 3 and 4 were rolled correctly.

A high temperature stress relief of the zone of high residual stress was carried out on all units prior to reactor start-up.

The stress relief was performed with tooling capable of heating the defined high tensile stress zone of the pressure tube rolled joint to a temperature of 500°C in an inert gas atmosphere.

Confirmation of the adequacy of the stress relief operations was obtained by removing a channel from Unit 2 after stress relief (Figure 6).

3.2.1 Unit 2

In February 1982, when Unit 2 was being returned to power after a hot shutdown, routine checks of the gas annulus system revealed the presence of primary coolant. Inspection of the flow gauges in the reactor vault permitted identification of the reactor quadrant where the leakage was occurring. This was followed by acoustic emission testing of all 480 channels. These inspections identified one leaking fuel channel.

The leaking channel was defueled, drained and the pressure tube examined using eddy current methods; it revealed the presence of a short axial crack at one end. Ten additional rolled joint regions were inspected using ultrasonic techniques. These comprised four joints that had been stress-relieved with equipment calibrated at the same time as the cracked joint, and six that had not been stress-relieved and would be expected to have the highest residual stresses. No indications of cracks were found.

The reactor was returned to service on March 16, 1982, 36 days after the initial detection of a leak. One fuel channel was replaced with a total radiation exposure to maintenance staff of 10 rem.

4.0 FAILURE CAUSE

Detailed examinations of the cracked rolled joints removed from Pickering Unit 3 showed that the joints had been incorrectly rolled resulting in extremely high residual stresses, and that the failure mechanism was delayed hydride cracking which resumed and progressed during each reactor cooldown.

The improper rolling was caused by over insertion of rollers, by approximately 13 mm, expanding the pressure tube beyond the end of the supporting surface on the end fitting. Residual stresses up to 470 MPa were measured. Figure 7 shows the residual stress distribution in a "good" and an over-extended rolled joint.

The hydrogen normally found in new pressure tube material (about 10 ppm) is increased by the absorption of deuterium from the coolant during the operation of the reactor. At reactor operating temperatures this hydrogen/deuterium is in solution. During cooldown hydride platlets are formed, and due to the high residual hoop stresses that existed, these platlets formed in a radial axial plane. Fracture of these platlets during reactor shutdowns resulted in the through-wall cracks.

The lengths of leaking cracks were between 18 mm and 21 mm. Laboratory testing shows that the mean critical crack length at reactor operating conditions is 80 mm, with a 95% confidence lower limit of 56 mm.

Extensive tests using the tubes removed from Pickering Units 3 and 4 has shown that the cracks were stable at operating pressure and could in fact withstand more than five times the operating pressure.

The examination of the leaking Bruce Unit 2 pressure tube has led to the conclusion that the crack is similar to that seen at Pickering (Figure 8), and initiated during the two years between rolling and stress relief. At the time of stress relief this crack was about 0.3 mm deep permitting it to propagate during reactor shutdowns by delayed hydride cracking during the reactor shutdowns.

5.0 CORRECTIVE DESIGN CHANGES

As a direct result of the Pickering NGS-A experience, extensive development work was carried out to provide more information on the relationship between rolling variables and residual stresses in the rolled joints.

It was subsequently shown that with the diameteral clearances permitted between components used in the rolled joint, proper rolling produced high residual hoop stresses. This was compounded by incorrect location of the rollers during the rolling, which made the residual stresses even higher.

In all CANDU reactors built since Bruce NGS-A, the rolled joint design has been changed to eliminate the initial clearances, and rolling procedures modified to ensure no "over-extended" rolling can occur. This is called a zero clearance rolled joint, for which the typical residual stresses are shown in Figure 9.

6.0 BRUCE NGS FUEL CHANNEL REPLACEMENT — EQUIPMENT AND PROCEDURES

The recent fuel channel removal and replacement at Bruce NGS-A, Unit 2 is described.

6.1 Reactor Vault Layout

Several features of the Bruce NGS-A reactor vault layout and reactor equipment facilitate fuel channel replacement (Figure 10).

The reactor vault is a box-like concrete structure, on the floor of which stands the reactor assembly straddling the fuelling machine tunnel or duct, which runs the length of the 4 Unit station. Fuelling machine bridges at each end of the reactor are supported from columns mounted on the vault floor. These bridges are lowered to floor level to pick-up fuelling machine heads from trolleys in the fuelling machine duct, and then elevated to the fuel channel to be refuelled. These bridges are thus elevators, the tops of which are used to position and support personnel and equipment at an appropriate elevation for reactor-face work, Figure 11.

Access to the reactor vault is through an equipment airlock which will permit the passage of equipment up to 2.5 m high by 2.5 m wide by 4.5 m long. A second equipment airlock, essentially a pipe about 9 m long with a 1 m bore, and with airtight hatches at both ends, is also available for the passage of long, small diameter reactor components and tools.

Two 10-ton cranes, one at each end of the reactor, facilitate movement of equipment such as shielding flasks within the vault and may also be used for the controlled insertion of long tools and components into the reactor.

Periscopes permit observation of each reactor face from outside the vault.

6.2 Preparation

Prior to the start of on-reactor fuel channel replacement operations it was necessary to prepare the vault. Since the fuelling machine bridge is only about 1 m wide, it was necessary to mount working platforms on both bridges. These were relatively light structures, designed to support personnel and light equipment with local reinforcing to transfer the weight of the shielding flasks to the bridge structure. Access facilities between the working platforms and vault floor was relatively straightforward since the fuel channel to be replaced was located only slightly above the floor level. Relatively low general fields (100 mR/hour) did not necessitate the use of personnel shielding cabinets, Figure 12.

NOTE: Where background radiation fields warrant it, shielding cabinets with built-on floor extensions, are available for mounting on the bridges to provide working areas for personnel as well as radiation shielding.

6.3 Tooling and Procedures

When the heavy water was detected in the gas annulus system and identified as primary heat transportation system (PHTS) coolant, it was considered likely that at least one pressure tube had cracked and preparations were started immediately for the provision of tooling and procedures to enable the station to replace the leaking fuel channel.

The design of most tools required for this activity already existed and the redesign work, procurement of new tooling and modification of existing tooling, including some of that previously used for Pickering NGS-A Units 3 and 4, was completed within eighteen days.

Simultaneously, detailed replacement procedures were prepared and transformed into Site Work Plans.

Wherever practical the tools were designed to be self-shielding and temporary shield plugs were built to eliminate radiation beams from the channel, Figure 13. Two-handed tools, to enable personnel to work at the sides of the radiation beam, were used for removal and installation of tooling.

As tooling became available at site it was performance-tested independently and then in conjunction with interfacing tools and components. Accurate mock-ups of the geometry of the reactor faces around the fuel channel to be replaced were built, one to permit the commissioning of the bellows and stop collar weld cutting and rewelding equipment, the second to allow commissioning of the balance of the tooling. Both were utilized, first for the development of procedures and later for personnel training and operation rehearsal.

Two crews were fully trained in all procedures to allow work to proceed continuously when on the reactor.

The final rehearsals were performed using a simulation platform and full personnel radiation protection equipment. Tritium concentrations (30-50 MPC) in the reactor vault necessitated personnel to wear air-ventilated plastic suits while in the vault.

Radiation protection personnel, as well as maintenance personnel, participated in these final rehearsals. Procedures required that they be present on the working platforms during all operations when a high radiation beam existed, and when tools and components were removed from the reactor core.

Two-way audio communication sets were utilized for communication between in-vault personnel, and between in-vault and out-of-vault personnel. A closed circuit television system with cameras in the vault allowed personnel outside the vault to monitor in-vault activities at all times.

6.4 Defective Fuel Channel Removal

On March 6, 1982 the first crew started work on the reactor face. The fuel channel to be replaced had by this time been isolated from the PHTS, by the installation of feeder blanks. Shield and closure plugs (Figure 2) had been removed from both East and West end fittings and replaced by temporary shield plugs (Figure 13).

Following removal of the shield plug at the fuel channel West end, a radiation survey was performed to assess the beam strength and envelope, Figure 12.

6.4.1 West End Fitting Removal

A pressure tube cutting tool was inserted into the end fitting and the cutters positioned 200 mm into the pressure tube and beyond the end of the end fitting, Figure 13. The severing operation was performed using an air motor to drive the rotating tube cutter type wheels through the tube wall (no swarf is produced).

The next operation is to detach the end fitting from the stop collar by machining out the 6.4 mm weld joining the two components. A cutting head was assembled around the end fitting adjoining the stop collar and was driven by an air motor through a drive shaft supported from the end fitting. The cutting head is of a special design as it must pass through very small clearances between the feeder pipes and the end fittings and operate remotely close to the reactor end shields. After removal of the weld metal, the West end fitting was then ready to be withdrawn from the reactor.

A shielding flask was aligned with the end fitting and a support trough about 5 m long was inserted through the flask to extend under the end fitting as far as the stop collar, Figure 14. A cable from a winch mounted on the outboard end of the flask was passed through the flask and secured to the end fitting, which was then pulled along the trough into the flask, by means of the winch and cable. The operator was located behind the flask and was shielded by it.

The trough was removed, the flask doors were closed, and following a satisfactory radiation survey, the flask was removed from the vault.

6.4.2 East End Fitting and Pressure Tube Removal

An extension sleeve extending from the West calandria tube insert to just outboard of the reactor end fitting face, and with a bore slightly larger than that of the calandria tube, was then installed in the West end shield lattice tube, Figure 15. This formed a bridge between the calandria tube and the working platform.

