PROBLEMS EXPERIENCED DURING OPERATION OF THE PROTOTYPE FAST REACTOR, DOUNREAY, 1974–1994

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Abstract

The UK Prototype Fast Reactor, PFR, was designed in the 1960s and was operated at Dounreay in Scotland from 1974 to 1994. By the time it was shut down it had demonstrated the feasibility of the technology of a large sodium-cooled fast breeder reactor, and had been shown to operate safely and reliably. It had also provided an invaluable test facility for advancing the technology, particularly in developing advanced fuel and cladding materials that had achieved high burnup and neutron dose.

As is usual in prototype plants the operation of PFR revealed the weak points of the original design concept. Several difficulties were encountered in the course of its operating life, all of which were successfully overcome. The purpose of this paper is to describe some of these difficulties and the steps taken to master them. In this way the benefit of the experience gained and the lessons learnt can be made available to the designers and operators of reactors of similar type. The intention is that future generations will not follow false trails in the further development of this promising technology.

INTRODUCTION

Nine major incidents and unforeseen developments are described. They are

1. A series of steam-generator gas-space leaks,
2. A major under-sodium leak in a steam generator,
3. The incidence of cracking in the steel of various secondary sodium circuit components,
4. Blockage of the secondary sodium cold trap,
5. Seizure of the primary sodium cold trap pump,
6. The effect of sodium aerosol deposits of the operation of primary circuit components,
7. Malfunctioning and cracking in the air heat exchangers of the decay heat rejection loops,
8. Neutron-induced distortion of core components and its effect on plant_operation, and
9. A major oil leak into the primary circuit.

Each of these is summarised below, with diagrams. The causes and the steps taken to rectify the problem are explained, and the general lessons learnt are set out in the context of the future development of LMFR technology. References are given in the cases for which more detailed information has been published.

1. PFR STEAM GENERATOR GAS SPACE LEAKS

PFR had three secondary circuits, each of which had an evaporator, a superheater and a reheater. Figure 1.1 shows the general arrangement, and Figure 1.2 shows an evaporator in more detail. A total of 37 gas-space leaks was experienced in PFR steam generator units in the period 1974 to 1984 with 33 of these occurring in evaporators, 3 in superheaters and 1 in a reheater. All of the gas-space leaks originated at the welds between the tubes and the tubeplates. The effect of these leaks on PFR availability was considerable, so that the highest annual load factor prior to 1984 was only 12%.

PFR went critical for the first time in March 1974 and commissioning of the steam generators followed. Up to 1976 there were failures in gas-space leaks in one evaporator, two superheaters and one reheater. These early failures are believed to have been due to manufacturing faults.

In the case of the austenitic superheaters and reheaters the leaks gave rise to considerable concerns about the design. Although both the damaged superheaters continued in use up to 1986, having had the leaking tubes plugged, one of the superheaters and the reheater had suffered from caustic stress corrosion cracking of the tube plate caused by the products of the sodium-water reaction. The superheater was salvaged by grinding out the cracks and thoroughly washing the tube plate with hot sodium to remove the reaction products. Damage to the reheater tube plate was so extensive that the tube bundle was

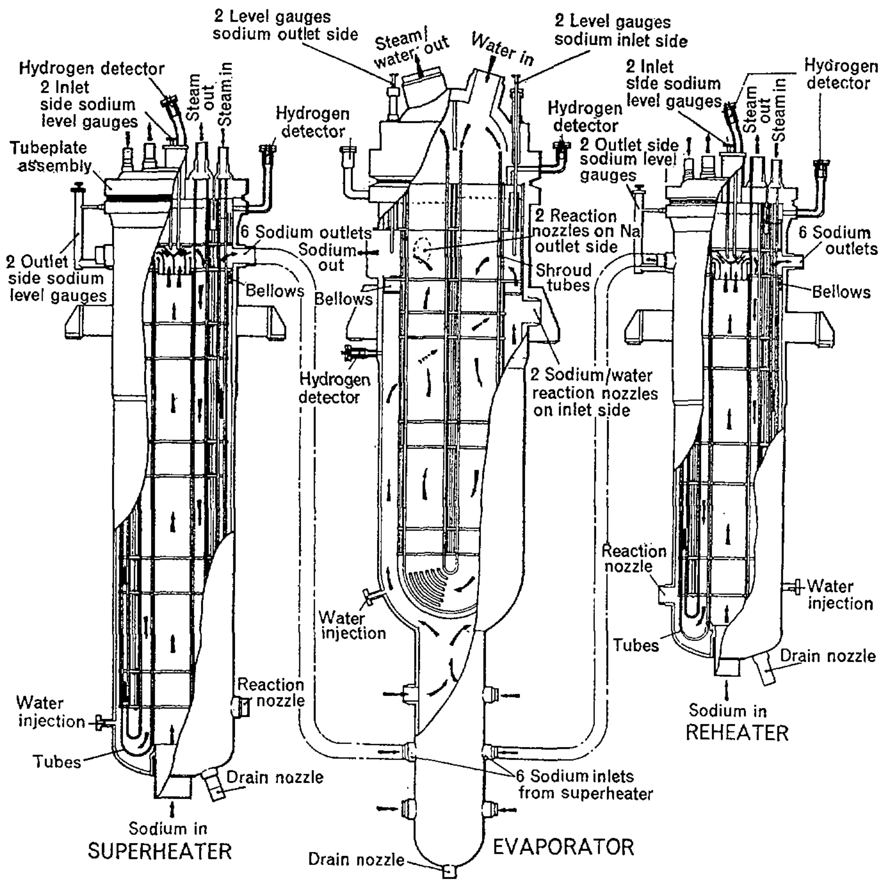


Figure 1.1. PFR Steam Generators

scrapped. It was replaced by a plug in the empty reheater vessel until a replacement was fitted in 1984. For this period the plant had to be operated with reduced reheat capacity.

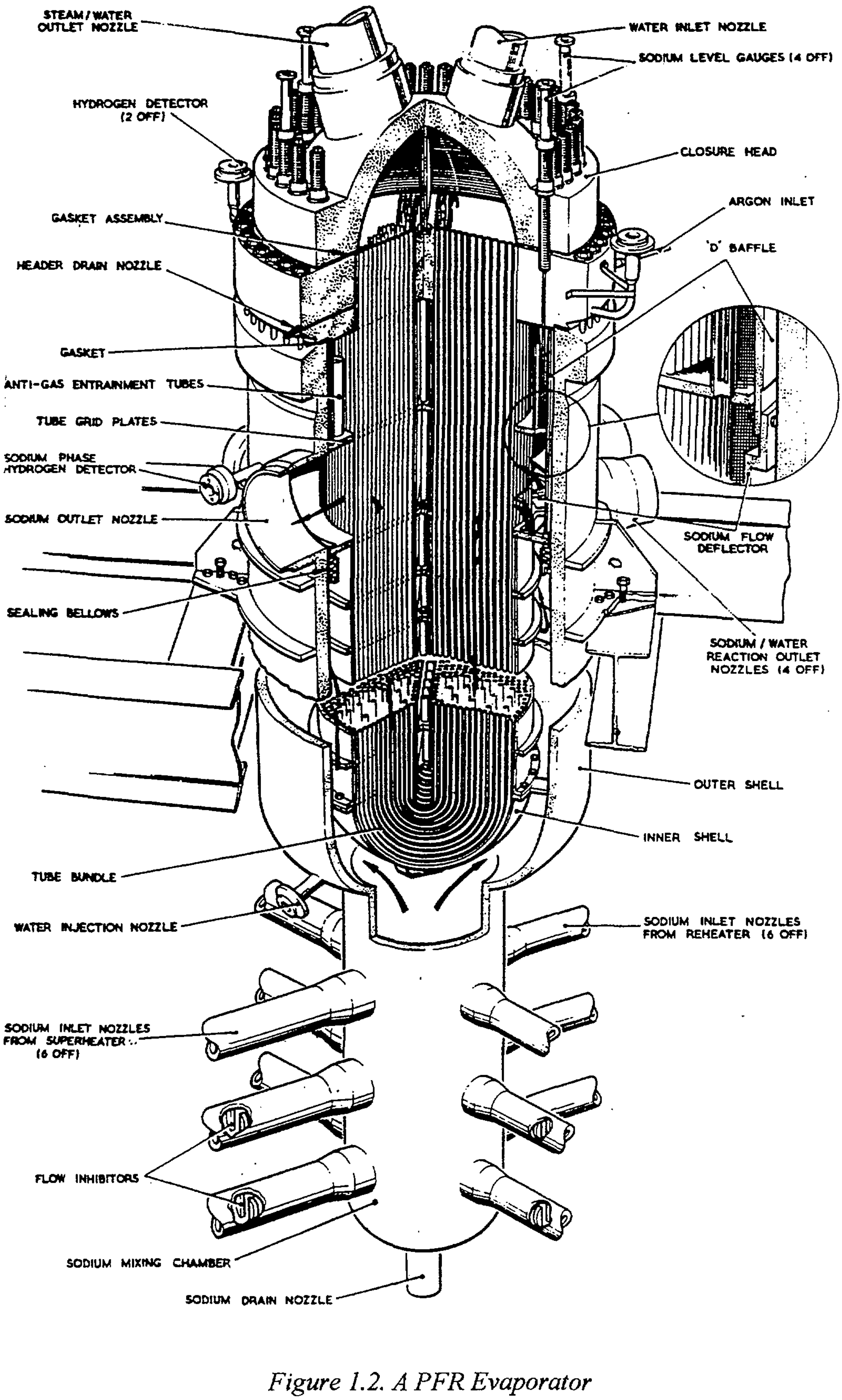


Figure 1.2. A PFR Evaporator

Following the early failures of tube-to-tubeplate welds in the two superheaters and the reheater no further failures occurred in the austenitic units until 1986, when a superheater tube leaked while the unit was being pressurised with steam prior to being put on line. This incident is described below.

In the period 1984 - 1987 all the six austenitic tube bundles were replaced by new tube bundles, shown in Figure 1.3. The design benefited from the early experience of caustic stress corrosion following the leaks in the austenitic units. The new tube bundles were fabricated in 9Cr 1Mo ferritic steel, and six were available to replace the original units by 1984. The replacement work was completed by 1987.

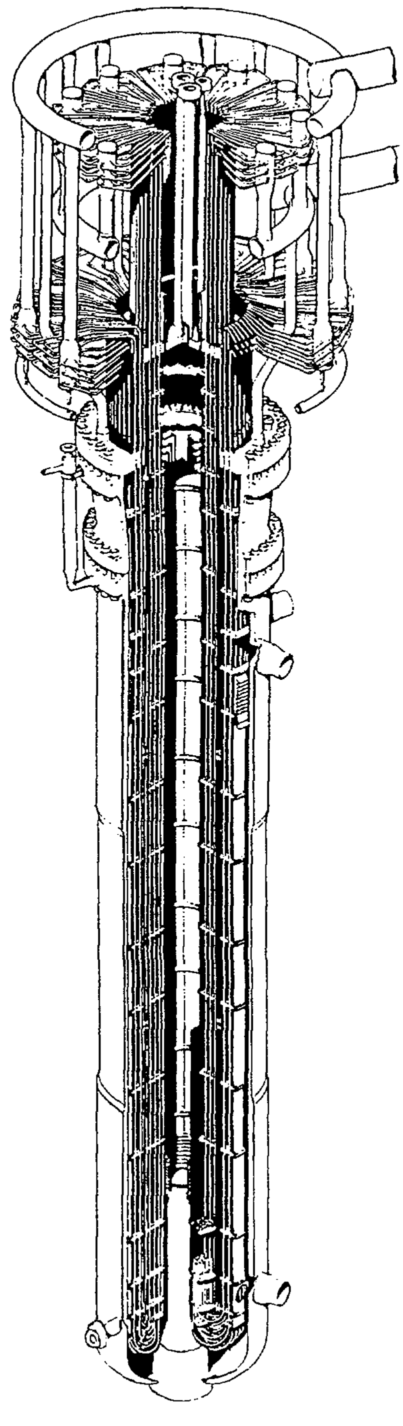


Figure 1.3. A PFR Replacement Reheater Tube Bundle

The 33 leaks experienced in the ferritic steel evaporator units (Figure 1.2) were relatively benign as ferritic steel is not so subject to caustic stress corrosion, but the effect on availability while leaking tubes were being repaired was considerable.

The evaporator gas space leaks were all associated with cracking of the tube-to-tubeplate welds. These were hard and had high residual stresses because there was no post-weld heat treatment. None of the evaporator leaks gave evidence of wastage damage to the neighbouring tubes, probably because they were detected early by the installed gas-space hydrogen detection system. This was based on katharometers and was very sensitive, being capable of detecting leaks as small as 0.1 mg/s. The leaks were repaired by plugging the affected steam tubes.

Nevertheless it appeared that one leak would, after a few days or weeks of further operation, cause others. It was concluded that residual caustic reaction products in the gas space above the sodium caused further cracking of welds and initiated more leaks after an incubation period. Sodium flooding of the tubeplate at a temperature in excess of 400 °C for periods in excess of 24 hours had some success in removing reaction products. It was also required to wash out sodium hydrides which could lead to false hydrogen detection signals when they dissociated at high operating temperatures.

This did not cure the problem completely, however, and eventually it was concluded that washing with hot sodium did not remove caustic material from the roots of pre-existing fine cracks in the welds, so that in the presence of the residual stresses corrosion continued and the cracks grew to give rise to further leaks.

The problem was finally solved by the fitting of sleeves which spanned the original welds, as shown in Figure 1.4. In all 3000 sleeves were fitted over a 14-month period. The work was completed early in 1984. While sleeving was underway the station operated on a single circuit. Following sleeving no further problems were experienced with the evaporators.

Conclusions

Gas space leaks in PFR steam generators provided valuable information on the behaviour and detection of such leaks. They proved to be readily detectable by means of the hydrogen generated. Careful washing of the tubeplates with hot sodium limited the numbers of leaks and avoided major further plant damage, but did not cure the problem. Eventually a radical solution, involving sleeving all the tub-to-tubeplate weds in the three evaporators, had to be adopted.

The type of direct tube-to-tubeplate weld adopted initially at PFR, which could not be heat treated after manufacture, should be avoided in future fast reactors. Austenitic steels are unsuitable for LMFR steam generators because of the high risk of caustic stress corrosion damage following even small leaks.

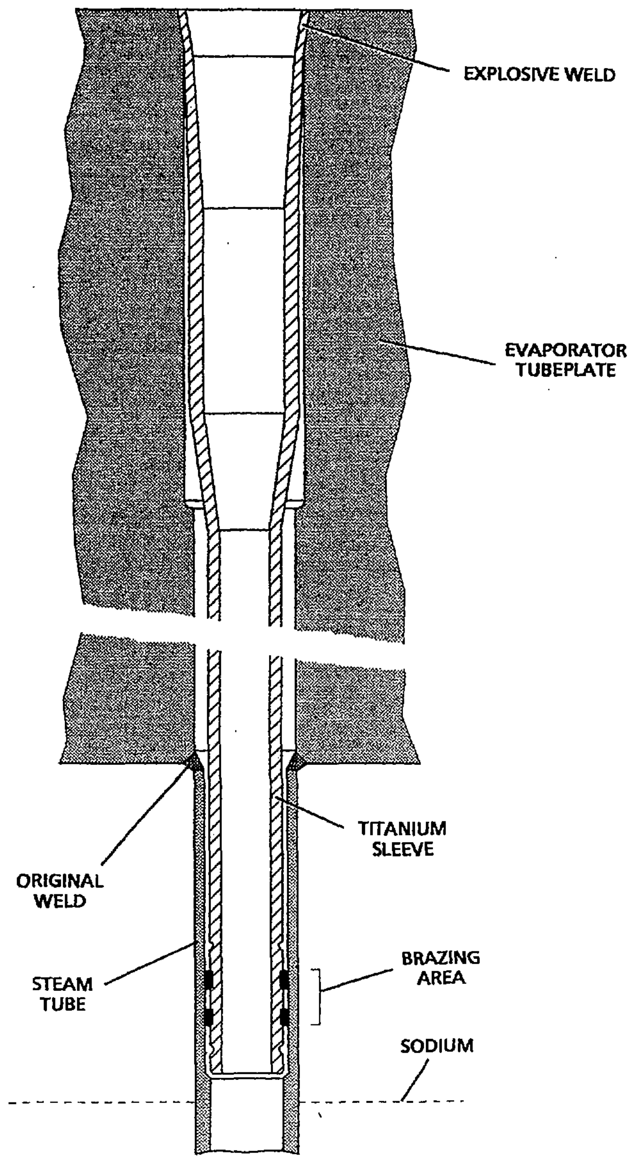


Figure 1.4. A PFR Evaporator Weld Repair Sleeve

REFERENCES TO SECTION 1

E R Adam and C V G Gregory "A Brief History of the Operation of the Prototype Fast Reactor at Dounreay": *The Nuclear Engineer*, 1994, **35**, 112 - 117

2. THE UNDER-SODIUM LEAK IN PFR SUPERHEATER 2

An under-sodium leak occurred in PFR superheater 2, one of the original units made from austenitic steel, in February 1987. It provided valuable information on the behaviour of sodium-water reactions in an operating steam generator and led to a complete re-assessment of the design-basis steam generator accident for subsequent fast reactors.

On 27 February 1987 PFR was operating at full power when a sodium-water reaction trip was caused by the rupture of a bursting disc on the stem side of superheater 2. This initiated a dump of the steam and sodium in the secondary circuit and automatic shutdown of the plant. Shutdown to a safe state took approximately 10 seconds, as designed.

It was confirmed shortly after the incident that a large under-sodium leak had occurred in superheater 2. Figure 2.1 shows one of the original PFR superheaters. After the sodium circuit had been cleaned to remove reaction products the superheater tube bundle was removed from its vessel in a nitrogen-filled bag and examined. This revealed that between two tube support grids one of the six baffle plates forming the central sodium inlet duct had become detached, and the remaining 5 plates in this region were deformed. Considerable distortion of steam tubes could be seen through the aperture left by the missing baffle plate.

The entire tube bundle was then dismantled and forty steam tubes were found to have ruptured, with longitudinal gapes of such a size as to be effectively equivalent to double-ended guillotine breaks. The locations of the failed tubes are shown in Figure 2.2. Sixteen of the failures were in the row adjacent to the central sodium duct and faced towards the duct. Four of the failed tubes had wear flats facing the central baffle and on one of these (tube 16) a circumferential crack at right angles to the main fracture was apparent. This was opposite a small wastage pit in the baffle, and it is concluded that this was the initiating leak.

Evidence of fretting on a further 13 tubes adjacent to the baffle was found, all close to the seams between the six plates forming the baffle. It was concluded that sodium leakage through the seams, shown in Figure 2.3, caused flow-induced vibration of the tubes resulting in contact with the duct and giving rise to wear and the eventual failure of tube 6.

Subsequent analysis of the event gave rise to the following conclusions:

1. A total of 40 tubes failed, 39 due to overheating in a period of 8 seconds following the plant trip. The remaining tube, which initiated the event, failed due to fretting damage caused by tube vibration.

2. The primary small leak grew after passing steam for some tens of seconds. Finally it grew rapidly to give a leak rate of 0.5 to 1.0 kg/s for a period of a few seconds.

3. This induced a plant trip by rupturing a steam-side bursting disc. Isolating valve closed. causing the steam flow to stop, but the steam pressure fell relatively slowly over a period of about 10 seconds.

4. Other tubes already weakened by fretting failed quickly, within a second or so, causing an increase in the heat generated by the sodium-water reaction.

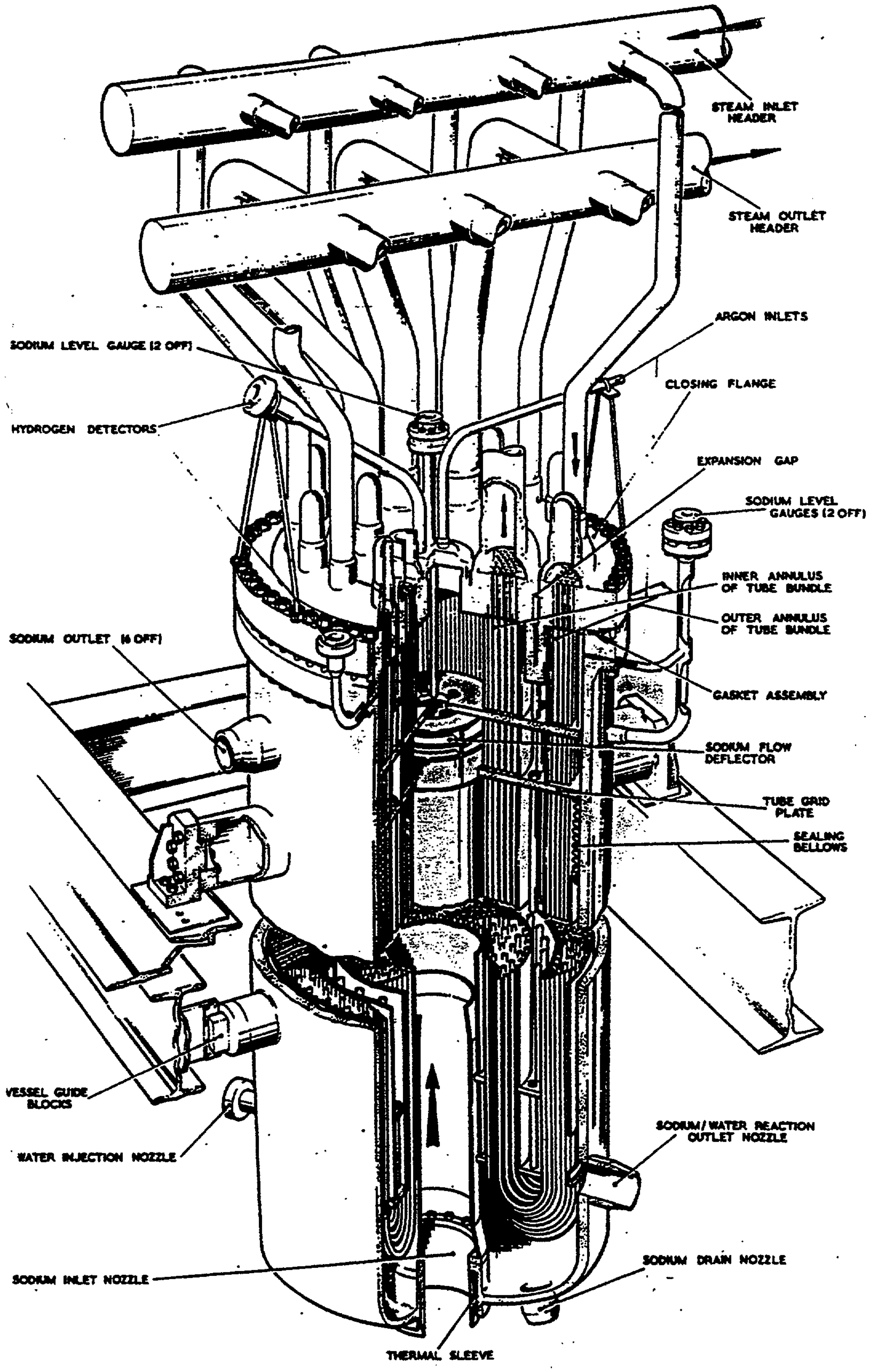


Figure 2.1 An Original PFR Superheater

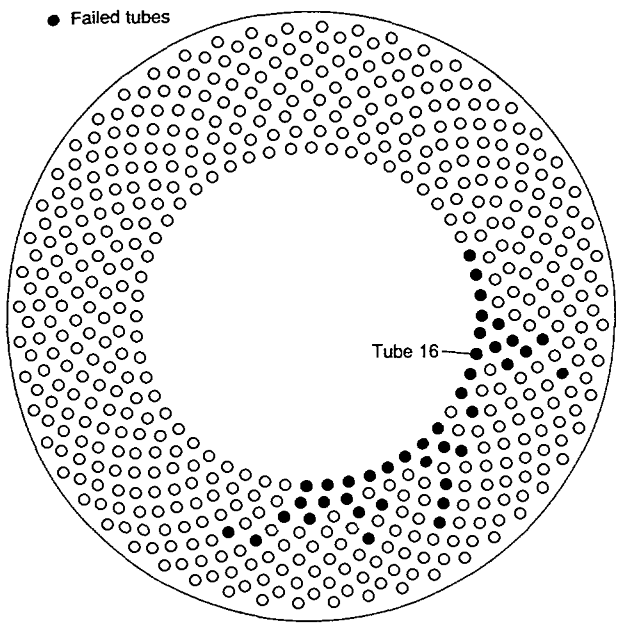


Figure 2.2 The Failed Tubes in PFR Superheater 2

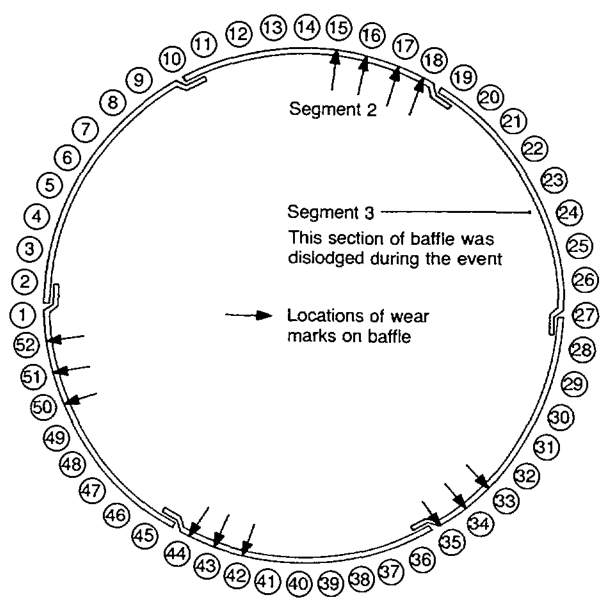


Figure 2.3 The Location of Fretting Marks on the PR Superheater 2 Central Baffle

5. The loss of internal cooling when the steam flow stopped and external heating by the sodium-water reaction made tube wall temperatures rise.