A pressure tube support sleeve, made up from three sections coupled together, was then passed through this extension sleeve and through the annulus between the pressure tube and calandria tube until its leading end reached a distance of about 450 mm from the East end fitting, Figure 15. While passing through the annulus it pushes the two garter springs ahead of it towards the East end fitting. This support sleeve has three main purposes; it moves and positions the garter springs; it protects the calandria tube during movement of the cut pressure tube, and it contains the cut pressure tube from the reactor into the shielding flask.

A pressure tube pusher tool was next inserted into the support sleeve and advanced until it touched the West end of the pressure tube. It was then coupled to the support sleeve to prevent independent motion of either tool.

The pressure tube cutting tool was then inserted through the East end fitting, and two pressure tube cuts made, one in the middle and one about 200 mm from the East end fitting, Figures 16 and 17.

The pressure tube could then be removed in two halves, reducing the overall length of the shielding flask. (This was necessary for it to pass through the equipment airlock.)

Following the cutting of the bellows/end fitting weld, the East end fitting was removed in the same manner as the West end fitting and a similar calandria tube extension sleeve was then inserted and secured in this end of the reactor. A shielding flask was next positioned at the East end of the reactor in line with the extension sleeve ready to receive the first half of the pressure tube, Figure 18.

At the West end, the pusher tool, hence the pressure tube and garter springs, were advanced towards the East end until the leading end of the support sleeve just entered the flask. At this point the support sleeve was uncoupled from the pusher tool and restrained axially, the pusher tool was then advanced until the first half of the pressure tube and the garter springs had entered the flask, Figure 18. The flask was then lowered about 110 mm by means of the bridge controls until the support sleeve was correctly aligned with the upper half of the flask bore. The pusher tool was further advanced until the second half of the pressure tube was fully in the flask, Figure 19. The pusher tool and support sleeve were then withdrawn and removed from the reactor. The flask doors were then closed and following a satisfactory radiation survey, the flask removed from the vault.

NOTE: Since the pressure tube removal was deemed to be the most hazardous phase of all the operations, the East side was chosen for the flasking operation, as the equipment airlock is located on the West side of the vault. Thus, any problem in the pressure tube removal operation, would not impede evacuation of personnel from the vault, Figure 10. No personnel are required to be in the East vault during this operation.

The calandria tube was then cleaned by pulling swabs through it, the bore visually examined using a borescope and its condition was found to be excellent, Figure 20. The calandria tube extension sleeves were then removed from both ends of the reactor.

Removal of the fuel channel was completed in 2 1/2 days.

6.5 Replacement Fuel Channel Installation

While the removal operation was in progress, the replacement fuel channel components were prepared. The replacement pressure tube and East end fitting were prepared as a sub-assembly with the pressure tube rolled into the bore of the end fitting. The pressure tube was cut to the correct length, it being necessary to add a 13 mm allowance to cater for the axial creep of the pressure tube during its first five years of operation. The 8.5 m long assembly was then boxed and transported through the second equipment airlock into the vault.

6.5.1 Pressure Tube/End Fitting Sub-Assembly Installation

The sub-assembly was removed from its box using a special lifting beam necessary to limit bending stresses in the rolled joint during handling. A nose cone with rollers was inserted in the leading end of the pressure tube and the sub-assembly was advanced into the East end of the reactor using the vault crane, Figure 21. The leading end of the pressure tube was supported by the nose cone as it passed through the end shield and calandria tube to the West side of the reactor. After advancing it some 4 m, a counterbalance tool was assembled to the outboard end of the end fitting and the lift was transferred to this tooling using an A-frame hoist, while the lifting beam was removed, Figure 22. The sub-assembly was then advanced until the East end fitting was resting on its bearings, at which point the counterbalance tool and nose cone were removed. The end fitting was then correctly aligned and oriented before clamping.

6.5.2 Garter Spring Spacer Installation

Installation of the garter springs, was performed from the reactor West end, Figure 23. A pressure tube support tool, of the same outside diameter as the pressure tube was spigoted into the pressure tube bore and was used to center the West end of the pressure tube in the calandria tube. A garter spring was installed on the outboard end of the support tool and advanced by a pusher tool along the support tool and pressure tube, to its correct position in the reactor. This was repeated for a second garter spring and the support and garter spring pusher tool were removed.

6.5.3 West End Fitting Installation

After garter spring installation, the West end fitting was lifted using a counterbalance tool, Figure 24. A nose cone projecting from the inboard end of the end fitting, picked up the pressure tube end, and centered it in the end fitting bore during the last part of insertion. The end fitting was then aligned with adjoining end fittings, and the position of the pressure tube end in the end fitting checked. This verified that the pressure tube length was satisfactory, before performing the bellows/end fitting weld at the East end of the reactor, Figure 25.

The West end of the pressure tube was then roll-expanded into the end fitting bore, and a vacuum test was performed to check the leak-tightness of the joint, Figure 26. The stop collar/end fitting weld was then completed satisfactorily.

The duration of the installation phase was just under two days.

6.6 Radiation Exposure

The total elapsed time for the on-reactor fuel channel replacement operations was less than five days with a total whole-body radiation exposure of just over 10 rem. The total extremity radiation exposure for the job was just under 11 rem. The highest individual whole-body radiation exposure was 890 millirem and the highest individual extremity exposure was 975 millirem. The welding and cutting of the bellows and stop collar, when personnel were required to spend several hours at the reactor face, were the largest radiation exposure operations.

7.0 SUMMARY AND CONCLUSIONS

The cracking of the fuel channel pressure tubes in Pickering NGS-A Units 3 and 4 and Bruce NGS-A Unit 2 was due to high residual stresses induced into the pressure tube near the end fitting by improper joint rolling during installation. Detailed metallurgical and joint rolling investigations have shown that, by reducing joint clearances and adopting improved rolling procedures, sufficiently low residual stresses can be achieved to avoid delayed hydrogen cracking.

The three reactors that have experienced leaking fuel channel pressure tube cracks to date, have confirmed the leak-before-break criterion, verifying the high degree of confidence in the safety of zirconium alloy pressure tubes.

The recent Bruce NGS-A experience has, once again, satisfactorily demonstrated the unique CANDU-PHW capability of fuel channel replaceability. The bridge and columns for the on-power fuelling machines, provide excellent access to the reactor faces for repair and maintenance work.

CANDU-PHW fuel channel replacement is accomplished with simple manual tooling made from standard product forms, that can be used by plant maintenance staff. For a quick and successful replacement operation, emphasis is placed on thorough training of work crews on mock-ups using actual tools and detailed procedures.

Experience acquired from Pickering NGS-A fuel channel replacement work, was incorporated into the improved equipment tooling, and procedures used at Bruce NGS-A. This is a planned and continuing process, whereby the recent Bruce experience is being fed back to the designers, operators, fabricators and developers to maintain the state-of-the-art in its most current form.

The CANDU fuel channel replacement experiences have been very satisfactory as reflected in the maintenance of high capacity factors of the Ontario Hydro commercial nuclear generating stations.

8.0 ACKNOWLEDGEMENTS

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9.0 REFERENCES

(1) **McConnell, L.G., Woodhead, L.W., Fanjoy, G.R.** — *CANDU Operating Experience,* Ontario Hydro Report NGD-9 (1981) of March 1982.

(2) **Perryman, E.C.W.** — *Pickering Pressure Tube Cracking Experience,* J. Nucl. Energy, 1978, 17 April, No. 2, 95-105 (AECL-6059).

(3) **Kupcis, O.A. et al** — *Non-Destructive Inspection of Pressure Tubes at the Pickering Generating Station.* Third conference on Periodic Inspection of Pressurized Components. Institute of Mechanical Engineers, London, 1977.

(4) **Ross-Ross, P.E. et al** — *Some Engineering Aspects of the Investigation into the Cracking of Pressure Tubes in the Pickering Reactors,* Annual Conference of the Engineering Institute of Canada, 1975 (AECL-5261).

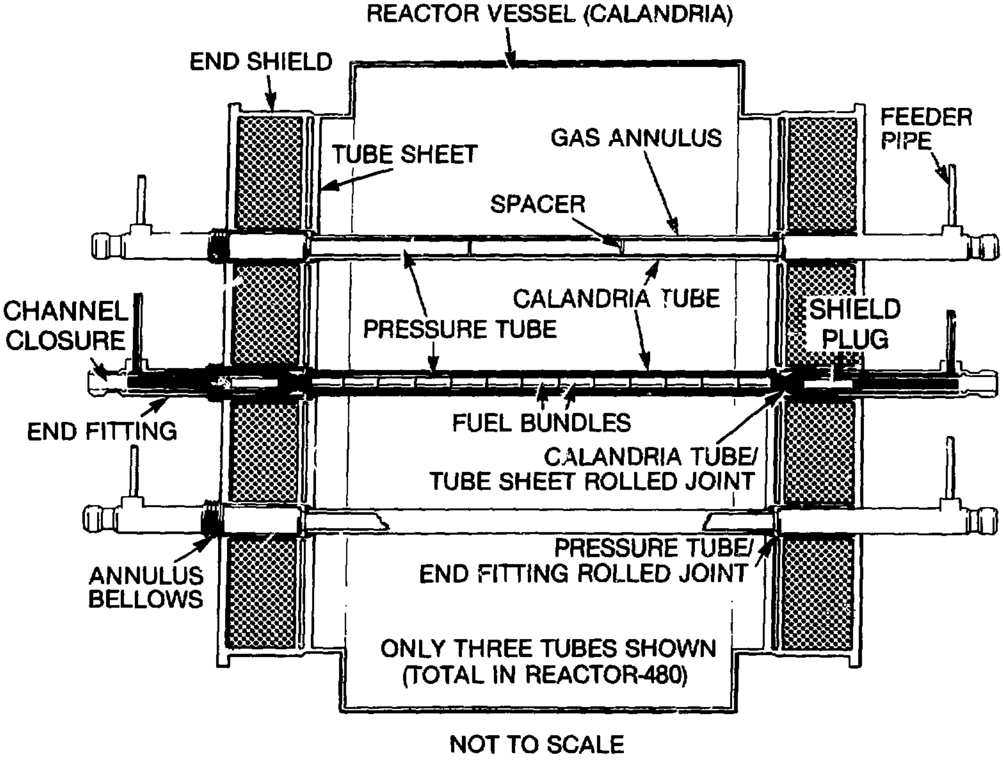

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FIGURE 1 SCHEMATIC OF CANDU — PHW REACTOR