6. Reaction zone temperature increased above the boiling point of sodium. This caused high temperature tube failures at 1 325°C to 1345°C, even though the pressure in the tubes had fallen from 130 bar to 70 - 40 bar by this time.

7. As temperatures increased further more tubes would have failed but eventually the steam pressure dropped low enough to prevent further failures and tube swelling.

8. In spite of the large number of tube failures the pressure transient in the secondary sodium circuit was relatively mild. The maximum pressure in the intermediate heat exchanger did not exceed about 10 bars, well below its design pressure.

Conclusion

The under-sodium leak in superheater 2 demonstrated that it is possible for a large number of tubes to fail due to overheating in a period of a few seconds, but that such an event is unlikely to cause significant overpressurisation damage in the secondary circuit or the intermediate heat exchanger. As well as leading to modification of the PFR sodium-water protection system and replacement of the tube bundle, the incident led to a reassessment of the design-basis accident for the steam generators of both PFR and EFR. In the case of PFR the design basis accident was changed from a single double-ended guillotine fracture to 40 double-ended guillotine fractures spread over a period of 10 seconds.

REFERENCES TO SECTION 2

A M Judd, R Currie, G A B Linekar and J D C Henderson: "The Under-Sodium Leak in the PFR Superheater 2, February 1987": *Nuclear Energy*, 1992, **31**, 221 - 230

3. CRACKS IN PFR STEAM GENERATOR VESSELS

From 1983 until its shutdown in 1994 PFR experienced cracking in type 321 stainless steel components in its secondary circuits, some cracks leading to sodium leaks. As a result a substantial repair and inspection programme was required in the final seven years of PFR operation. Although the two earliest leaks were in pipework (in 1983 and 1986) the majority were in steam generator vessels. The pipework leaks were only retrospectively identified as being caused by the same mechanism.

Figure 3.1 shows a typical steam generator vessel. The vessels were manufactured from cold rolled annealed type 321 (18Cr 10Ni 1Ti) austenitic steel plate. Three cylindrical courses, together with a flanged upper end and a domed lower end, were welded using type 347 (19Cr 9Ni 1Nb) weld metal. Manual metallic arc weld root runs with submerged arc weld fill were used. During manufacture the welds were inspected and in some cases inclusions and other defects were found. These were ground out and made good with additional rectification welds. None of the welds was stress-relieved. The vessels were considerably overdesigned to withstand a continuous pressure of 34.5 bar, although the operating pressure was only 2 bar.

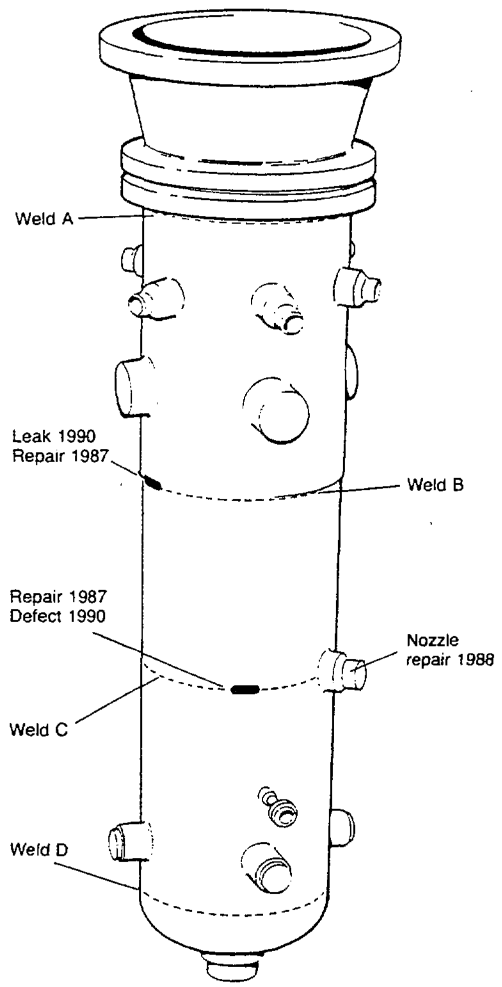


Figure 3.1. PFR Reheater 1 Vessel

The first crack occurred on a reheater pipework tee junction in 1983, and the second on a superheater pipework reducer in 1986. Both caused minor sodium leaks. The cause of the cracks was not identified at the time. Both leaks occurred at welds in which the large crystal grain size indicated severe overheating during fabrication.

In 1987 the first steam generator vessel leak occurred in seam B of the reheater 2 vessel. Following this cracks were found on vessels during every inspection except 1989. All the cracks occurred in the B, C and D welds with none in A. Seam A is in the gas space, above the sodium level, and therefore at much lower temperatures than the other seams. (Following the fitting of the new tube bundles in 1987 weld A also operated under sodium.) At the final inspection in 1993 cracking was found for the first time in the welds of set-on features such as the vessel supports and nozzles.

Detailed optical and scanning electron microscopic examination of samples cut from reheaters 1 and 2 and superheater 2 were carried out in 1987. In all cases the cracks were associated with deep weld rectifications in the circumferential weld seams. The cracks either passed along the fusion line of the rectification or were on or close to the fusion line of the original rectified weld. Cracks passed through the parent steel along or closely parallel to the fusion line and through the weld material. Cracking was intergranular.

The cracking mechanism was identified as delayed-reheat or stress-relief cracking. Titanium stabilised type 321 steel is subject to this form of cracking, caused by dissolution of titanium carbide close to a weld fusion line as a result of the high temperatures attained during welding. Subsequent re-precipitation of the carbides on dislocations produced by weld shrinkage during high temperature operation locks the dislocations and hardens the matrix. Relaxation strains occurring during operation causes intergranular failure.

In total some 27 cracks were detected on such welds during the period 1987 to 1993. Inspection was by ultrasonics and dye penetrant supported by acid etching of the surface. In general there was no sign of major escalation of the weld cracking problem during the period of monitoring. It is possible that cracking would have tailed off as the population of vulnerable sites diminished.

Repairs were made by removing the cracked region and welding on a stub nozzle with a blanked end. The technique is illustrated in Figure 3.2. It was chosen because it allowed all the work to be done from outside the vessel (so that it was not necessary to open the vessel to the atmosphere, which would have required scrupulous removal of all the sodium residues), and because it facilitated thorough ultrasonic inspection of the repair. Smaller defects were either ground out or backfilled and fitted with strain gauges for monitoring during operation. The PFR vessels were considerably overdesigned and although the excessive vessel thickness may have contributed to the cracking it was helpful when it was necessary to grind out defects.

Conclusion

The evidence indicated that cracking in PFR steam generator vessels was initiated by a delayed reheat mechanism driven by residual stresses in the non-stress-relieved welds. Weld rectification during manufacture gave rise to conditions which favoured cracking. It is

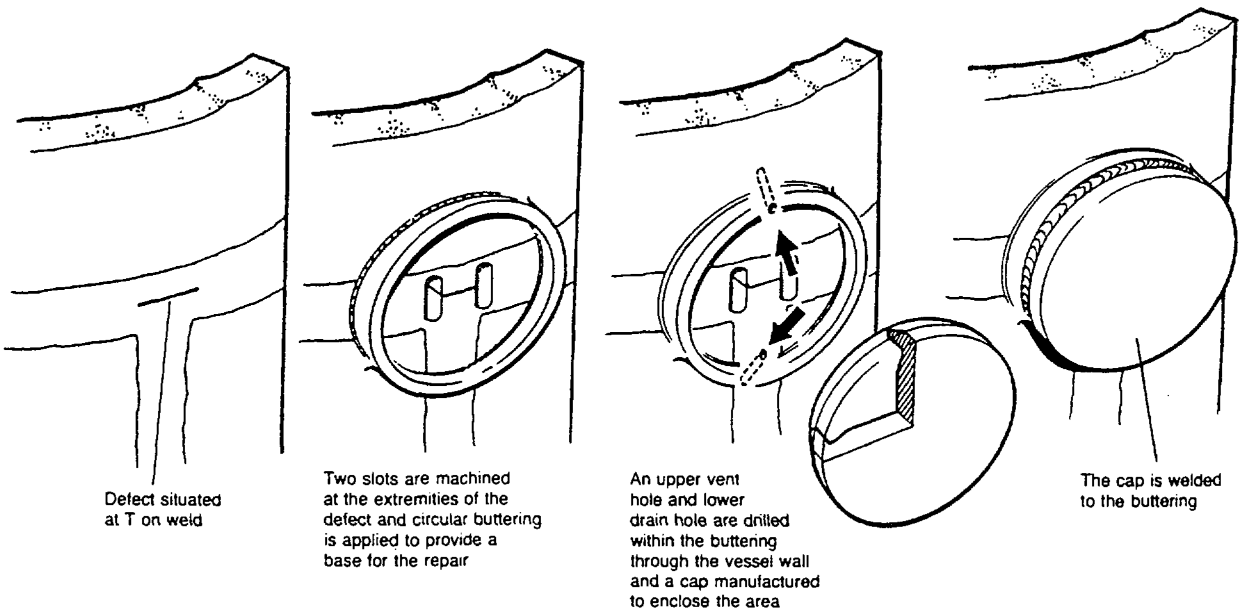


Figure 3.2. The Repair Technique for Cracks in the PFR Steam Generator Vessels

probable that replacement of the PFR steam generator vessels would have become necessary had operations been planned beyond 1994, as the repairs did not prove to be entirely satisfactory, with the repair welds beginning to develop cracks after a period of operation.

REFERENCES TO SECTION 3

D B Melhuish and A Sandison: "Engineering Improvements to PFR": *Nuclear Energy*, 1992, **31**, 193 - 205

4. BLOCKAGE OF THE PFR SECONDARY COLD TRAP

The PFR secondary cold trap vessel and basket are shown in Figure 4.1. Secondary sodium, partially cooled in a regenerative heat exchanger, was delivered to an annular chamber. From there it was injected through six 22 mm diameter holes into an annular space between the vessel and a mesh basket. The vessel was air-cooled and the intent was that the sodium should be cooled to about 20 °C below its current impurity saturation temperature before flowing through six annular "doughnuts" of steel wire mesh. The mesh presented a large surface area for the deposition of sodium oxide and hydride. Cold tap baskets were regularly removed and cleaned for re-use, fitted with new set of doughnut meshes, when a gradual reduction in the sodium flow as the mesh blocked up indicated that the maximum loading had been reached. Expected loadings were in excess of 100 kg of mixed sodium hydride and oxide.

The sixth basket was installed in July 1980 and had completely blocked by May 1981 with an estimated loading of only 52.3 kg of mixed oxide and hydride. Since this was not the first example of erratic cold trap behaviour it was decided to remove the basket for a special examination to ascertain the reasons for early blockage. Photographic records had been kept of earlier removals but until then no systematic detailed examination of the distribution of the deposits in the mesh had been made.

The deposits are pyrophoric and present a hazard that prevented close examination in air. On this occasion the trap was removed and kept under argon purge while an introscope examination was carried out. This gave an initial picture of the nature of the deposit before it was disturbed by being broken up.

The results of the examinations are shown in Figure 4.2. The deposits in the mesh were mixtures of sodium, sodium oxide and sodium hydride of varying composition, while the deposits on the sides and bottom of the vessel consisting virtually entirely of residual sodium. Detailed examination of the mesh doughnuts showed that the deposits were largely confined to the inner part of the annulus and the bottom, which was completely blocked.