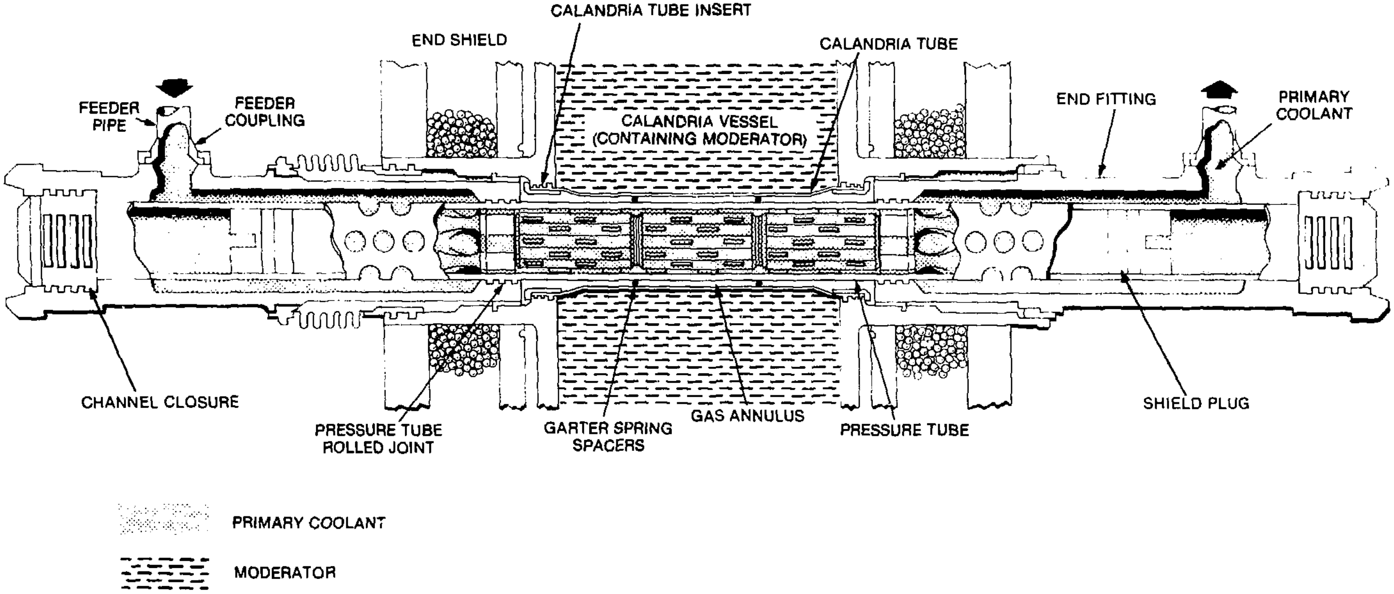

FIGURE 2 SCHEMATIC OF A CANDU-PHW FUEL CHANNEL ASSEMBLY

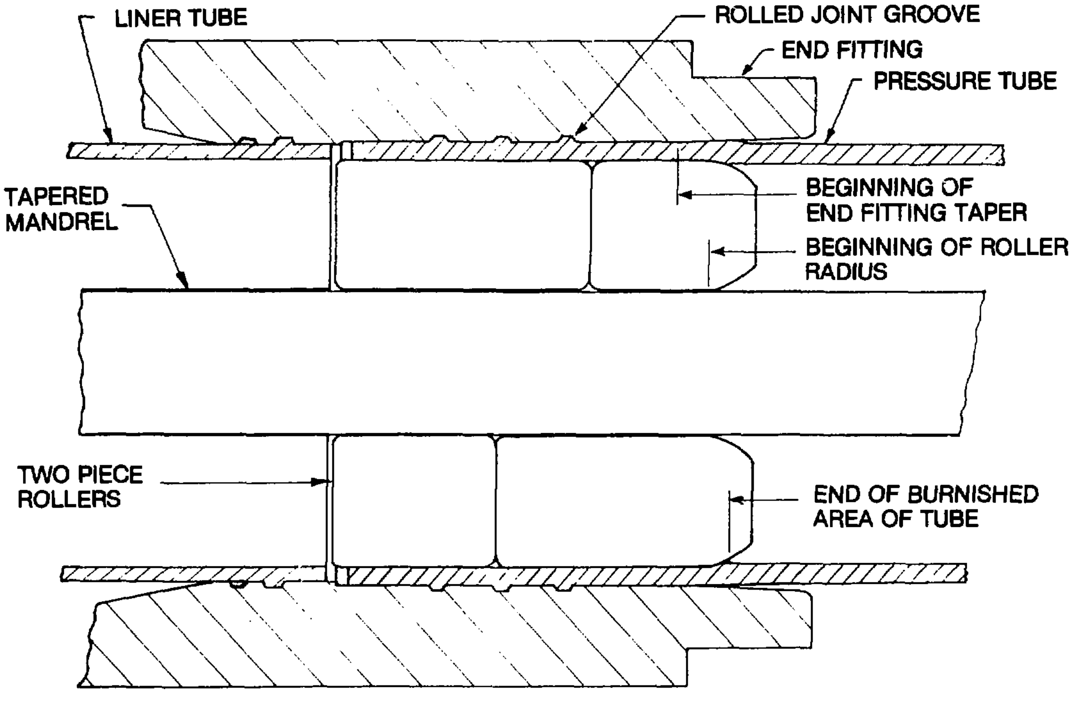

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FIGURE 3 CROSS SECTION OF A PRESSURE TUBE ROLLED JOINT DURING ROLL EXPANDING

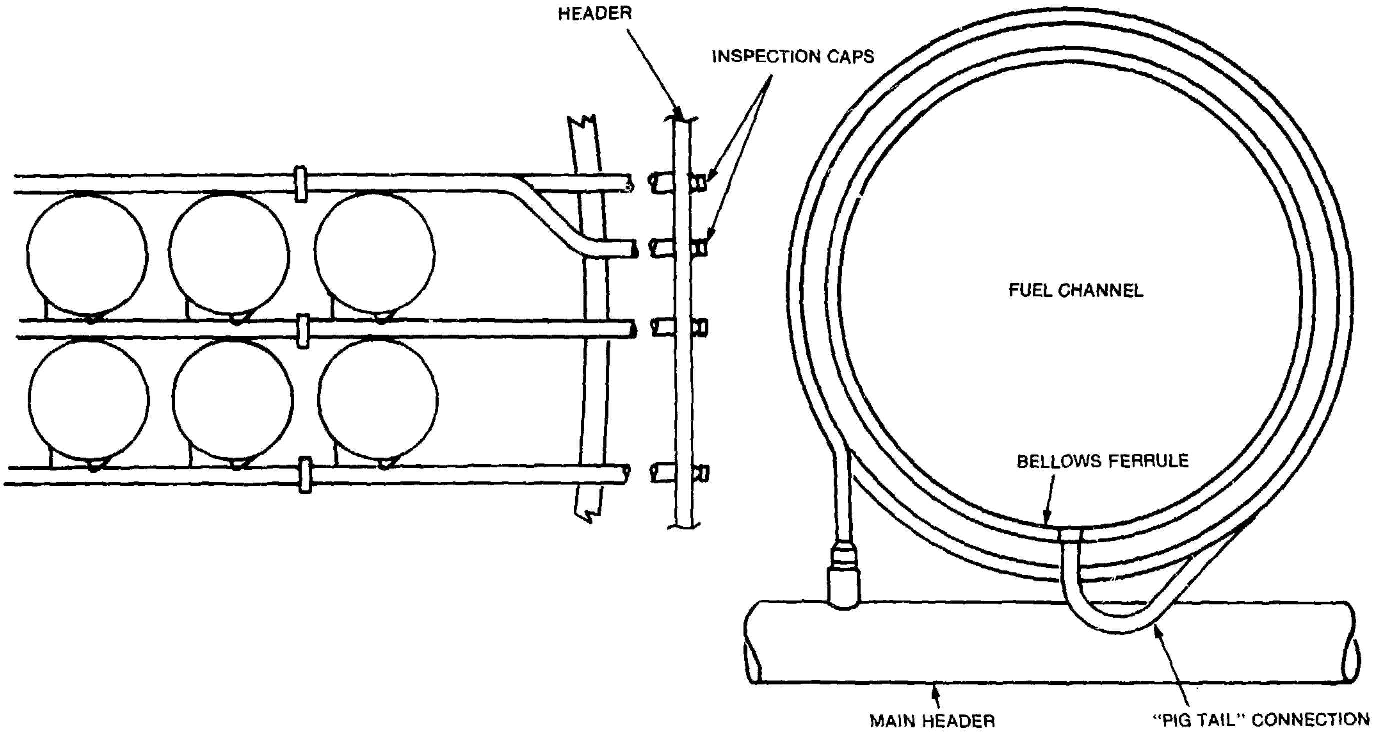

FIGURE 4 SCHEMATIC SHOWING GAS ANNULUS HEADER AND PIGTAILS

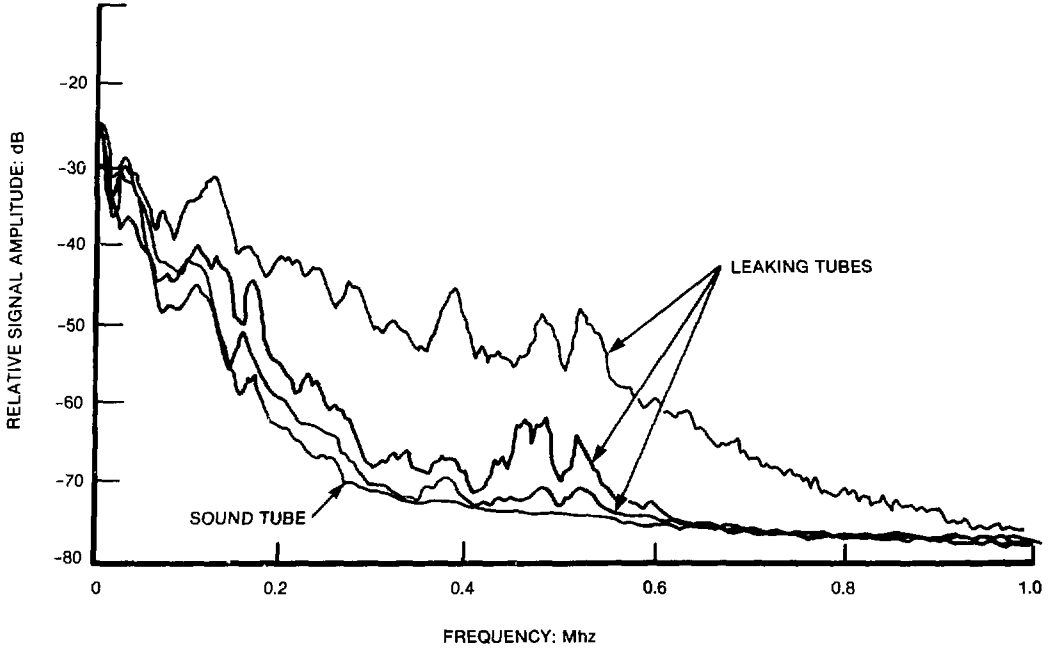

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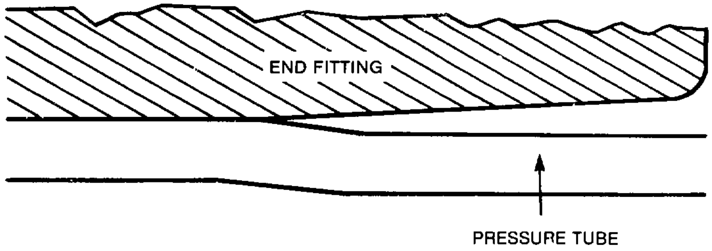
FIGURE 5 CHARACTERISTIC ACOUSTIC EMISSION FREQUENCY SPECTRA



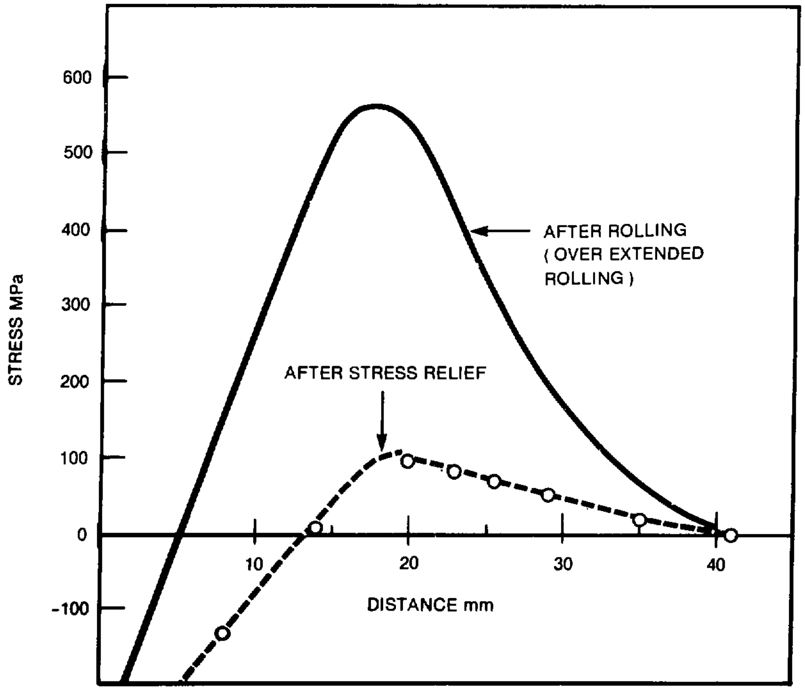


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FIGURE 6 BRUCE NSG-2, UNIT 2 RESIDUAL STRESSES (HOOP STRESS ON INNER WALL OF PRESSURE TUBE)