Subsequent analysis suggested that the cold trap had not always been operated in the most efficient manner. Attempts to trap impurities too rapidly had led to the use of too large a differential between the actual sodium temperature and the impurity saturation temperature. The overcooled sodium precipitated its impurity burden preferentially at the bottom of the vessel and on the outer surfaces of the mesh doughnuts, so that the trap rapidly became blocked with an impurity loading well below the expected maximum. Modified operating techniques and led to improved loading and the final cold trap basket trapped in excess of 140 kg of mixed oxide and hydride.

Conclusion

Inefficient operation of the PFR secondary cold trap led to impurity loadings below the theoretical maximum. Improved trapping techniques aimed at avoiding cooling the sodium too far below its saturation temperature, and more careful operation solved the problem after 1981.

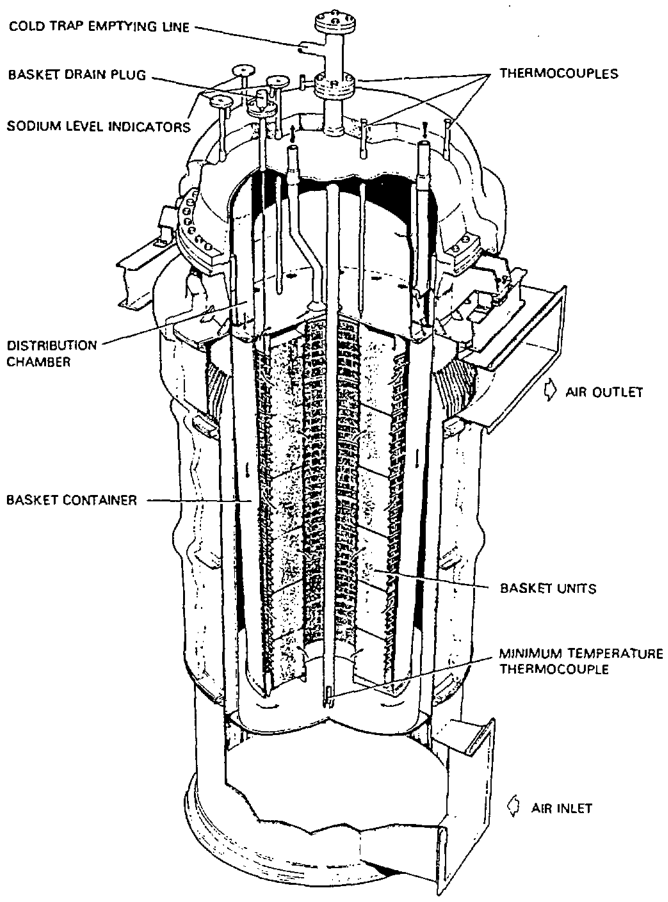
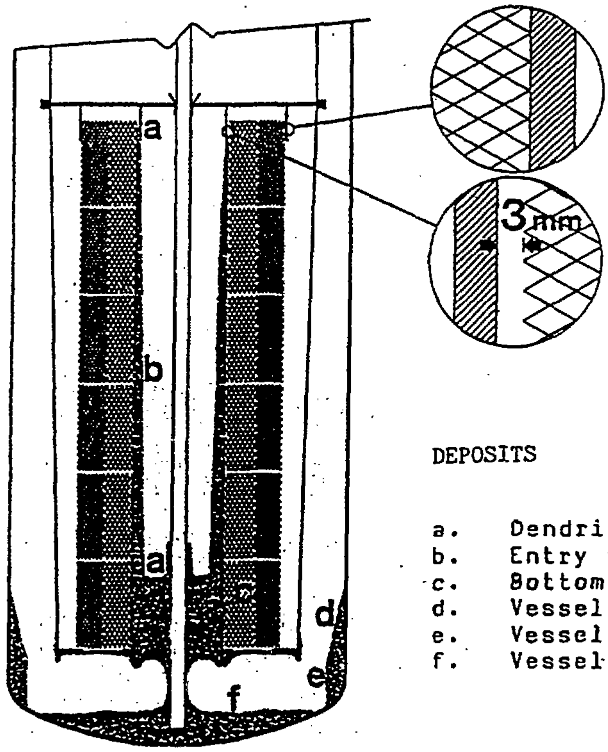


Figure 4.1. The PFR Secondary Sodium Cold Trap



DEPOSITS

a. Dendritic crystals.
b. Entry face - 68 kg
c. Bottom plug - 3 Kg
d. Vessel wall top - 40 Kg
e. Vessel wall bottom - 25 Kg
f. Vessel bottom - 6 Kg

Figure 4.2. The Location of Deposits in the PFR Secondary Cold Trap

5. SEIZURE OF THE PFR PRIMARY COLD TRAP LOOP SODIUM PUMPS

The PFR primary cold trap loop (PCTL), shown in Figure 5.1, is an auxiliary circuit connected to the primary circuit. It purifies the active sodium by removing impurities. During PFR operation it also supplied sodium coolant to the core melt-out tray situated below the diagrid in the reactor vessel. During the current decommissioning period it remains in use for cold trapping, if necessary, and measuring impurity levels in the primary sodium. The PCTL is situated in a shielded air-cooled concrete vault adjacent to the reactor vessel, as shown in Figure 5.2.

Throughout the life of the reactor both the main and standby PCTL pumps have seized frequently due to sodium rising too high in the pump vessel. The pump vessel sodium levels are controlled by venting gas from the space above the sodium or injecting gas into it. In principle the sodium level was raised or lowered in 10 mm steps by operation of the gas valves in an automatic sequence. In practice, however, it was often necessary to intervene manually because the gas valves often passed or were blocked. In these circumstances operator error could easily lead to the sodium level into the annulus round the pump shafts where it would solidify causing the pump to seize.

Each time a pump seized attempts were made to remove the sodium mechanically or melt it by use of the trace heating, but on no occasion did this succeed. Pumps had to be removed, decontaminated and stripped down in order to free the pump shaft.

During maintenance it was noted that oxide or hydride could be seen floating on the surface of the sodium in the pump tank. It was considered that such material, if raised inadvertently into the vent or feed lines, would contribute to the blocking of the gas valves. In 1988 both main and standby pumps had seized during the year, so during refurbishment a modification was carried out on the pumps to deal with the accumulation of these solid impurities. A jet of sodium was diverted from the discharge sides of the pumps to disperse the deposits and prevent valve blockages, as shown in Figure 5.3. This modification was not entirely successful in preventing further blockages and the pumps seized again in 1989 and 1990.

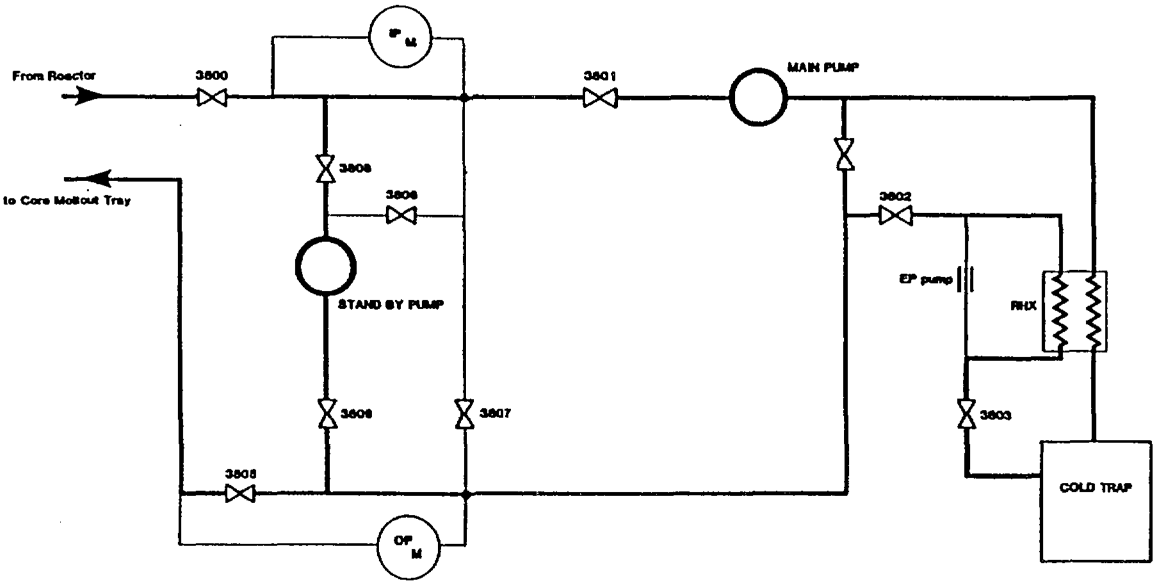


Figure 5.1. Schematic diagram of the PFR Primary Cold Trap Loop

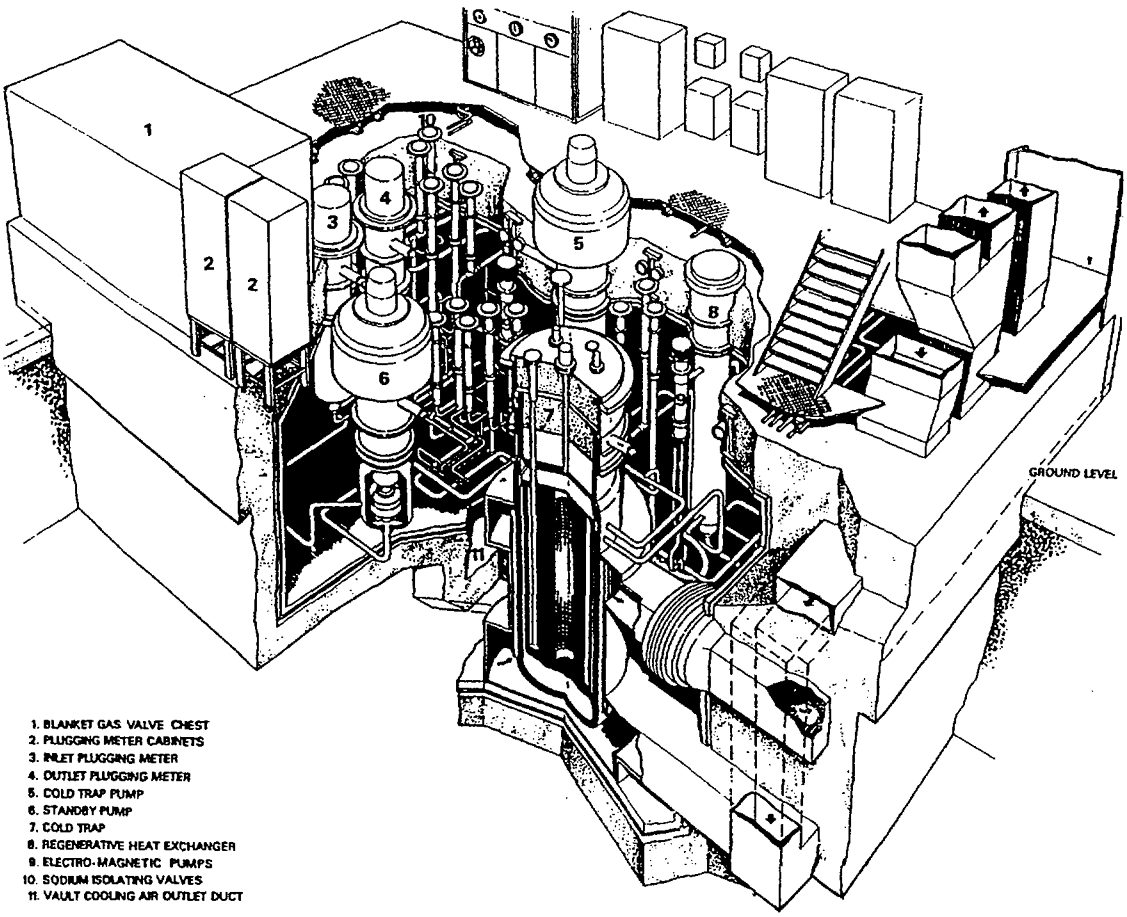


Figure 5.2. The PFR Primary Cold Trap Loop

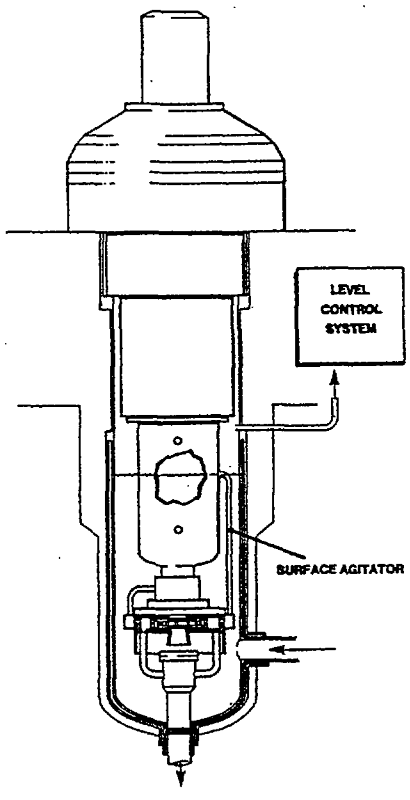


Figure 5.3. Modification of the PFR Primary Cold Trap Pumps

Conclusion

Sodium level control problems in the PCTL have frequently required stripdown and maintenance of the sodium pumps to clear sodium from the pump shaft annulus. The problem has never been fully resolved.