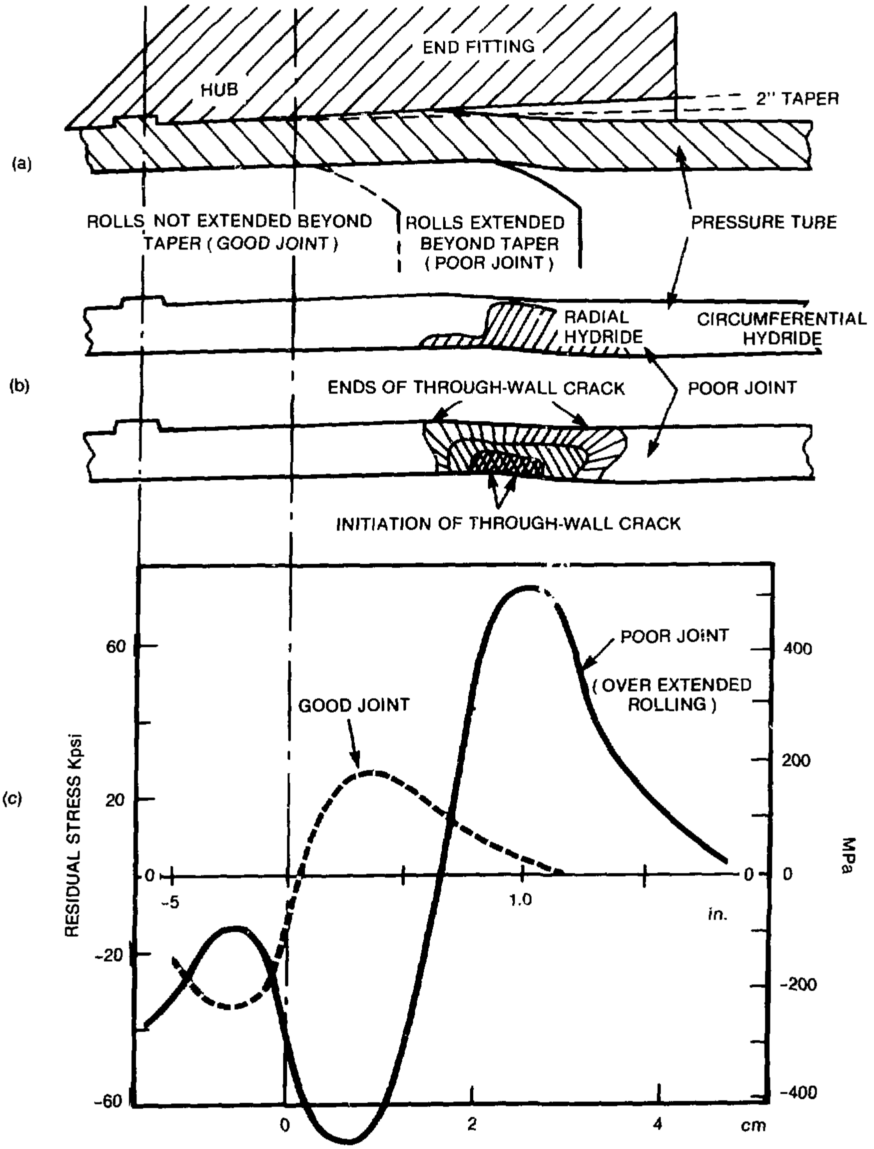

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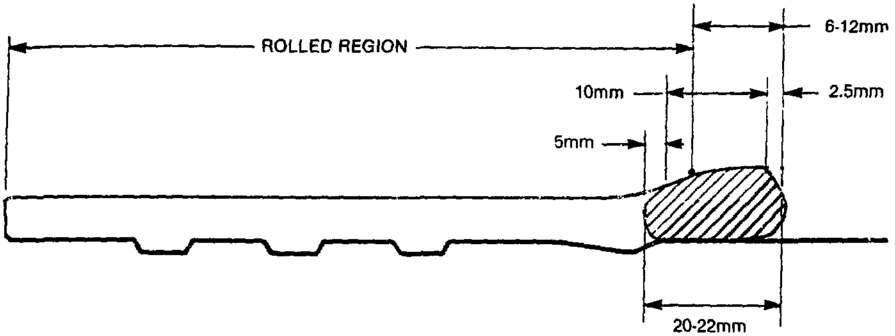
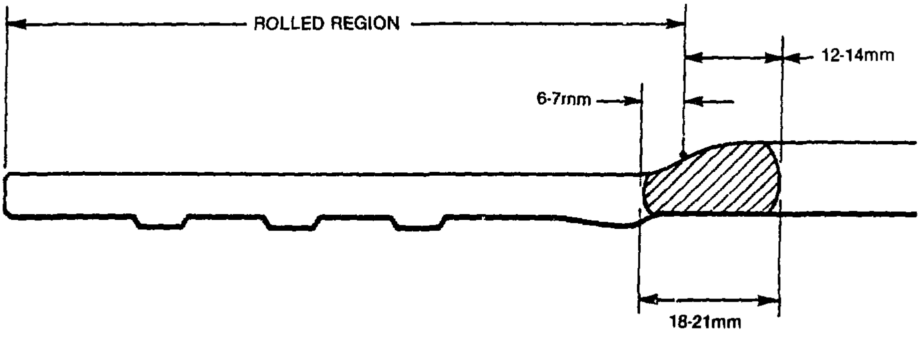
FIGURE 7 PICKERING ROLLED JOINT RESIDUAL STRESSES
(HOOP STRESS ON INNER WALL OF THE PRESSURE TUBE)



BRUCE NGS-A, UNIT 2



PICKERING NGS-A, UNIT 3

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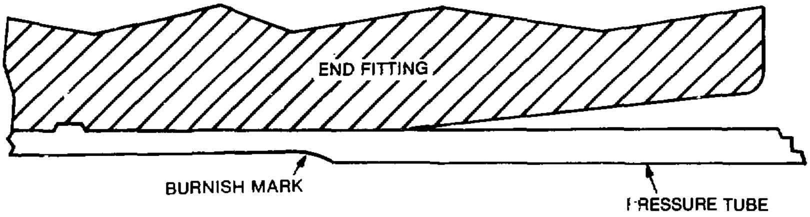
FIGURE 8 DIMENSIONS AND LOCATION OF DEFECT IN PRESSURE TUBE



RESIDUAL HOOP STRESS DISTRIBUTION IN PRESSURE TUBE BORE

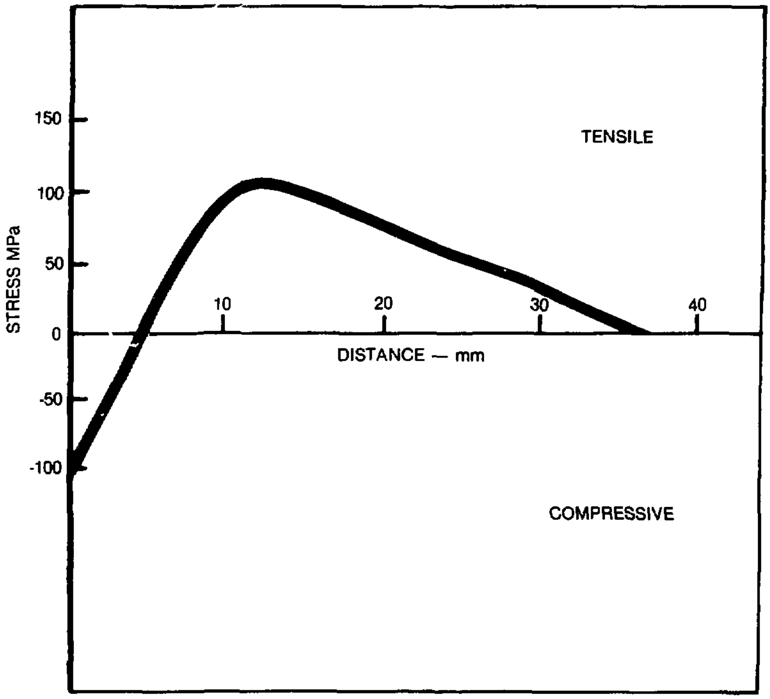


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FIGURE 9 ZERO CLEARANCE ROLLED JOINT RESIDUAL STRESS (HOOP STRESS)

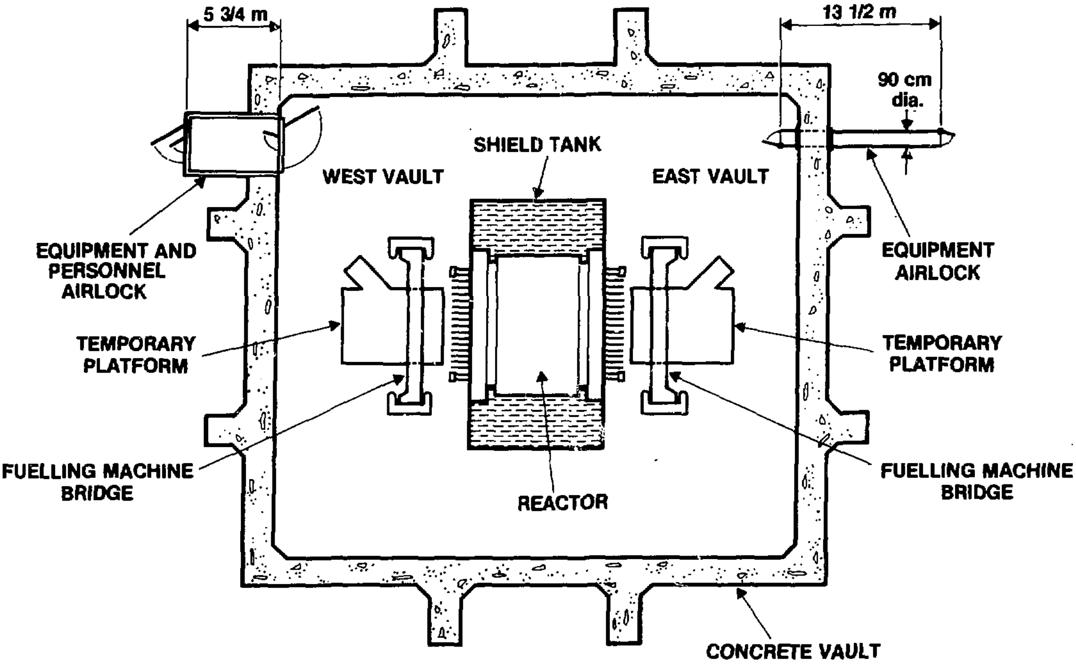

FIGURE 10 PLAN OF BRUCE UNIT 2 REACTOR VAULT

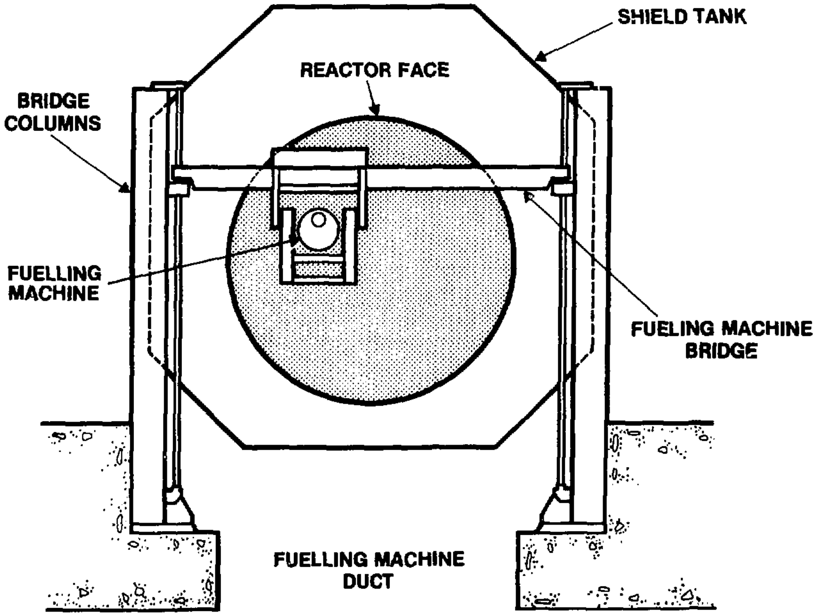

FIGURE 11 ELEVATION OF BRUCE REACTOR FACE

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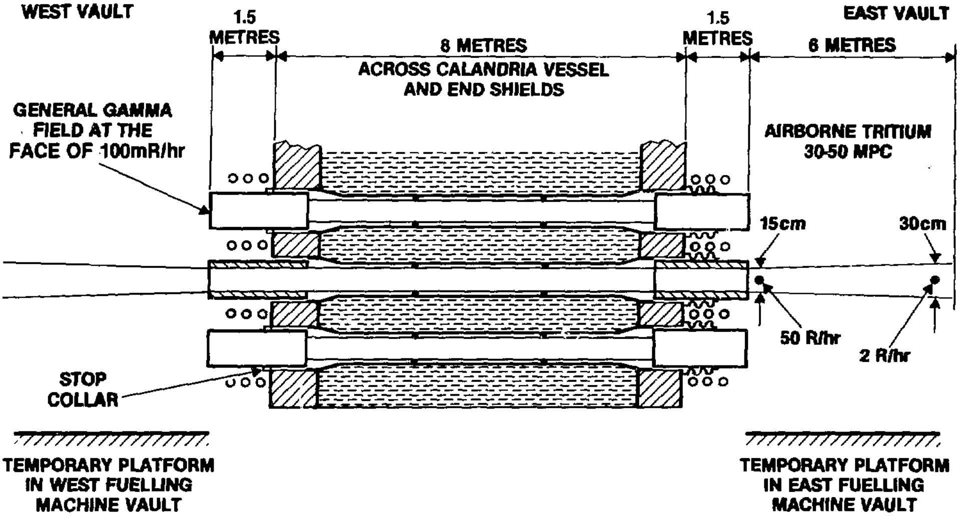

FIGURE 12 RADIATION BEAM IN THE FUELLING MACHINE VAULTS WITH SHIELDING PLUGS REMOVED FROM THE CHANNEL

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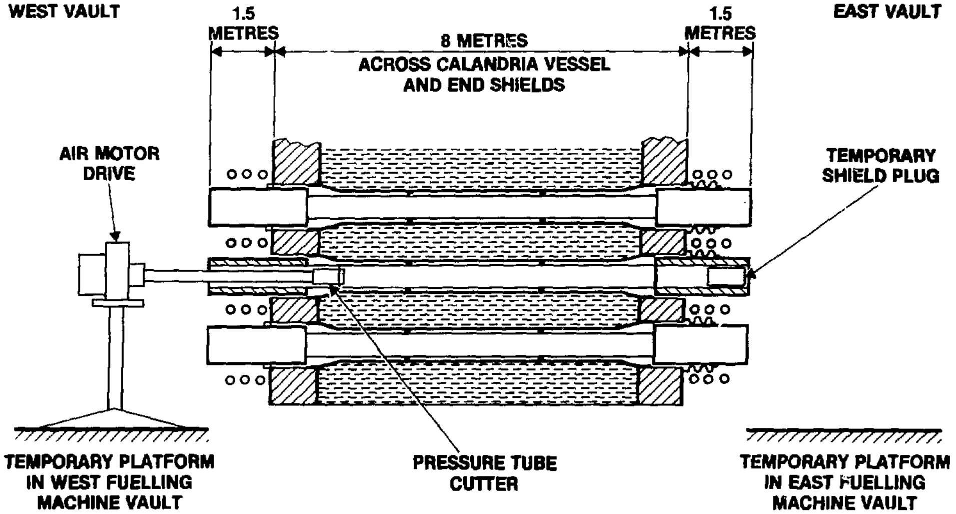

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FIGURE 13 THE FIRST PRESSURE TUBE CUT

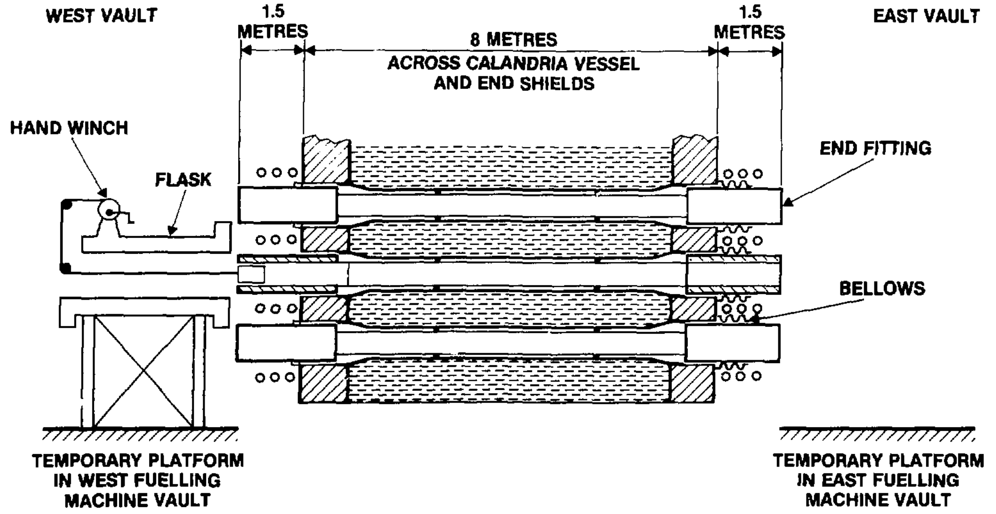

FIGURE 14 LEAD FILLED FLASK READY TO RECEIVE WEST END FITTING

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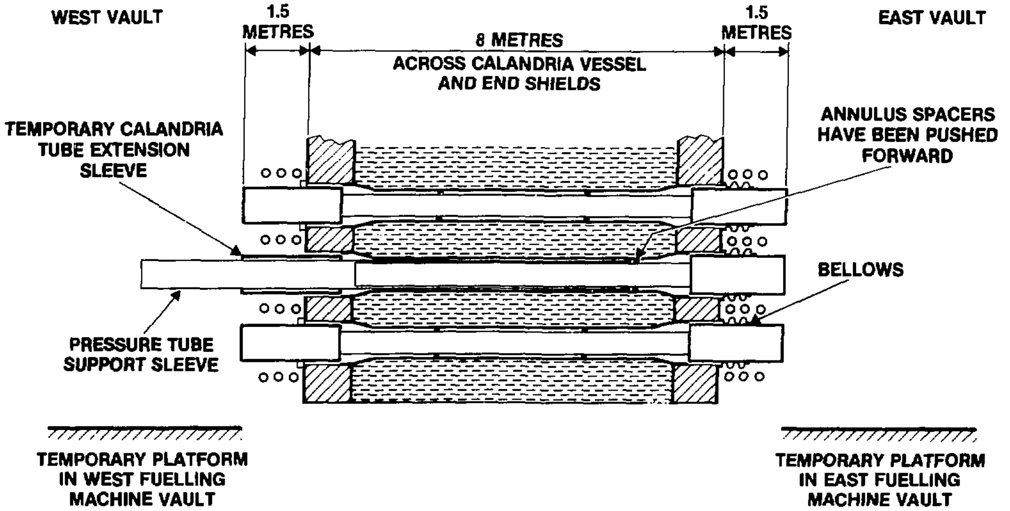

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FIGURE 15 SUPPORT SLEEVE INSERTED IN THE ANNULAR SPACE BETWEEN THE CALANDRIA TUBE AND PRESSURE TUBE

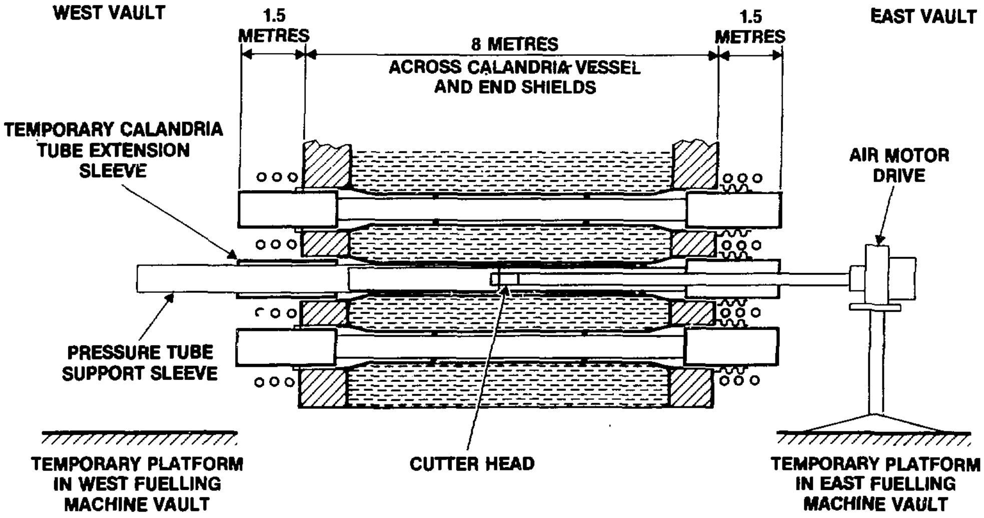

FIGURE 16 PRESSURE TUBE CUTTING AT THE CENTER OF THE REACTOR

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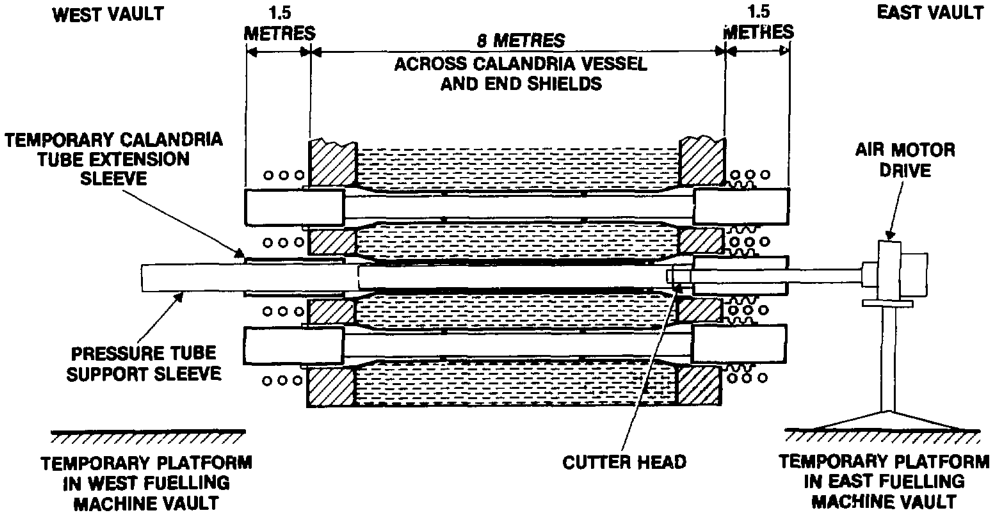