6. THE EFFECT OF SODIUM AEROSOLS ON THE OPERATION OF PFR

Sodium aerosols have been reported as a source of problems in all fast reactors. In PFR two particular problems arose during operation between 1974 and 1994.

A diagram of PFR absorber rods is shown in Figure 6.1. PFR had 5 shut off rods (normally fully raised) and 5 control rods inserted to control power. The rods were essentially identical B_4C assemblies supported by electromagnets. On a trip all ten rods dropped. Magnet current, apparent rod weight, rod release time and time of flight were measured by installed instrumentation.

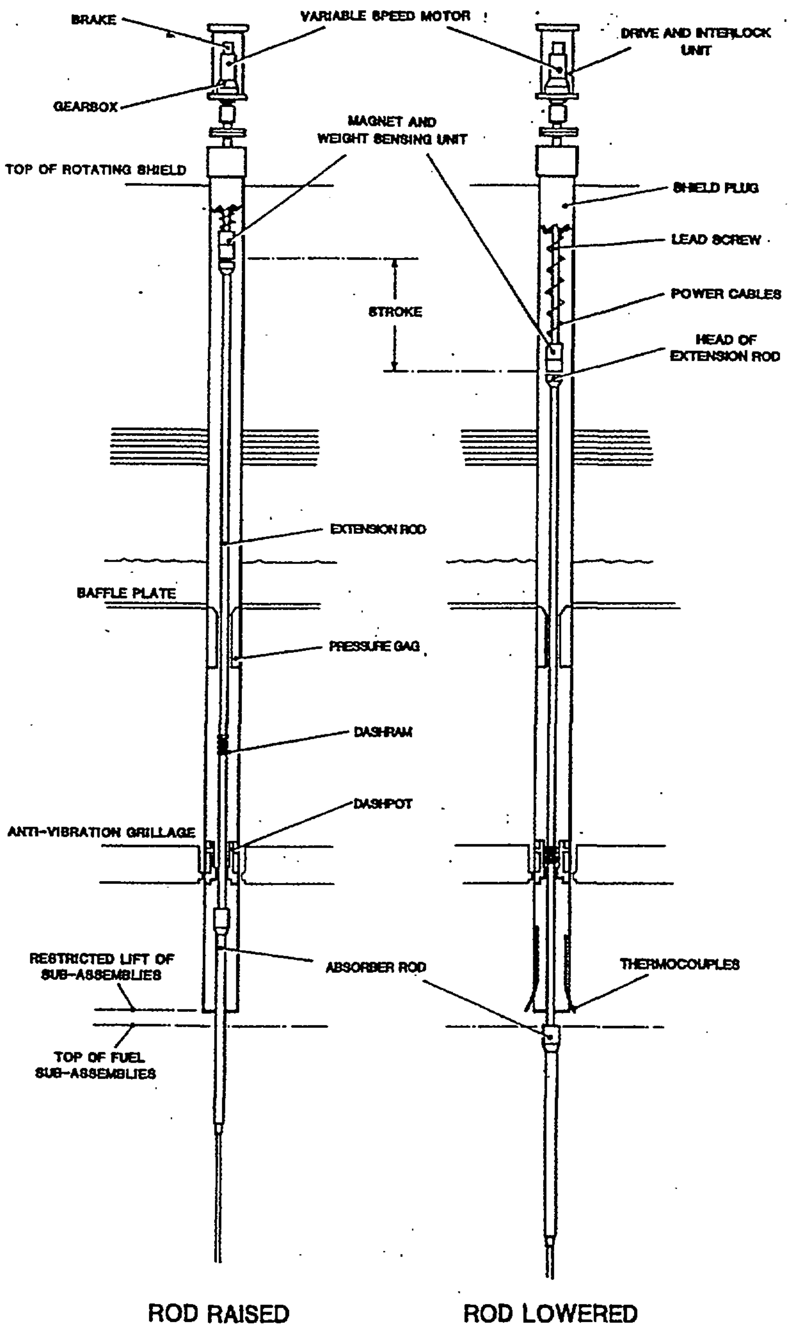


Figure 6.1. The PFR Absorber Rods

At the design stage it was recognised that following a trip there was potential for sodium aerosol to deposit on the parted magnet faces in the gas space. A continuous purge flow of argon gas was passed through the absorber liner tube to avoid the problem. Following trips the flow was enhanced to try to prevent aerosol building up. This however did not solve the problem fully and throughout its operational life PFR suffered from gradual reduction in the efficiency of the magnets due to the gradual buildup of sodium deposits on the faces of the magnets. This led to plant trips on a number of occasions due to absorbers dropping off their magnets during power operation.

At all shutdowns and after plant trips the electromagnet pick-up and drop-off currents were measured. These were the minimum magnet currents at which the absorber could be raised and at which it dropped off after being raised. On the basis of these figures a decision was made on whether the magnet faces had to be cleaned before return to power. If required the drive and magnet assembly were removed by simple bagging techniques and the magnet face was cleaned in an argon purged glove box. The extension rod face was cleaned in situ using commercial "Scotchbrite" cleaning pads, again making use of a simple bagging technique.

Because of the possibility of distortion due to swelling caused by neutron-induced voidage (NIV), which caused interaction between the rods and the guide tubes, absorber friction was measured on a regular, but initially infrequent, basis. Each rod in turn was raised and lowered while the reactor was operating (criticality being maintained by moving the remaining rods to compensate). The apparent weight of the rod, which varied along the stroke, was recorded. Up to 1985 measured friction was higher than expected, at a maximum of 40 kg compared with an expected level of about 25 kg due to the effect of NIV. The additional load was thought to be caused by the effects of aerosol but was not particularly worrying.

After 1985 when prolonged high power operation became more common friction levels were found to increase rapidly to about 80 kg. At such levels, coupled with the effect of magnet face contamination by aerosol, rods were liable to pull off and drop as they were being raised. Regular weekly exercising of the rods at intervals of seven days was initiated to monitor this effect.

Typical rod movements during exercising are shown in Figure 6.2. The exercising was found to have the effect of reducing friction. Peak friction occurred at about 900 mm (see Figure 6.2) but the magnitude depended on whether the rods were raised or lowered first. Lowering first reducing friction levels. Figure 6. 3 shows an actual trace of friction for a control rod, which in this case started by being raised. The actual weight of absorber plus extension tube is 230 kg. The dotted line marked FINISH is an immediate partial repeat of the test showing the decrease in friction resulting from the first test.

The friction is believed to have been caused by the buildup of sodium aerosol deposits on the inside of the liner tube and on the extension rod of the absorber in the cool region of the rotating shield, as shown in Figure 6.4. Lowering the rod moved the deposit on the guide tubes into the warmer region below the roof insulation where it melted off, reducing the friction.