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FIGURE 17 THIRD AND FINAL PRESSURE TUBE CUT

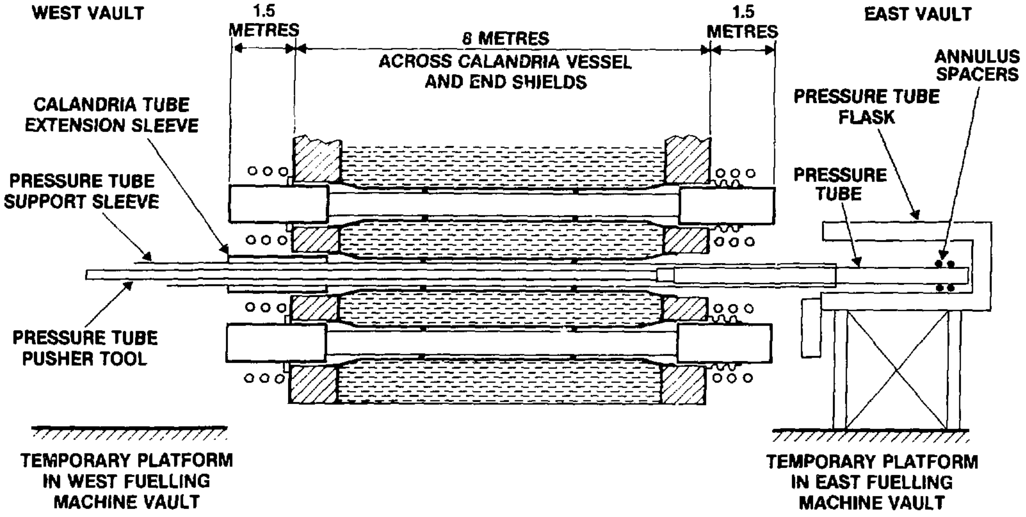

FIGURE 18 FIRST HALF OF THE PRESSURE TUBE BEING PUSHED INTO THE FLASK

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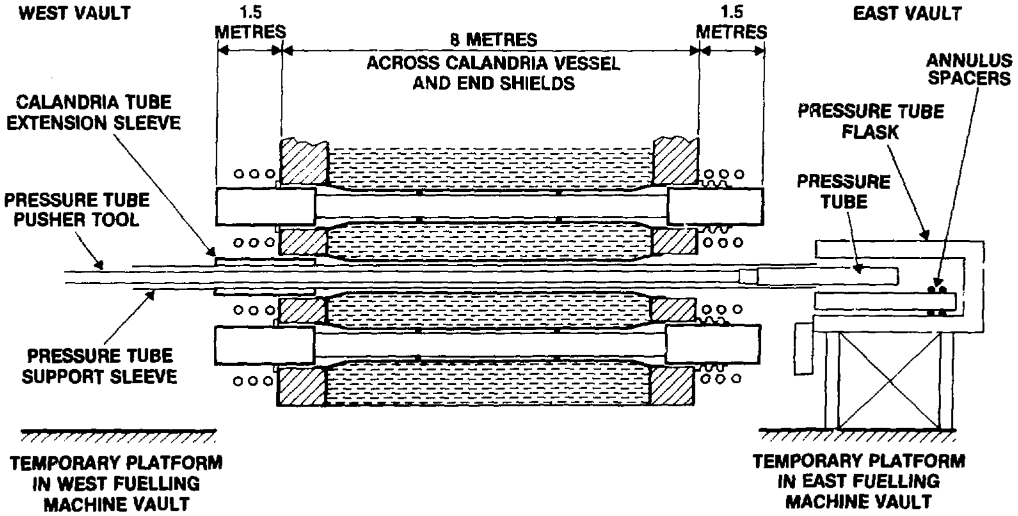

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FIGURE 19 SECOND HALF OF THE PRESSURE TUBE BEING PUSHED INTO THE FLASK

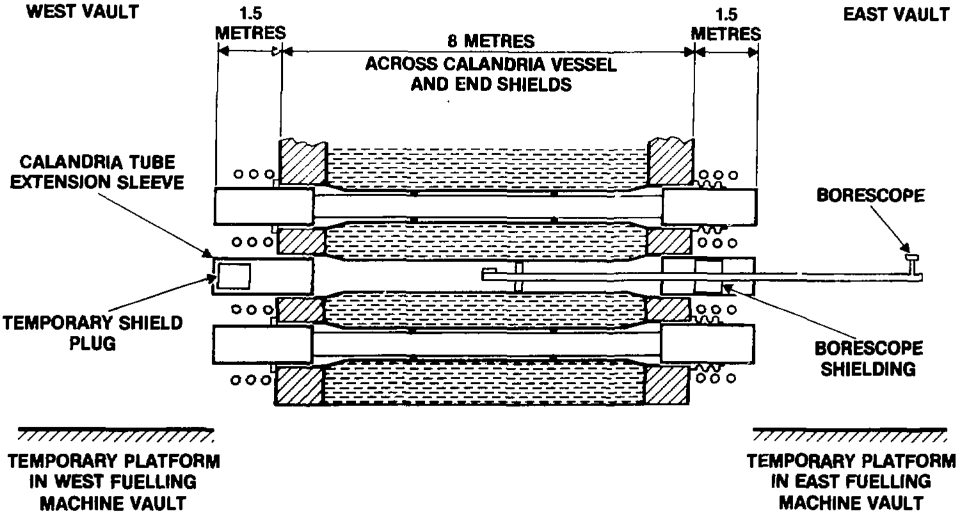

FIGURE 20 BORESCOPE INSIDE THE CALANDRIA TUBE

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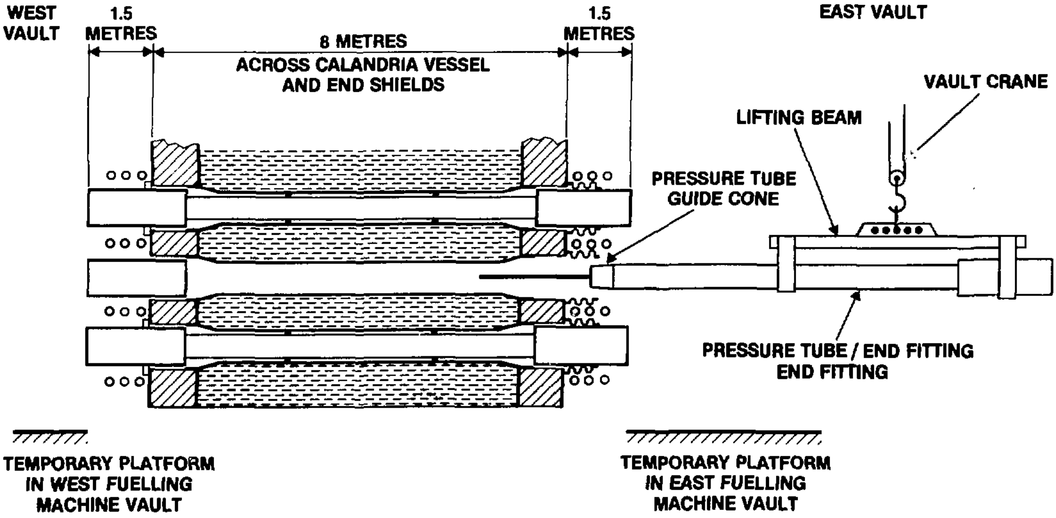

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FIGURE 21 REPLACEMENT SUB-ASSEMBLY SUSPENDED FROM THE EAST VAULT CRANE