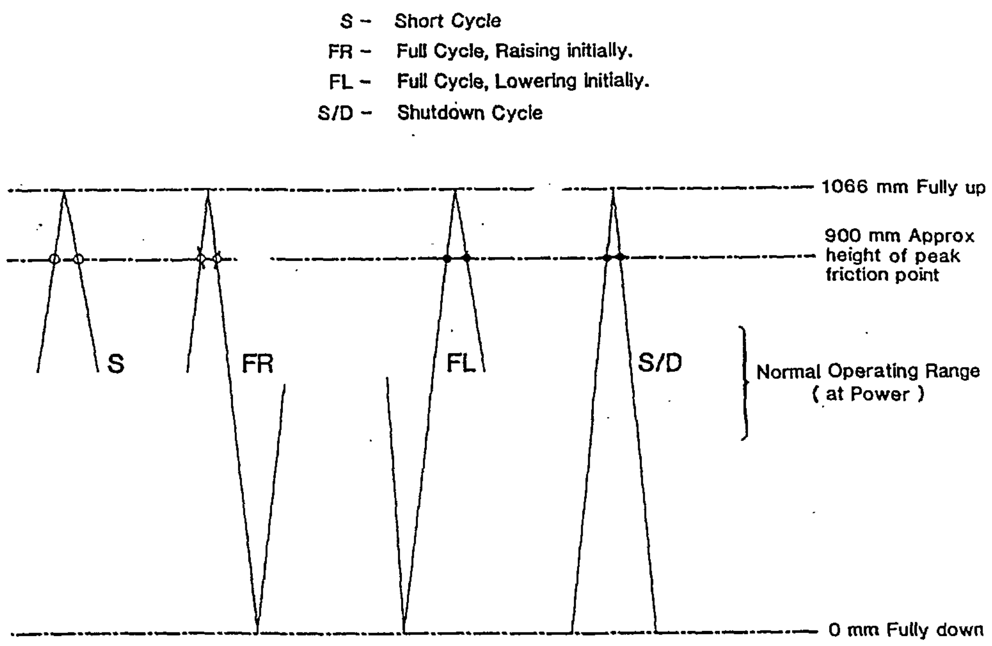


Figure 6.2. Movement of the PFR Absorber Rods during Exercising

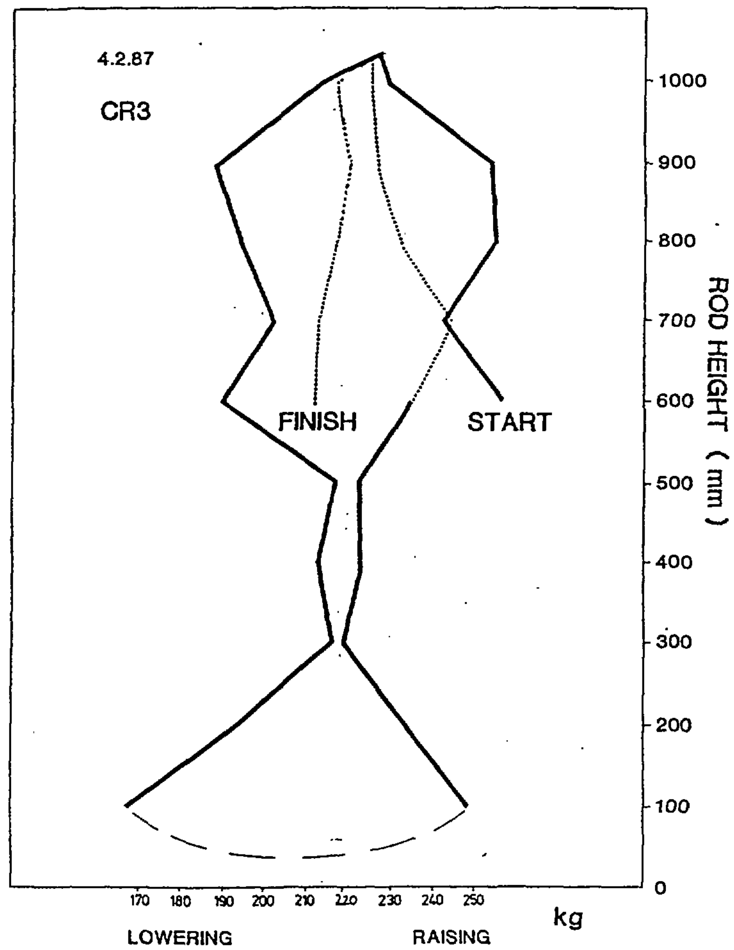


Figure 6.3. Apparent Weight of a PFR Absorber Rod during Exercising

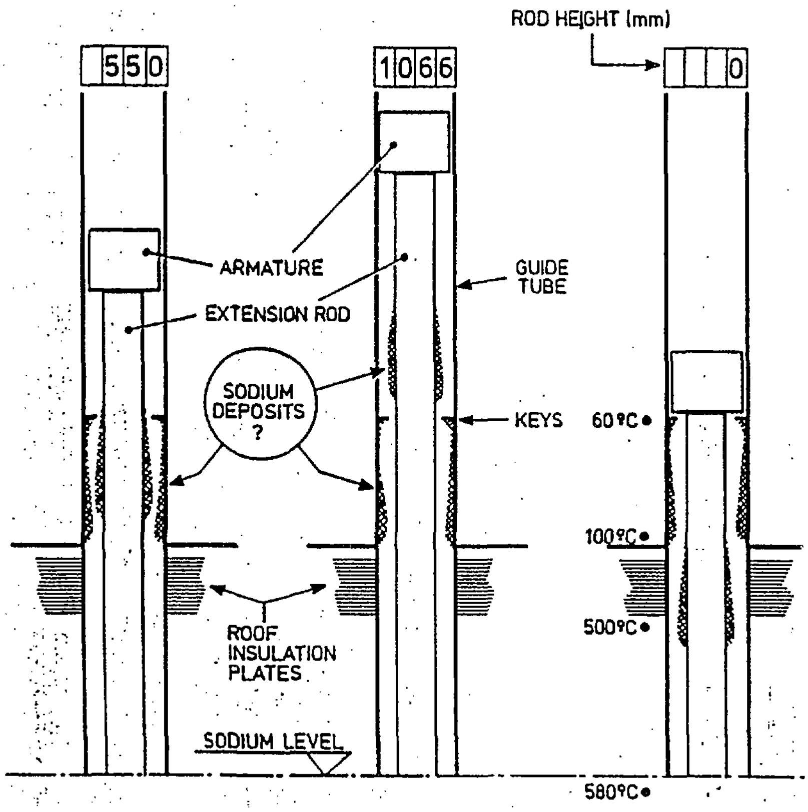


Figure 6.4. Location of Sodium Aerosol Deposits on a PFR Absorber Rod Mechanism

In 1988 a special glove box was made which allowed examination of the liner tubes and the extension rods. Examination of a number of rods confirmed that sodium deposits were present but in smaller quantities than expected and confined to the keyway of the extension rods. None were found on the liner tube as originally hypothesised. The sodium was soft and easily removed. Although the absence of deposits other than in the keyways was surprising, when they were removed the friction of the restored rods to normal. It took some 40 efpd of operation for friction levels to begin to rise noticeably.

Conclusion

In the case of PFR sodium aerosols caused no operational problems because movement of the absorber rods was carefully monitored and deposits were cleaned off well before they interfered with the mechanisms. The only effect was the operational burden of exercising the rods and cleaning the magnet faces. Aerosols had no observable effects on magnet parting times or rod drop times.

7. CRACKS IN THE PFR AIR HEAT EXCHANGERS

PFR had three thermal syphon decay-heat rejection loops, shown in Figure 7.1. Each consisted of a NaK-filled loop connecting a heat exchanger coil positioned in the main reactor vessel adjacent to an intermediate heat exchanger to an air heat exchanger (AHX) on the roof of the reactor containment building. In the event of loss of electric power supplies each loop was capable of removing 1.5 MW of decay heat from the reactor by natural convection. Each AHX was equipped with 2 fans connected to emergency diesel power supplies, which could enhance the decay heat removal to over 4 MW per loop. When the rector was operating normally heat removal was limited by dampers which restricted the airflow to the AHXs.

Although the thermal syphon system operated well, by 1984 it had become apparent that the AHXs suffered from a systematic fault leading to failures and leaks. Each AHX consisted of forty serpentine parallel tubes welded to pulled tees in two headers, as shown in Figure 7.2. The tubes were finned along the straight lengths but plain at the bends, which were clamped together and supported. Further rigidity was provided by cleats which were welded to the tops of the fins on adjacent tubes. Flow of NaK was from the top down. Leaks were occurred at the welds between the tubes and the pulled tees in the headers.

As an interim measure operational constraints were imposed as the frequency of failures could have invalidated the risk analysis in the safety report, and hence jeopardised the authorisation to operate the plant. Meanwhile the AHXs were heavily instrumented with strain gauges and thermocouples to identify the cause of the problem and indicate a solution.

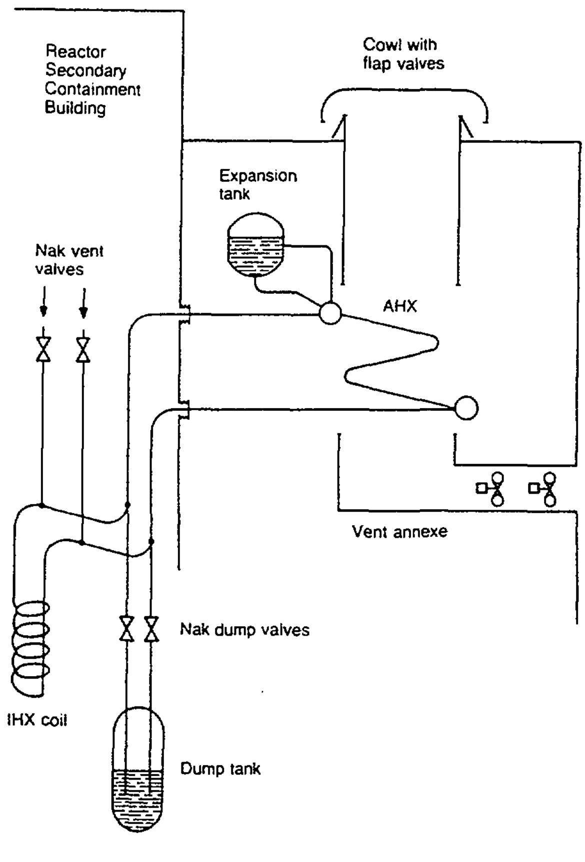


Figure 7.1. Schematic Diagram of a PFR Thermal Syphon Decay Heat Rejection Loop

The measurements indicated that the problems occurred essentially because the AHX tubes were in parallel, and were horizontal with no fall to ensure good filling. When the AHXs were filled gas locks were occurring at the pipe bends. The gas-locked tubes remained cold, and as a result oxide impurities could be precipitated causing permanent blockages. Because of temperature differences between a cold blocked tube and the adjacent hot tubes to which it was clamped, large stresses were imposed. As a result the weakest point in the system, the weld between the tube and the header, was stressed, suffered cracking and eventually leaked.

Replacement AHXs (RAHXs) were manufactured to an improved design which avoided the problem of gas locks and afforded greater toleration of loss of flow in individual tubes. The following changes were made and are shown in Figure 7.2.

1. A 2° slope was given to the tubes to give better venting and drainage.
2. Each tube was given individual support,
3. The tube-header connections were reinforced.
4. Larger diameter headers were fitted to give better NaK distribution.

Two of the new RAHXs were fitted in 1986 and the third in 1987. Operation was trouble free until 1996, when the thermal syphons were finally emptied for decommissioning.

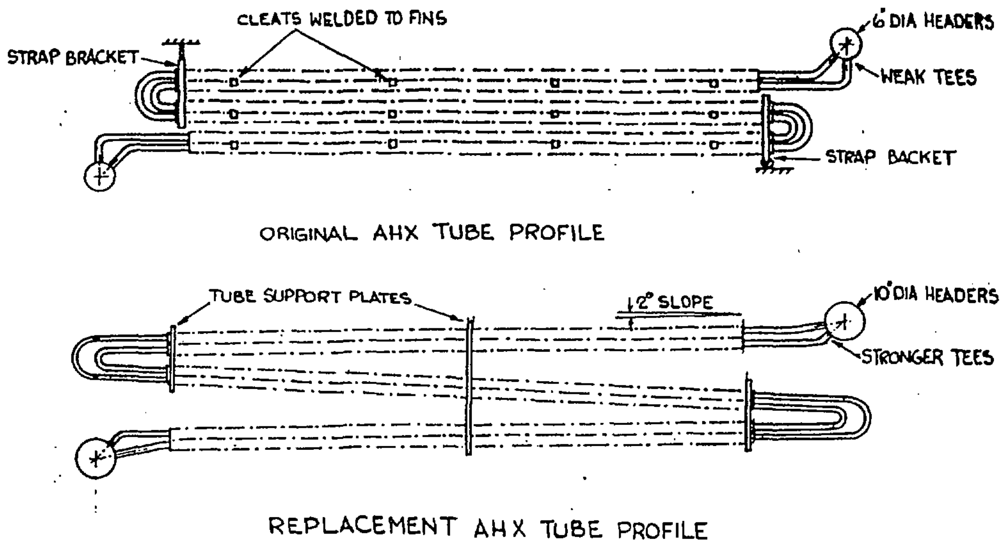


Figure 7.2. The Original and Replacement PFR THermal Syphon Air Heat Exchangers

Conclusion

In 1984 a common mode failure problem in the PFR thermal syphon AHXs was jeopardising the plant authorisation. A rapid research, development, manufacturing and installation programme solved the problem by 1986. It is notable that the design feature essential to solving the problem was the inclusion of a simple 2° slope on the tubes.

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8. THE EFFECT OF NEUTRON-INDUCED DISTORTION ON THE OPERATION OF PFR

Radiation damage resulting from the high neutron fluxes and operating temperatures of a fast reactor can give rise to dimensional changes in core components. The mechanisms involved are swelling caused by neutron-induced voidage (NIV), and radiation creep.

These phenomena affect core components by causing axial extension, bowing in transverse gradients of neutron flux or temperature, and dilation. NIV was first detected during post-irradiation examination of components from the Dounreay Fast Reactor (DFR) in 1965. PFR had been designed in 1963 without taking account of the need to accommodate the effects of NIV. In consequence calculation routes had to be developed to predict the distortion of PFR core components, so that they could be managed in such as way that operation would not be impeded. In particular it was essential to be able to ensure that no core component was at risk of becoming so distorted that it interfered with the movement of the absorber rods or could not be removed. The calculations, including the important effects of interaction between components, were based on empirical material deformation rules obtained from post irradiation examination of irradiated components. They were successful in guiding operations except when problems arose due to unexpectedly rapid growth of particular materials.

It was necessary to predict the bowing of fuel subassemblies in order to prevent handling problems. The operating limit was 14 mm bow at the subassembly shoulder. Bows beyond 21 mm at the subassembly shoulder would have presented difficulties when it came to extraction from the core. Subassemblies were routinely rotated through 180° part way through their residence in the core in order to correct the bowing.

Immediately before refuelling operations in PFR three sweep arms were employed to ensure that there were no obstructions above the core which would prevent rotation of the rotating shield, as shown in Figure 8.1. At the start of a reload in 1988 the sweep arms were found to contact or partially contact objects in two core positions. These positions were identified as containing subassemblies with cold-worked EN58B steel wrappers, with a calculated dose of greater than 60 displacements per atom (dpa).

Using a special tool the heights of all subassemblies of the same material were checked. A distinct trend of rapid increase of growth at doses above 50 dpa was revealed as shown in Figure 8.2, although not all subassemblies were affected. Two subassemblies in particular, "JRA" and "GYN" (see Figure 8.2), had measured growths of about 40 mm. As a result all components with predicted doses likely to exceed 50 dpa by the end of the next run were removed from the core. Considerable difficulty was experienced in handling the severely distorted components and special tools had to be manufactured for their extraction and removal from the reactor.

Although the absorber rods and associated components in PFR were manufactured from nimonic PE16, an alloy known to be subject to low swelling, it was important to ensure that NIV distortion would not prejudice operation of the system or hinder rod drop in a SCRAM. The major cause for concern was distortion of the guide tube in which the absorber rod moved, either by NIV bowing or by pressure on it from adjacent bowed fuel subassemblies. In addition to the calculations, regular exercising of the absorber rods over their full stroke gave assurance that no such problems were arising.

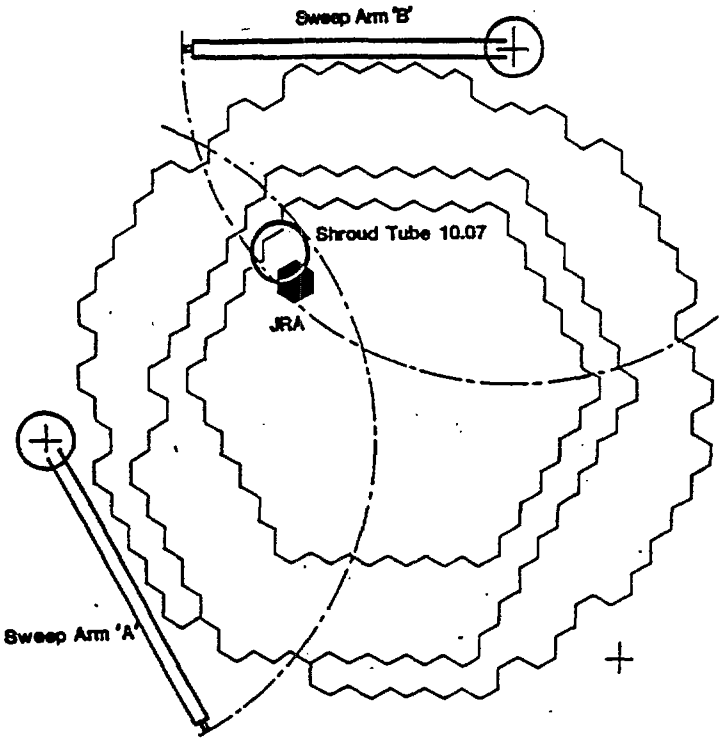

Figure 8.1. Location of an Elongated Subassembly in the PFR Core by the Sweep Arms

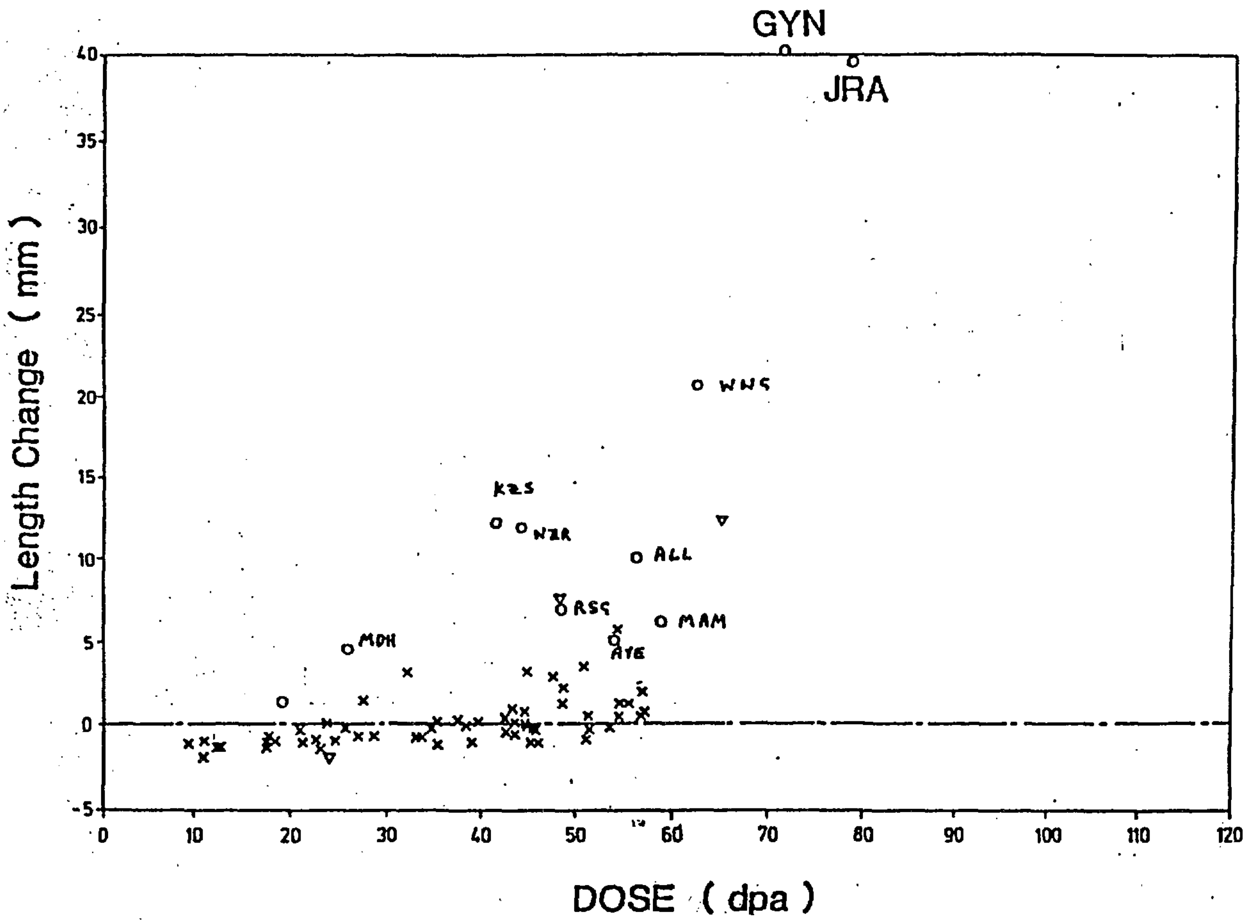

Figure 8.2. Increase in Length of EN58B Subassemblies in the PFR Core

On only one occasion were there observable effects in the operation of an absorber. In this instance a shut off rod developed unusually high and increasing friction at the top of its stroke while being exercised. Although the rod operated correctly during subsequent trips, calculations indicated that the problem was probably caused by interaction of the guide tube with an adjacent distorted fuel subassembly. The subassembly was discharged, and PIE confirmed the analysis. It was another subassembly clad in cold-worked EN58B, with higher than expected swelling. The allowed doses for EN 58B was reduced to prevent further problems of this sort.

Conclusion

Large differences in NIV swelling rates could occur in different batches of the same material. This led to handling problems in the case of components made of cold-worked EN58B. Materials chosen later in the lifetime of PFR, such as nimonic PE 16, had considerably lower swelling rates. Components manufactured from the ferritic steel FV 448, which was under test at the time of PFR closure, had extremely low swelling rates. NIV distortion was not expected to be life-limiting for this material.

9. THE PFR PRIMARY CIRCUIT OIL SPILL

The PFR primary sodium circulation system is shown in Figures 9.1 and 9.2, and Figure 9.3 shows details of a primary sodium pump (PSP). The sodium from each PSP flows through filters and a stop valve to the diagrid, and thence to the fuel subassemblies. Each subassembly has a filter at its inlet. Figure 9.4 shows the relationship between the pump and subassembly filters.

In 1974 primary sodium pump 2 (PSP 2) was removed from the reactor for modifications to its instrumentation and was noticed to be heavily contaminated by a black sooty deposit. In the same year the charge machine was removed, revealing that its immersed surface was black with adherent tarry lumps.

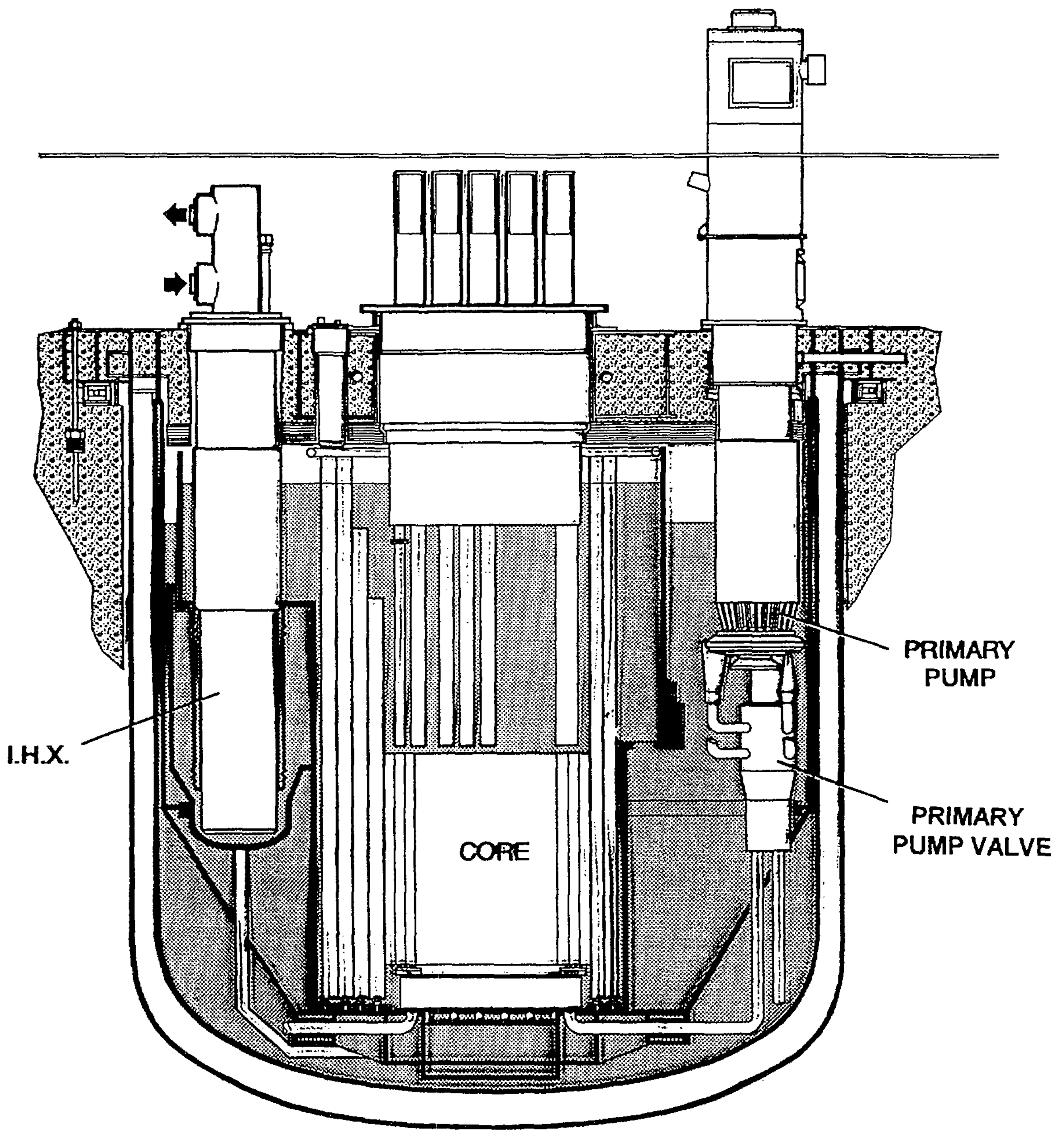

Figure 9.1. The PFR Primary Circuit

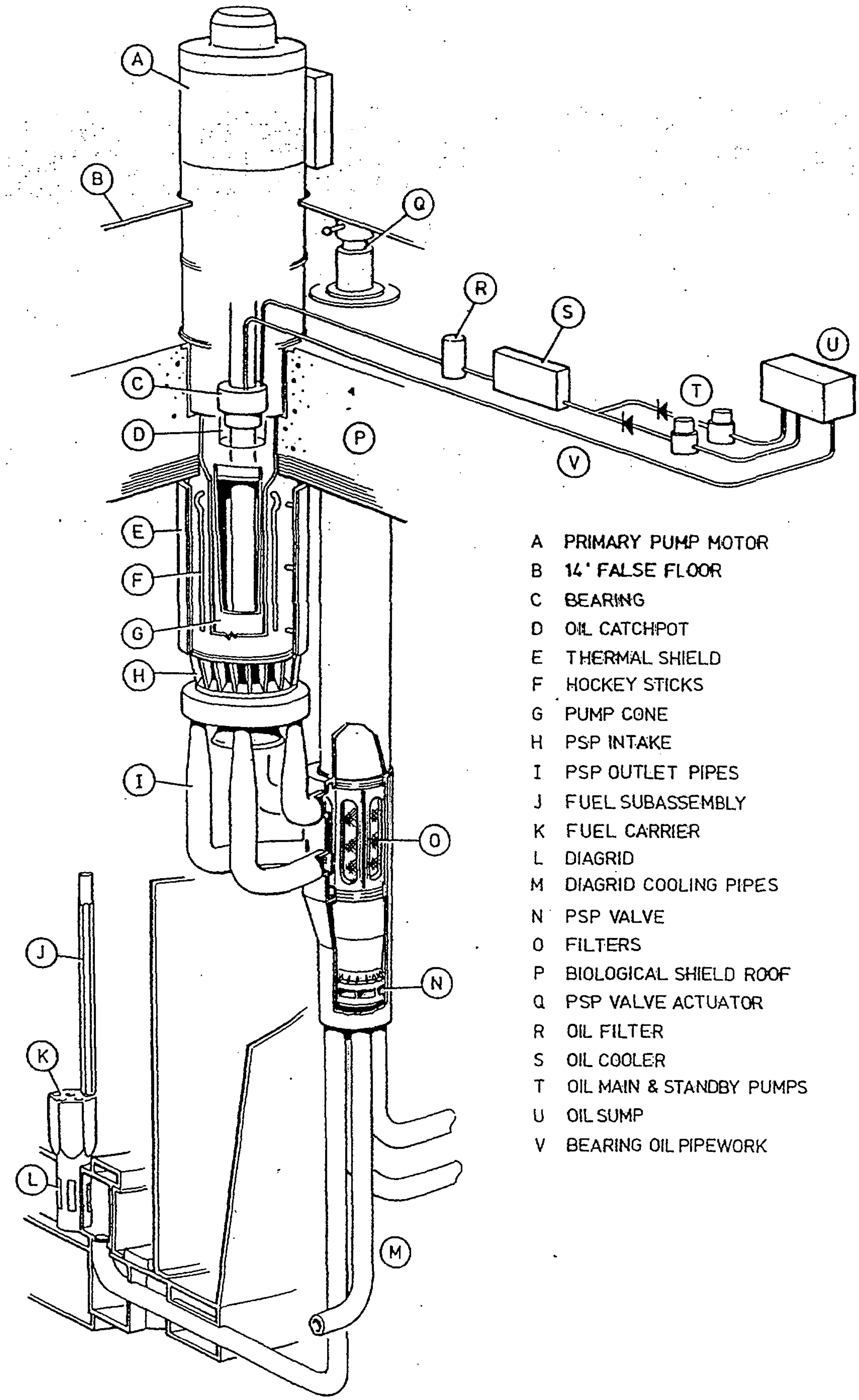

Figure 9.2. A PFR Primary Sodium Pump and its associated Valve and Filter Assembly