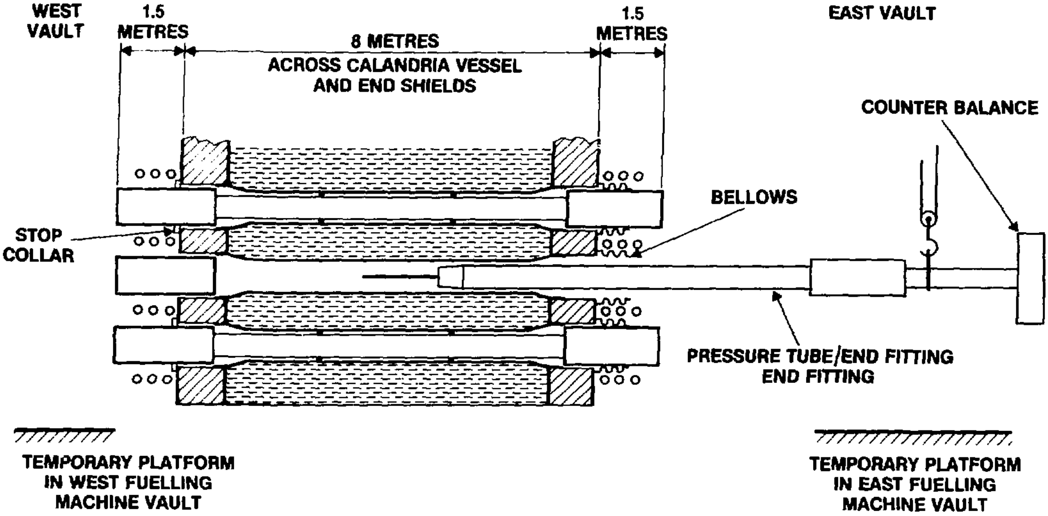

FIGURE 22 SUB-ASSEMBLY INSERTION USING THE COUNTERBALANCE WEIGHT

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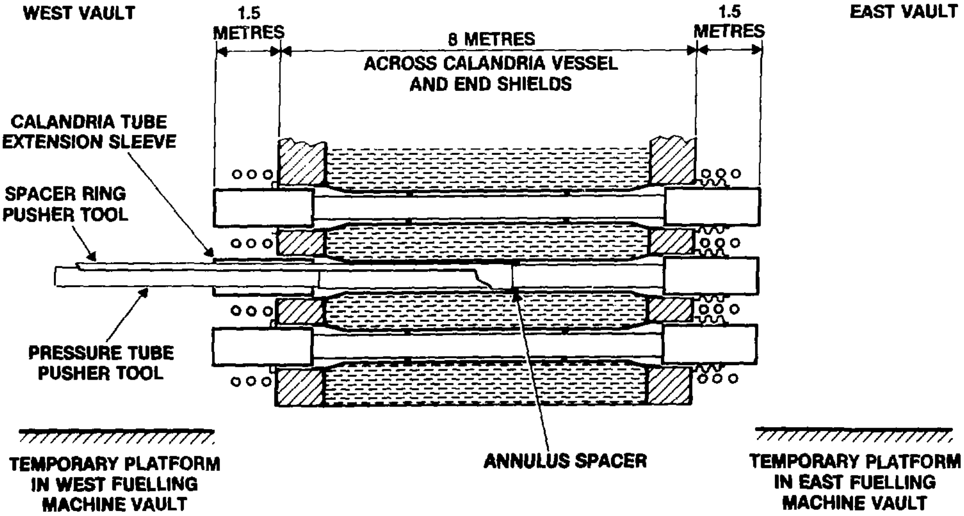

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FIGURE 23 INSTALLATION OF THE ANNULUS SPACER RINGS

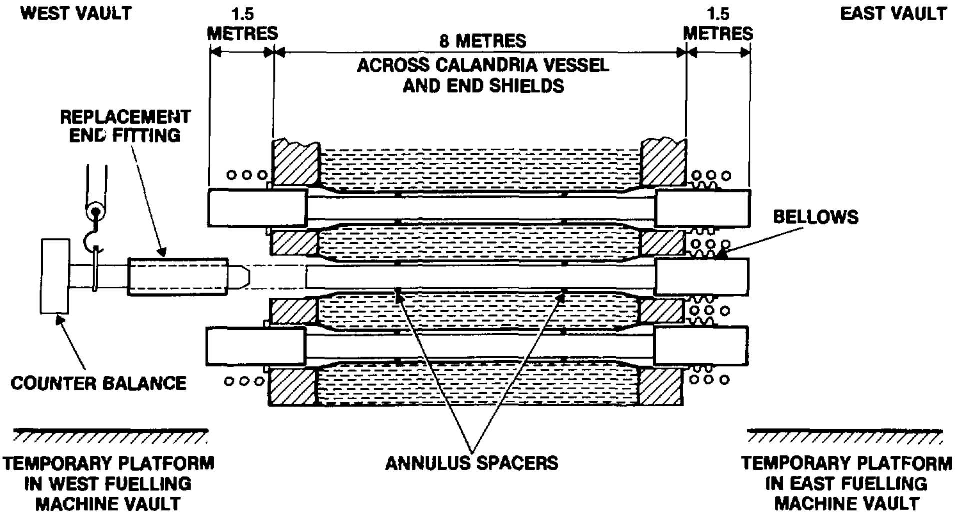

FIGURE 24 END FITTING INSERTION USING A COUNTERBALANCE TOOL

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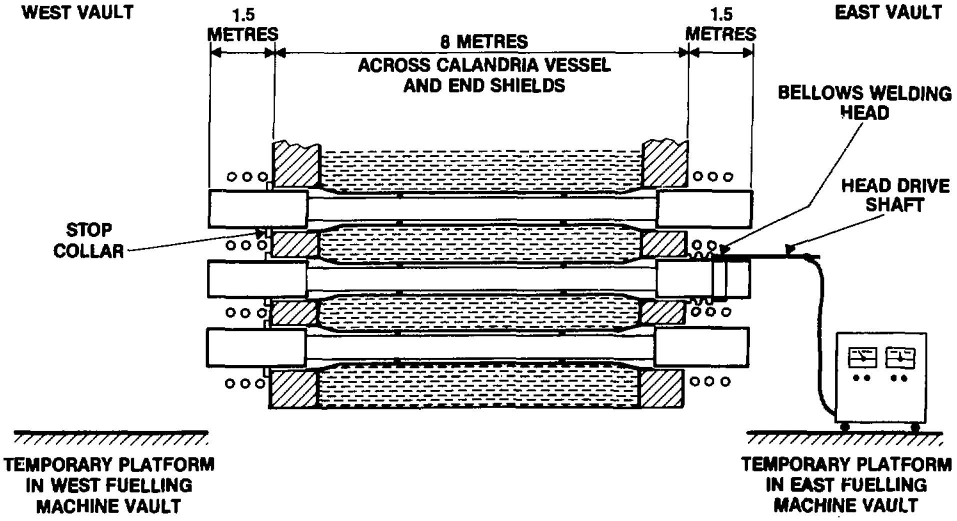

FIGURE 25 BELLOWS WELDING

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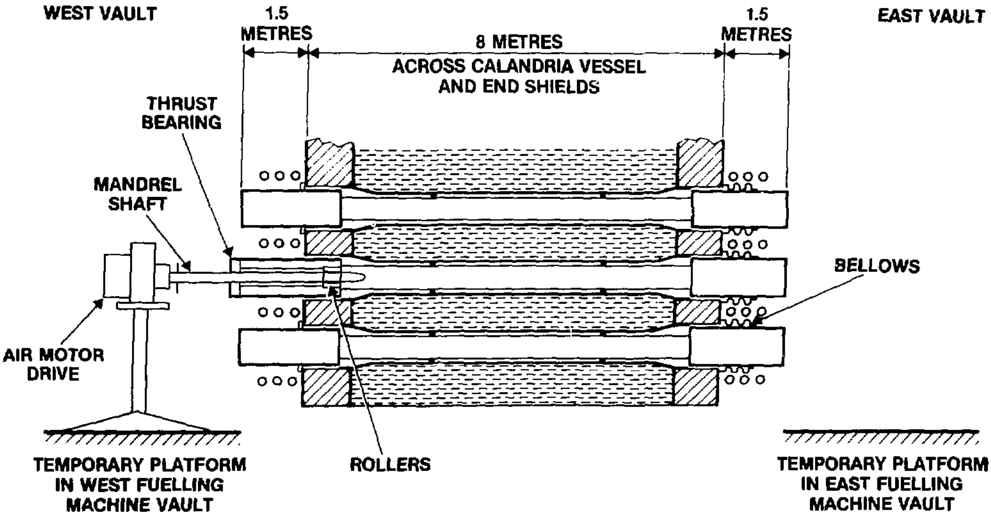

FIGURE 26 ROLLING THE PRESSURE TUBE TO END FITTING JOINT

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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

*SINCE FIRST PRODUCTION OF ELECTRICITY TO END OF DECEMBER 1981.

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

***INCLUDES 4 MONTH STRIKE IN 1972.

TABLE 1 CANADIAN RANKING IN WORLD'S 131 COMMERCIAL REACTORS 500 MWe AND LARGER — LIFETIME*

4 UNITS – 15.5 UNIT YEARS

CAPABILITY FACTOR: 83.5% INCAPABILITY FACTOR:*** 16.5%

CAUSE OF INCAPABILITY	INCAPABILITY (%)
ON–POWER FUELING	0.8
FUEL	0.0
HEAT TRANSPORT PUMPS	0.2
PRESSURE TUBES	0.3
BOILERS	2.4
TURBINE AND GENERATORS	6.6
INSTRUMENTATION AND CONTROL	1.7
HEAT EXCHANGERS	0.0
VALVES	4.5
OTHER	4.5

*LIFETIME MEANS SINCE IN–SERVICE DATE OF EACH UNIT.

**FIGURES INCLUDE A 4–MONTH STRIKE IN 1972. (UNITS 1 TO 3 WERE SHUT DOWN).

***INCAPABILITY FACTOR %: – ICbF %

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

TABLE 2 BRUCE NGS-A — LIFETIME* INCAPABILITY TO DECEMBER 31, 1981**

4 UNITS – 37.6 UNIT YEARS

CAPABILITY FACTOR: 80.2% INCAPABILITY FACTOR:*** 19.8%

CAUSE OF INCAPABILITY	INCAPABILITY (%)
ON–POWER FUELING	0.8
FUEL	0.1
HEAT TRANSPORT PUMPS	0.2
PRESSURE TUBES	4.9
BOILERS	0.5
TURBINE AND GENERATORS	5.8
INSTRUMENTATION AND CONTROL	0.7
HEAT EXCHANGERS	0.9
VALVES	0.4
OTHER	5.6

*LIFETIME MEANS SINCE IN–SERVICE DATE OF EACH UNIT.

**FIGURES INCLUDE A 4–MONTH STRIKE IN 1972. (UNITS 1 TO 3 WERE SHUT DOWN).

***INCAPABILITY FACTOR %: – ICbF %

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

TABLE 3 PICKERING NGS-A — LIFETIME* INCAPABILITY TO DECEMBER 31, 1981**

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	58.3				
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
1982 TO END OF JUNE	0	0	0	0	0	19.4	0	0

TABLE 4 INCAPABILITY DUE TO FUEL CHANNELS

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