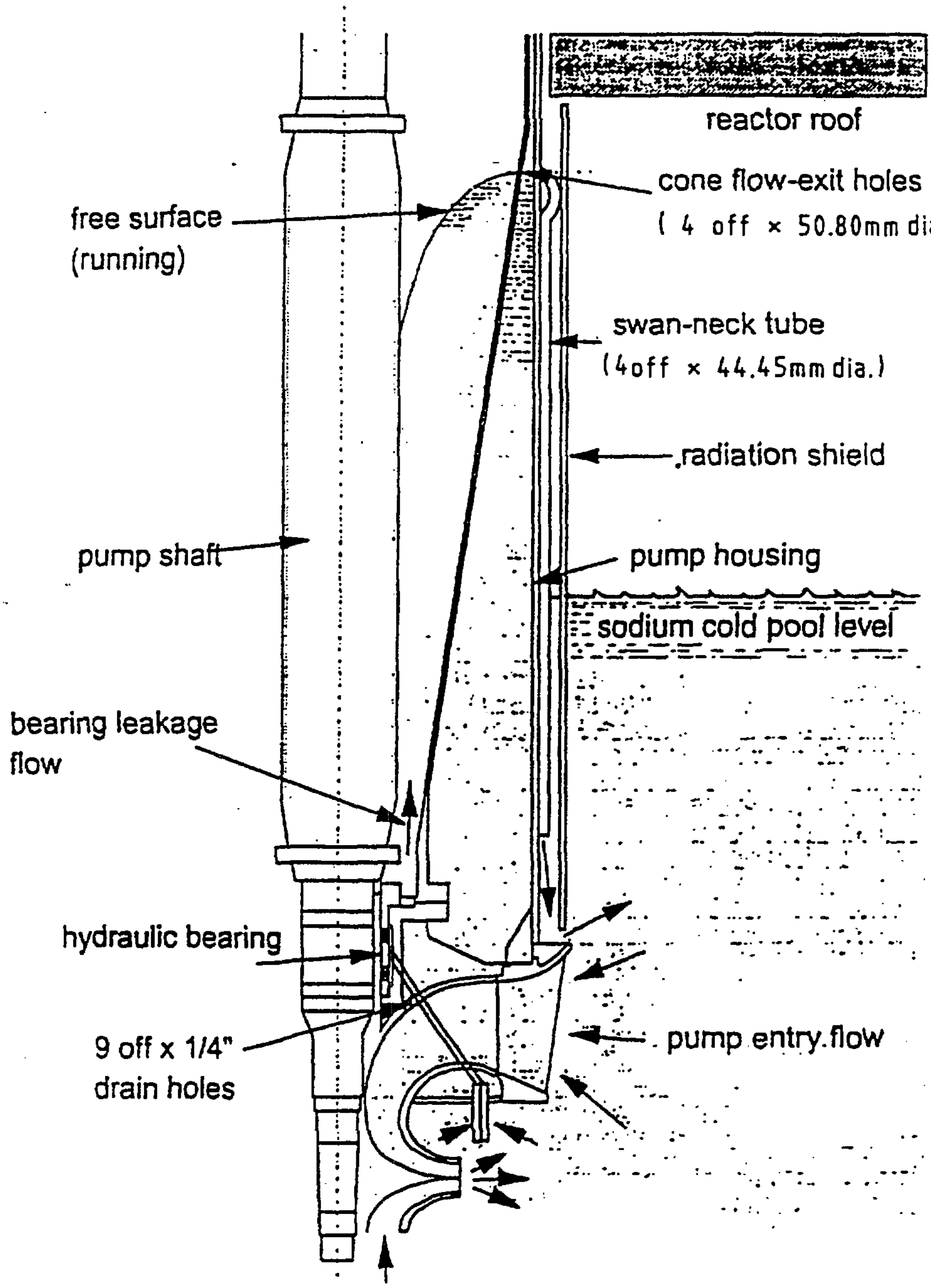

Figure 9.3. Detail of a PFR Primary Sodium Pump Housing

During this period of operation some 65 litres of oil had been lost from the pump upper seal oil systems, part of which is believed to have entered the reactor vessel. When the reactor was taken critical no effects of the oil were observed. It is suspected, however, that as a result of the spill the filter on PSP 2 valve ("O" in Figure 9.2) failed due to high differential pressure because it became blocked by oil-sodium reaction products. It is also thought that partial blockage of the pump casing overflow pipe (Figure 9.3) was to lead to the major problem in 1991.

No further problems were observed until 1990 when it was noted that PSP1 drive current was slowly dropping at constant pump speed and its discharge pressure was rising. The situation was under observation when suddenly the current rose, the discharge pressure

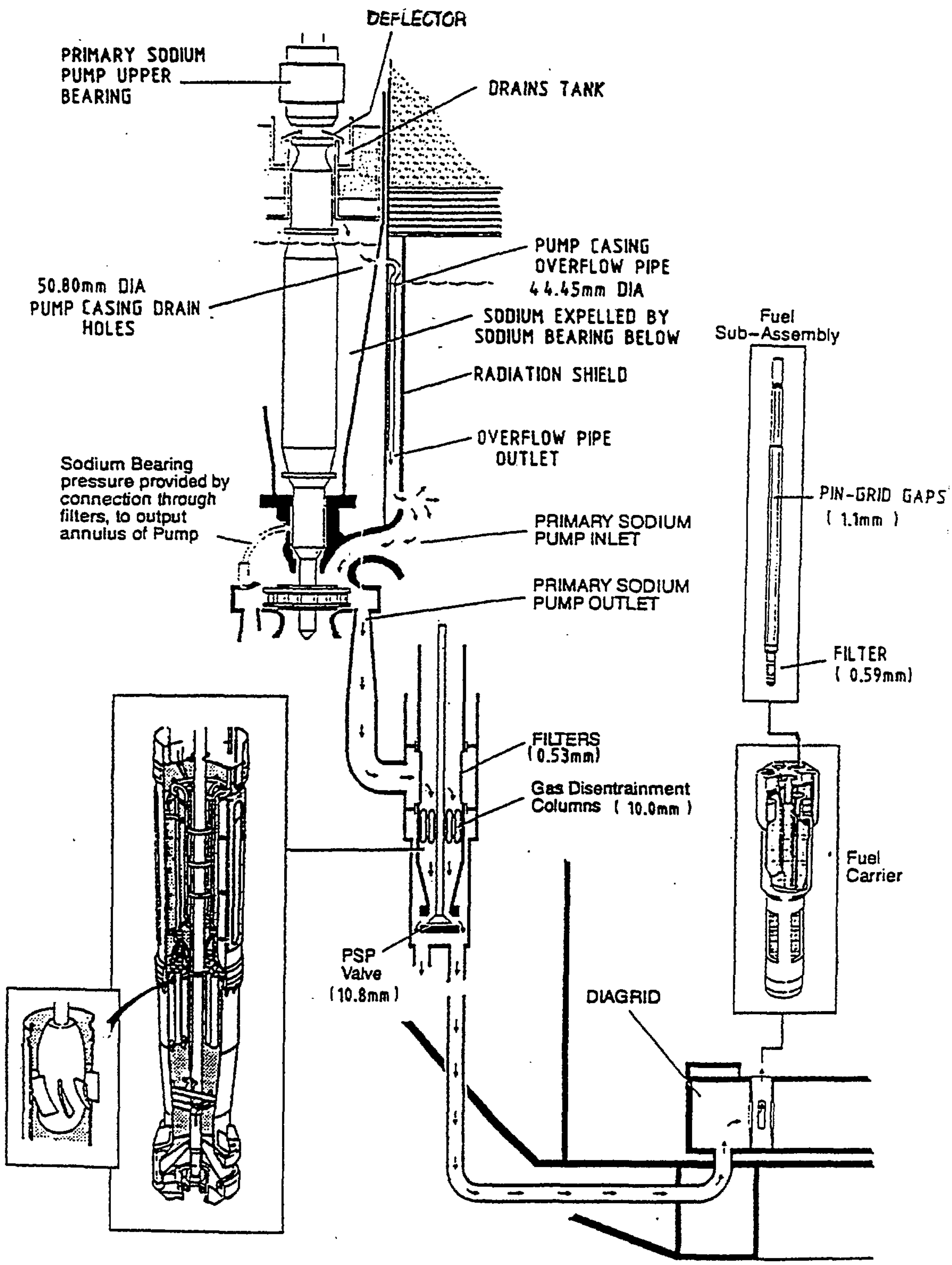

Figure 9.4. Filter Mesh Sizes in the PFR Primary Circuit

fell and the flow of coolant through the core increased. A similar sequence of events occurred a few weeks later, when flows and current returned to normal. Although no oil spill was recorded it is now believed that oil had entered the system and blocked the filter of PSP 1, causing it to fail in stages. When it had failed completely and offered no resistance the coolant flow returned to its normal value.

In 1991 a similar effect began to appear on PSP 3, and by the middle of the year the coolant flow was estimated to be 82 % of normal. Again no source of this apparent blockage is known but an oil spill is suspected.

On 24 June 1991 there was low flow in the argon gas blanket circulating system. During attempts to improve the flow by venting the gas blanket, high radiation levels in the PSP 2 well indicated that sodium had been raised into the top bearing drains tank. This was a result of the blocked overflow pipe causing a pressure differential between the gas blankets in the main vessel and the pump casing. The method used to vent the gas blanket resulted in a preferential flow of gas from the pump casing, reducing the pressure in it and forcing sodium up the pump shaft. On this occasion it is certain that oil was displaced from the drains tank into the reactor primary circuit.

Some subassembly core outlet temperatures in the sector of the reactor supplied by PSP 2 began to rise but stabilised after about 1.5 days. An accelerated increase in PSP 3 filter pressure differential was noted (by now of course PSP 3 filter was the only one intact). The plant was under close observation when on 29 June the oil bearing on PSP 2 failed completely causing a further oil spill, and the plant was tripped. It was observed from flow and differential pressure readings that PSP 3 filter failed at this time.

It is estimated that up to 17 litres of oil was released into the cone of PSP 2 during the June 1991 incidents. Release of oil debris from the pump cone into the main primary circuit was gradual, taking about 1.5 days as indicated by the increase of core subassembly outlet temperatures and the increase in the primary pump valve filter pressure drop.

A major effort was required to remove all three valve and filter assemblies from the reactor for examination. These were the longest components in the reactor vessel, at 12 metres, and required considerable care in handling. Examination showed that at least one panel of each valve filter had failed, and oil-related debris was found on all the filters. A number of the fuel subassemblies which had showed outlet temperature rises during the incident were removed, and oil-related debris was found on their inlet filters and wrappers.

The result of the oil ingress was an 18-month shutdown while PSP valve and filter assemblies were removed and new filters were fitted. The pump seal oil systems were modified to prevent any further possibility of oil ingress, and alarm and trip systems were added to prevent blockage of the pump filters in order to protect the subassembly filters.

Very fine particles of carbon were found in primary sodium samples after 1974. These are believed to have come from the 1974 oil ingress, and it appears that in the long term oil debris breaks down into finely-divided carbon particles which are dispersed in the sodium, pass through the filters, and circulate without obvious effect.

Conclusion

Oil ingress into the primary circuits of an LMFR is undesirable because of the potential release of methane gas through the core causing reactivity effects, and possible blockage of the subassemblies by solid carbon debris. In the case of PFR no reactivity effects were seen, possibly because the oil was retained in the pump cone for a prolonged period and broken down slowly without the formation of large bubbles. In the long term oil bearings are probably best avoided. The EFR design was changed to gas bearings for its PSPs following the PFR oil ingress incident.

ACKNOWLEDGEMENT

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